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THE OAK RIDGE RESEARCH REACTOR – SAFETY ANALYSIS

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OAK RIDGE NATIONAL LABORATORY

operated by

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F. T. Binford

With Appendixes by

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MARCH 1968

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CONTENTS

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PREFACE

The original description and safety analysis of the Oak Ridge Research Reactor[1] was published in 1954. Although the basic design has remained unchanged, this report referred to a 5-Mw version of the reactor. The safety analysis was subsequently updated by a series of letters and documents, which have been summarized in Appendix A. On the basis of these and the original safety analysis, routine operation of the ORR at power levels up to 30 Mw was authorized by the USAEC on May 22, 1958. Operation at power has continued since that time with no serious difficulties or interruptions.

The present analysis of the consequences of potential accidents to the ORR is the second of two documents which may be considered to be a current safety analysis of the reactor. The first of these, *The Oak Ridge Research Reactor – A Functional Description*, by T. P. Hamrick and J. H. Swanks, will be published as ORNL-4169 (Vol. I). It presents a reasonably detailed description of the ORR and its ancillary facilities. Although the reactor is currently operating at a power level of 30 Mw and is expected to remain at this power level in the foreseeable future, the safety analysis has been based upon operation at 45 Mw because of the possibility that an increase in power to this level may be contemplated at some future date. Clearly the results obtained under the assumption of 45-Mw operation contain those which apply to 30-Mw operation. A third document will be published if it is decided to increase the reactor power to 45 Mw. This document will give the details of the physical modifications required in order to permit operation of the ORR at power up to this level. The purpose of the present document is twofold: (1) to collect and to clarify the existing information concerning the safety aspects of the operation of the ORR as it now exists, and (2) to extrapolate and, where necessary, to supplement this information in order to present a safety analysis of the conditions associated with operation at 45 Mw.

This safety analysis, which deals primarily with potential environmental contamination and its attendant hazards to the local as well as the general population, has been prepared under the assumption that the reader is familiar with, or at least has access to, the descriptive report. For convenience, references to various sections of the descriptive report have been included whereever this seemed appropriate.

The modifications which would be required in order to permit 45-Mw operation are relatively simple and straightforward. In the main, they involve an increase in the external heat-removal capacity. Because no design work on these modifications has been started, they cannot be described in detail. Their required function can, however, be stated; it will be assumed, for the purpose of the safety analysis, that the design will be adequate to permit this function to be realized.

1. INTRODUCTION

The Oak Ridge Research Reactor (ORR) is a light-water-moderated and -cooled, beryllium- and water-reflected research reactor designed and built for use as a general-purpose research tool. The reactor, which is described in detail elsewhere,[2] employs highly enriched uranium-aluminum alloy plate-type fuel of conventional

[1]F. T. Binford (ed.), *The Oak Ridge National Laboratory Research Reactor Safeguard Report*, ORNL-1794 (Oct. 7, 1954).

[2]T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I) (to be published).

design. Since 1960, it has been operated at its present power level of 30 Mw.

Conceptual work began on the ORR in 1954; construction was started in 1955, and initial criticality was achieved on March 21, 1958. The original design concept called for a power level of 5 Mw; however, this was revised to 20 Mw in early 1955, and, with the exception of the external heat-removal system, the reference design was finally based upon operation at 30 Mw. Ample safety factors were included in the design of the reactor vessel, underground piping, and biological shield in order to permit eventual escalation to a higher power level, which would be determined after operating experience was gained and refinements of the heat-removal calculations were made.

Following an initial testing period, in which it was found that the existing external heat-removal system was inadequate to handle 20 Mw during periods of hot weather, routine operation at 15 to 20 Mw was begun and continued for two years.

During the summer of 1960 extensive modifications to the cooling system were completed. These included the replacement of the original water-to-air forced-draft heat exchangers with shell-and-tube heat exchangers which utilize a secondary water system cooled by a two-cell forced-draft cooling tower.[3] These modifications permitted year-round operation at 30 Mw, which has continued ever since.

The design of the reactor and its ancillary facilities was, to a very large extent, influenced by two important requirements: (1) that the reactor be capable of accepting a variety of different types of experiments with a minimum of interference with operation, and (2) that significant radioactive contamination of the environment from any cause whatsoever be prevented. The former is necessary in order for the reactor to fulfill its mission as a research tool. The latter is, of course, required of any reactor built and operated at ORNL; it is provided for by the control and safety systems[4] and by the operating procedures[5] and is guaranteed by the reactor containment features.[6]

In addition to the fuel cladding, the reactor is provided with two other types of containment. The first of these is the reactor vessel and the primary coolant system.[7] Because the primary cooling system does have an interconnection with the reactor pool, it does not provide absolute containment. Fission products escaping through the equalizer leg into the reactor pool may find their way into the reactor building. Material conveyed along this path must pass through a demineralizer and a degasifier and then be carried upward through approximately 14 ft of pool water before reaching the atmosphere of the reactor building. Experience[8] has shown that virtually all of the iodine which may escape in this fashion is retained in the water; however, some of the noble gases do reach the reactor building.

The second type of containment, the principal containment, is provided by the ORR dynamic containment or "confinement" system.[6] Air is constantly removed from the building and exhausted through the building ventilation system. This system, designed to remove ~9000 cfm of air, directs the effluent through appropriate filters to the 250-ft ORNL stack, with stack discharge at an elevation of 1064 ft. It provides inleakage of air to the potentially contaminated areas, thus preventing outleakage of air- or vapor-borne fission products from the building at ground level. The effluent air in passing through the filters deposits therein, with the exception of the noble gases, virtually all of the airborne activity.

Although in many cases the principal containment is not required to prevent environmental contamination, it is a maximum-reliability system capable of handling any credible accident. Administrative safeguards require that the containment be operative whenever the reactor is in operation. Air flow through the system is continuous; however, to put the building into

[3]T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I), sect. 6.22 (to be published).

[4]T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I), sects. 8.3 and 8.4 (to be published).

[5]*Operating Manual for the Oak Ridge Research Reactor*, ORNL-TM-506.

[6]T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I), sect. 4.2 (to be published).

[7]T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I), sect. 6.2 (to be published).

[8]T. M. Sims and W. H. Tabor, *Report on Fuel-Plate Melting at the Oak Ridge Research Reactor July 1, 1963*, ORNL-TM-627 (October 1964).

containment, it is necessary that certain damper-equipped ventilating fans be shut off. This is done automatically by any one of three separate radiation-detection systems or by the activation of a fire alarm, or it may be done manually from the control room or from outside of the building.

A second type of environmental contamination involves the accidental introduction of contaminated water into White Oak Creek as a result of spillage or runoff or because of the failure or malfunction of the ORR aqueous waste system.[9] Such events are effectively prevented by the design of the system and by operating procedures[10] but in any event would be relatively minor in terms of hazard to the population. The activity level of the creek is constantly monitored,[11] and its rate of discharge through White Oak Lake can be controlled. Moreover, the time lag between the introduction of contaminated water into the watershed at the ORR site and the appearance of activity at the mouth of the creek is in excess of 12 hr. This gives ample time to spread the warning downstream.

2. MODIFICATIONS FOR 45-Mw OPERATION

On the basis of calculations which have been confirmed by experimental work and experience (see Appendix B), it has become obvious that the internal heat-removal mechanisms of the ORR are capable of safely handling heat loads considerably in excess of those imposed by 30-Mw operation. Since an increase in reactor power offers a number of advantages to the experimenters who utilize the ORR, it may be proposed to increase the power of the reactor from 30 to 45 Mw. The basic modification required to permit this involves an increase in the capacity of the external heat-removal system to permit wasting of the additional heat to the atmosphere. No increase in primary coolant flow or in core heat-transfer area is needed or is contemplated. In addition, a number of relatively minor changes will be necessary to permit orderly operation at 45 Mw. The changes which would be required are described in Appendix D.

In general, an increase of 50% in power would not represent a significant departure from the current mode of operation. Virtually all of the safety problems associated with 45-Mw operation are identical with those encountered over the past eight years. The only significant differences are: (1) The margin between the maximum operating heat flux and the burnout heat flux under normal operation conditions is reduced from 4.0 to 2.7. (2) The margin between burnout heat flux and maximum heat flux at the low-flow scram point (14,000 gpm) is reduced from 3.6 to 2.4. (3) The margin between burnout heat flux and maximum heat flux at the level safety trip point[12] is reduced from 2.7 to 2.1. If the flow is just above the low-flow trip point, this margin is reduced from 2.4 to 1.9. (4) The maximum (hot spot) fuel surface temperature under normal operating conditions will increase from 230 to 274°F. (5) The inventory of short-lived fission products will increase by a factor of 1.5.[13] Thus the safety margin between normal operating conditions and burnout remains well above 2, and the safety margin under emergency conditions is nearly 2. These changes will have no material effect on the consequences of an accident beyond the fact that the inventory of iodines and noble gases available for release is increased by about 50%.

3. METHOD OF ANALYSIS

The method which has been selected to pursue the investigation of the environmental consequences of a reactor accident consists first in a detailed examination of the results of a "unit" release of activity, and then the application of the results so obtained to the releases which could result from the various credible accidents. Unless otherwise specified, the accidents discussed will be assumed to occur under the assumption that the normal operating power level of the reactor is 45 Mw. Thus, in general, the consequences discussed

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[9] T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I), sect. 11.3 (to be published).

[10] F. L. Culler, "Criteria for Handling Melton Valley Radioactive Wastes," internal memorandum (May 1961).

[11] R. J. Morton (ed.), *Status Report No. 1, Clinch River Study*, ORNL-3119 (July 1961).

[12] For the purpose of this comparison the level safety trip point has been arbitrarily chosen to be 125% of the 45-Mw power level.

[13] The inventory of long-lived fission products is essentially unchanged because at 45 Mw the fuel burnup will be about the same as at 30 Mw.

are associated with a power level 50% greater than the present 30 Mw.

The only credible cause for a massive release of fission products from the ORR fuel is overheating to an extent sufficient to cause melting and destruction of the integrity of the fuel cladding. This can be brought about either by an increase in power beyond the capacity of the heat-removal mechanisms, a decrease in the heat-transfer capability of the system, or a combination of both.

The consequences of such an occurrence will vary depending upon the extent to which the reactor core is affected and also upon whether the fission products escape directly into the pool water, because the reactor vessel is open at the time of the accident, or whether the fission products must take the more tortuous path through the equalizer leg before reaching the pool. In the latter case, most of the iodine will remain behind on the demineralizer or in the water. The escape of the noble gases will be slower, but it can be assumed that eventually they will all be conveyed to the 250-ft stack serving the ORR.[14] Despite this, it will be arbitrarily assumed for the purpose of this analysis that the reactor vessel is ineffective in containing the volatile fission products and that they all escape directly into the pool water. There are only five apparent conditions which could cause sufficient overheating to melt the fuel. These are:

1. A nuclear excursion caused by the inadvertent addition of sufficient reactivity to permit the reactor to reach a very high power level before being shut down either by the safety system or by one or more of the various negative reactivity coefficients.
2. An increase in power beyond the capacity of the cooling mechanisms. This would require failure of both the control and safety systems.
3. A decrease in reactor flow to a point well below the low-flow scram point. This too would require multiple failure of the safety system.
4. A combination of conditions 2 and 3 above.
5. A local flow blockage in the core due to the presence of foreign material or the failure of a core component, either undetected at startup or occurring suddenly during power operation.

In addition to these five conditions, all of which involve reactor fuel, there are three other types of accidents which, although they are in general much less severe, nevertheless require consideration. These are:

6. An accident to an experiment which contains radioactive material.
7. An accident involving radioactive material other than that contained in the fuel or an experiment; that is, contaminated waste, ion exchange resin, etc.
8. A criticality accident involving reactor fuel.

Each of these types of accidents will be discussed, including their possible causes and the likelihood that they can occur. Those that may result in significant environmental contamination will be considered in detail.

4. THE CONSEQUENCES OF THE EMISSION OF FISSION PRODUCTS FROM THE ORR STACK FOLLOWING AN ACCIDENTAL RELEASE

Calculation of the downwind radiation doses to be expected as a result of the emission of fission products from the ORR stack have been made using the Gaussian plume formula.[15] The method, which takes into consideration decay, growth of daughter products, and the augmentation of the stack height due to emission velocity, is described in ORNL-4086 (ref. 16). In principle, it is quite similar to the method used to compute the doses for the HFIR safety analysis;[17] however, as was pointed out in that study, the procedure used to convert atmospheric concentrations to whole-body dose rates was based upon an approximation which grossly underestimates the dose near the source of emission and, it has been found by comparison with the present work, which overestimates the doses at distances greater than about ten stack heights from the source. The method described in ORNL-4086 makes use of actual space integration over

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[14] J. F. Manneschmidt and E. J. Witkowski, *The Disposal of Radioactive Liquid and Gaseous Waste at Oak Ridge National Laboratory*, ORNL-TM-282 (Aug. 17, 1962).

[15] F. R. Gifford, Jr., *The Problem of Forecasting Dispersion in the Lower Atmosphere*, USAEC pamphlet (July 1961).

[16] F. T. Binford, F. B. K. Kam, and J. Barish, *Estimation of Radiation Doses Following a Reactor Accident*, ORNL-4086 (to be published).

[17] F. T. Binford, T. E. Cole, and E. N. Cramer, *The High Flux Isotope Reactor, Accident Analysis*, ORNL-3573.

the volume of the plume, taking into consideration the gamma energies of each of the radioisotopes involved as well as the appropriate buildup factors. Thus, it is believed that the method used gives results which are quite accurate within the limitations inherent in the use of the Gaussian plume model.

As will be seen in Sect. 6, even a 100% meltdown of the ORR core presents no significant problem beyond the controlled area boundaries. On the other hand, because the ORR is located within the main ORNL complex, an accurate estimate of the radiation doses to be expected near the reactor are important so that proper precautions can be taken to protect the Laboratory personnel. For this reason the whole-body gamma-ray dose that results from direct radiation from fission products within the building has been computed as a function of time and must be added to that delivered by the stack plume. In this calculation no credit has been taken for the shielding effect of buildings and other structures which may be along the line of sight from the building to the receptor.

The integral method outlined above is not amenable to the dose maximization procedure used in the case of the HFIR. Consequently, two types of meteorological conditions have been examined: (1) most representative conditions, in which moderately unstable (C-type) atmospheric conditions prevail, the wind speed u is 100 m/min, and the effective stack height is given by $h = 76.2 + 2106/u$ m,[18] and (2) inversion conditions, where stable (F-type) atmospheric conditions prevail, the wind speed is 50 m/min, the stack height is fixed at 76.2 m, and vertical dispersion is restricted when the vertical dispersion parameter reaches the value 38.1 m. In both cases, the fractional building exhaust rate is taken to be the normal value of 0.01 min^{-1}.

Parameter studies of accident conditions at the HFIR[17] have led to the conclusion that the inversion condition described above gives sufficiently pessimistic results to cover virtually all possible cases, including fumigation, at distances greater than about ten times the stack height. Moreover, as in the case of the HFIR, it is only the noble gases and the iodines which make a significant contribution to the doses. Consequently, attention has been restricted to the internal dose which results from the inhalation of iodine and to the whole-body gamma-ray dose delivered by the noble gases in the stack plume and by direct radiation from the building contents.

The basic data for the iodine-inhalation dose are given in Figs. 4.1 and 4.2. The former represents the doses to be expected under most representative conditions (i.e., slightly unstable meteorological conditions and a wind speed of about 3.7 mph), whereas the latter provides the same information for inversion conditions. In both cases, the doses are given for several different exposure times. It can be seen that, although inversion conditions produce a lower peak dose, the peak is displaced several kilometers from the emission source and thus is more likely to occur outside the site boundary.

Figures 4.3 and 4.4 give the whole-body noble-gas doses from the stack plume for the most representative and the inversion case respectively. Here the displacement of the peak is not so marked, but the doses delivered under inversion conditions are nearly a factor of 3 higher than those to be expected under the most representative conditions.

The gamma-ray dose delivered by the radioactive material in the building is displayed in Figs. 4.5 and 4.6. The former includes the dose from the noble gases, their daughters, and the noble gases which appear as a result of the decay of iodine. The latter includes only the iodines themselves.

In addition, the external gamma-ray dose delivered by the building contents has been computed as a function of time for several nearby locations. The results are given in Figs. 4.7 and 4.8. A perusal of these curves will give some idea of the time available for persons located near the reactor to escape following any given accident. It should be realized that the values given are gross upper bounds because of the very liberal assumptions used. These include the assumption that the fission products appear uniformly distributed in the building at time zero, and no credit has been taken for shielding by the building wall, structures within the building, or other structural material that may intervene between the building and the receptor. (These assumptions also hold for the values given in Figs. 4.5 and 4.6.) Thus, it is quite likely that the doses computed are conservative by a factor of 2 or more.

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[18] The constant 2106 is based upon a minimum emission rate of 95,000 cfm and an orifice diameter of 8 ft.

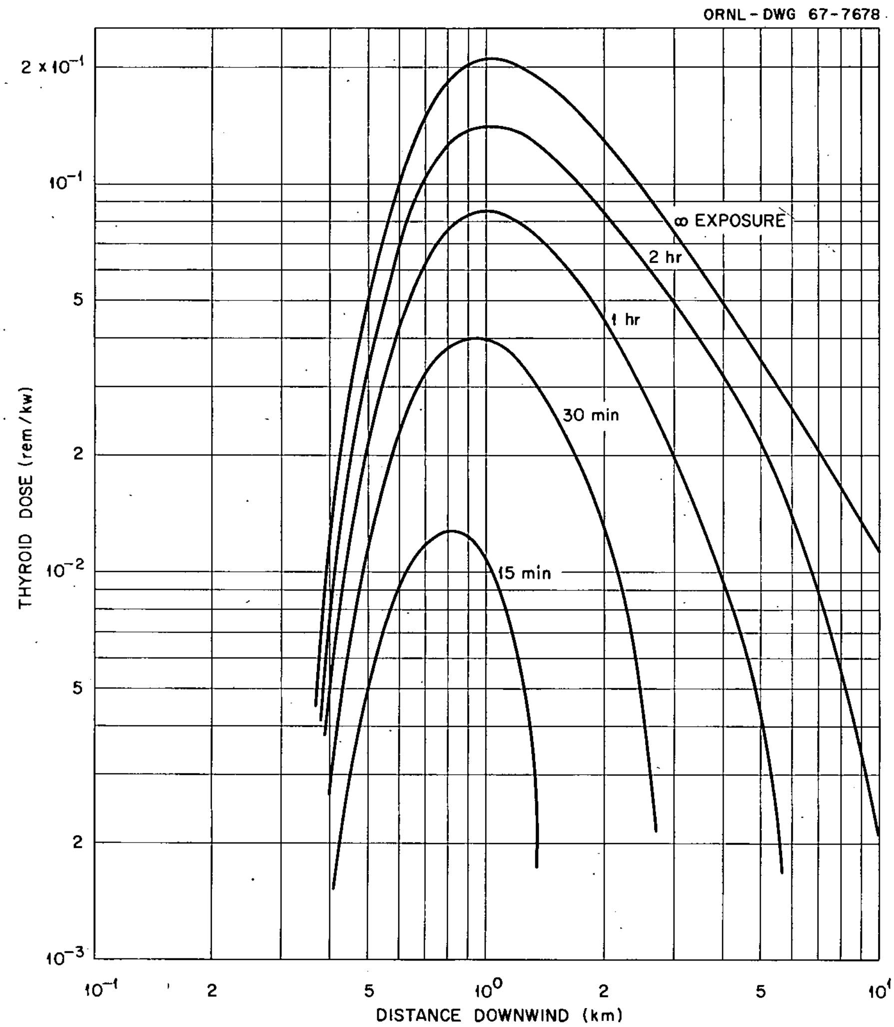

Fig. 4.1. Internal Iodine Dose as a Function of Distance Downwind and Exposure Time. Most representative conditions.

All of the data cited above are given on a "per kilowatt" basis, that is, for 30-Mw operation under the assumption that 1/30,000 of the fission products are released or for 45-Mw operation under the assumption that 1/45,000 of the fission products are released. It is emphasized that these curves (Figs. 4.1–4.8) contain no correction for filtration, release efficiency, or other processes that reduce the amount of activity released. The appropriate correction factors are incorporated in the results of the accident analysis given in Sect. 6. Application to any given accident situation can easily be made by the use of appropriate multipliers. For example, if at 45 Mw an accident occurs which releases from the fuel 10% of the noble gases and 5% of the iodines present in the core, then noble-gas doses given in Figs. 4.3, 4.4, 4.5, and 4.7 should be multiplied by 4500, while the iodine doses given in Figs. 4.1, 4.2, 4.6, and 4.8 should be multiplied by 2250. Appropriate factors must also be included to account for iodine retention in the pool water and, in the case of Figs. 4.1 and 4.2, decontamination by the charcoal filters.

5. ANALYSIS OF CREDIBLE ACCIDENTS IN THE REACTOR CORE

As has been pointed out previously, overheating of the fuel is the only mechanism which can cause

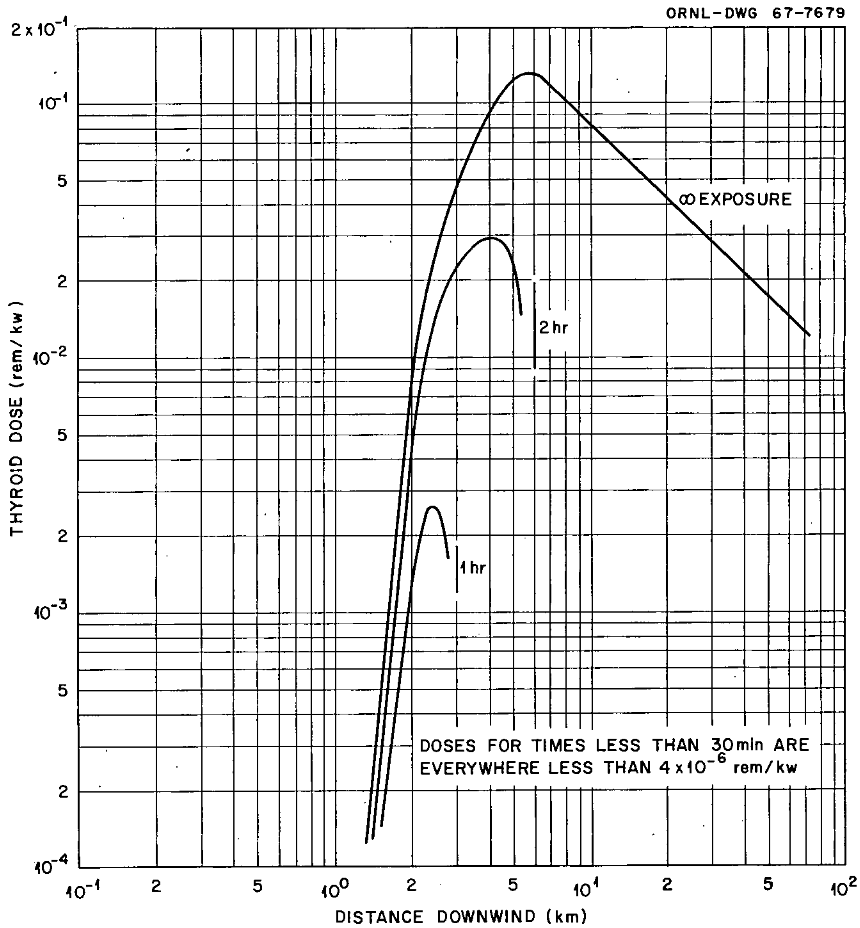

Fig. 4.2. Internal Iodine Dose as a Function of Distance Downwind and Exposure Time. Inversion conditions.

serious damage to the ORR core. Overheating could only result from an increase in reactor power, a loss of cooling capacity, or a combination of both. The various circumstances which could lead to one or the other of these conditions are discussed below.

5.1 Reactivity Accidents

Reactivity accidents are situations in which reactivity is added to the reactor in an uncontrolled manner at a rate sufficient to cause the reactor power to rise to a dangerous level. Two types of reactivity accidents are considered possible: (1) a startup accident and (2) a rapid insertion of reactivity due to failure or malfunction of some component or because of misoperation of the reactor.

The conventional "startup" accident may be defined as a situation in which all of the control rods are withdrawn from the reactor at their maximum rate of travel. In the case of the ORR, this would amount to an average rate of 0.071% reactivity per second over the entire length of the rods, with the maximum rate being 0.14% at the position of greatest worth. It is assumed in this accident that, with the exception of the level safety trips, all of the control and safety instrumentation fails; so that, unless the reactor is shut down by an intrinsic mechanism such as the temperature or void coefficient, the power level will continue to rise exponentially until the level safeties produce a reactor scram.

Startup accidents in the ORR have been investigated by analog techniques. The results of this investigation, which are presented in Appendix C,

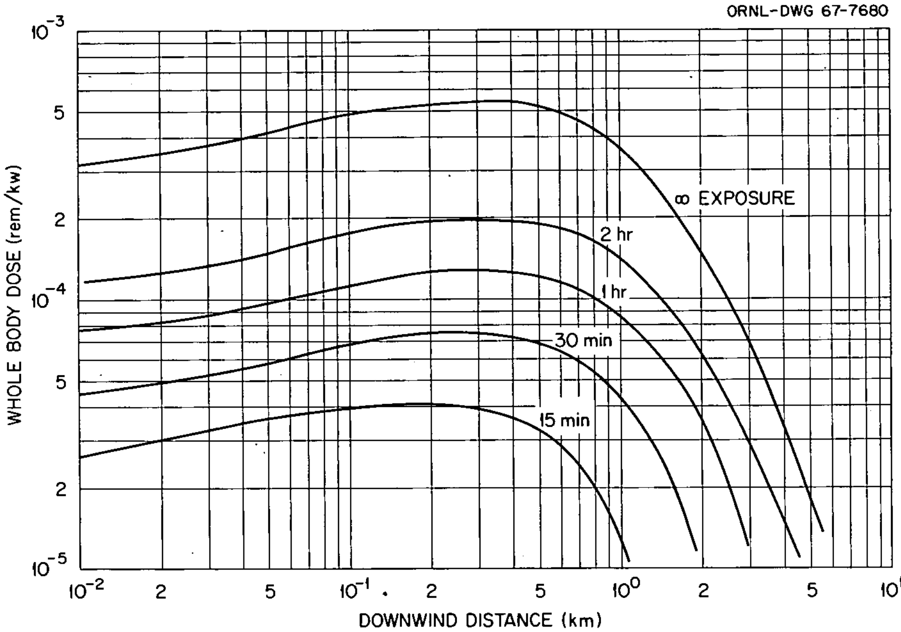

Fig. 4.3. Whole-Body Noble-Gas Dose as a Function of Downwind Distance and Exposure Time. Most representative conditions.

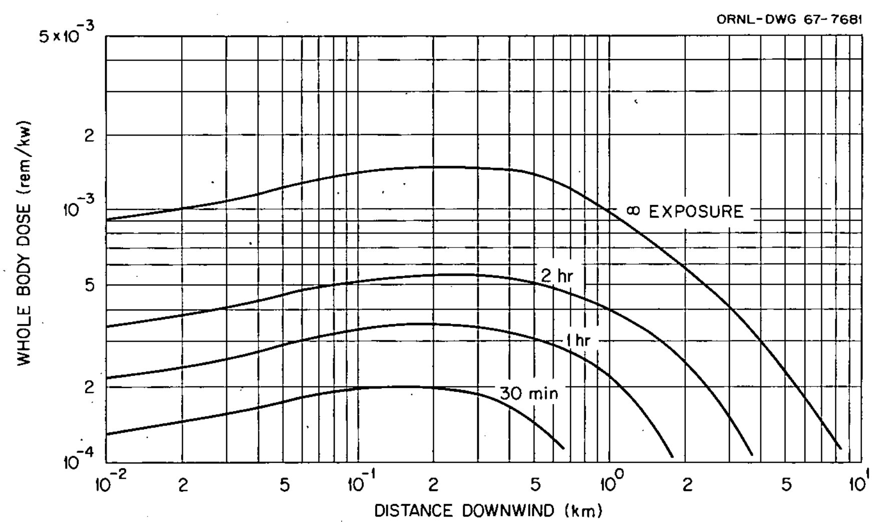

Fig. 4.4. Whole-Body Noble-Gas Dose as a Function of Downwind Distance and Exposure Time. Inversion conditions.

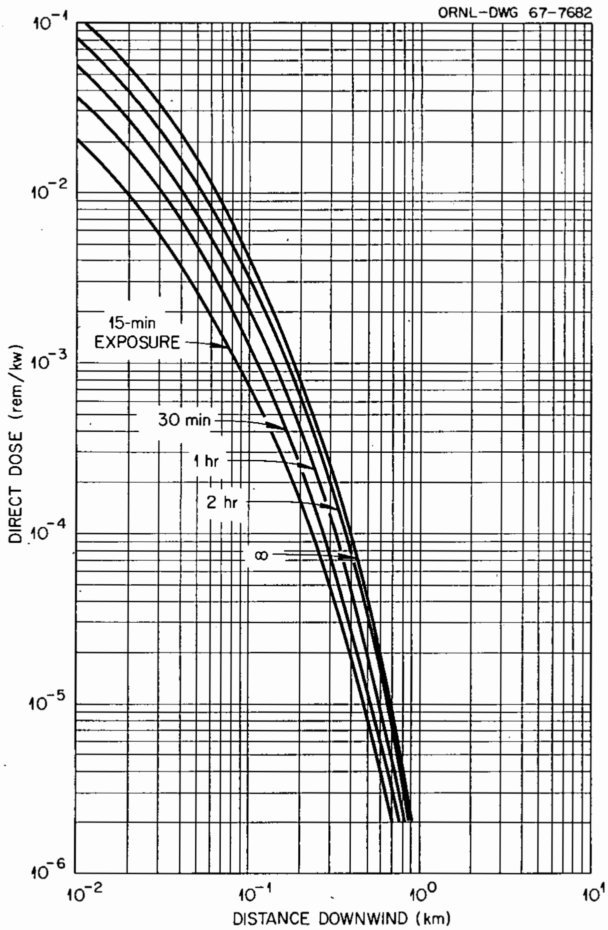

Fig. 4.5. Whole-Body Gamma Dose Due to Direct Radiation from the Building vs Distance from the Building. Noble gases, noble-gas daughters, and noble-gas daughters of iodines.

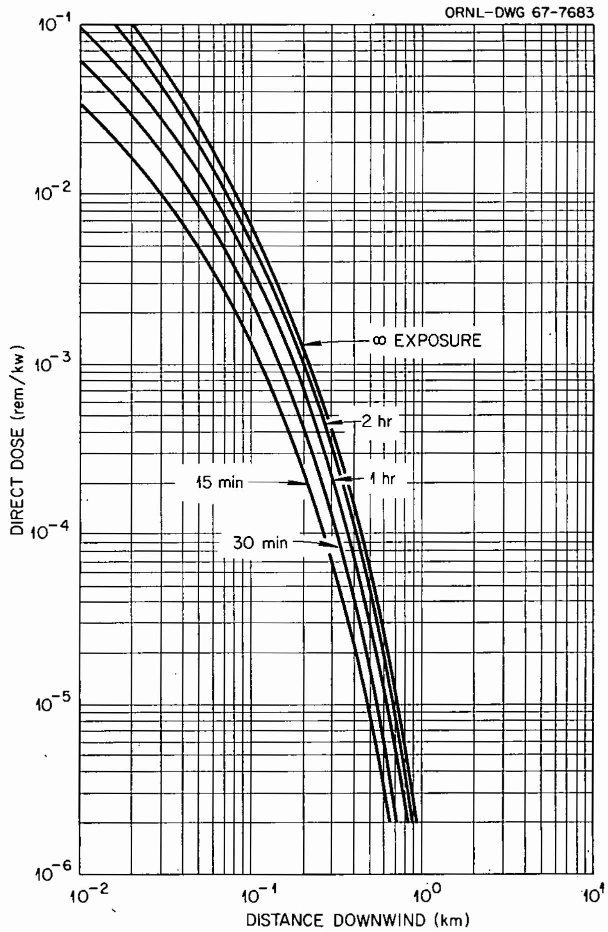

Fig. 4.6. Whole-Body Gamma Dose Due to Direct Radiation from the Building vs Distance from the Building. Iodines.

indicate that the consequences of the startup accidents are trivial and would not cause damage to, or even a significant temperature rise in, the reactor core. Calculations were made under full-flow conditions and also under the assumption of essentially zero flow, starting, in both cases, with the reactor subcritical. Initial moderator temperatures of 70 and 90°F were investigated; however, the temperature effect is negligible. The only case in which significant overheating could occur is that in which a startup accident occurs under zero flow conditions with the level safety trip set at a value corresponding to full-power operation. However, the level safety trip point is automatically reduced to ~1 Mw when the coolant flow rate is less than 14,000 gpm. The analog study shows that a scram at 1.35 Mw is more than adequate to prevent overheating.

It is concluded, therefore, that a startup accident cannot cause damage to the ORR core.

Because an ORR core may initially contain as much as 16% excess reactivity, it is necessary to determine how much of this reactivity could be added rapidly to the core in an inadvertent fashion and to estimate the consequences of such an addition.

Analog calculations, also shown in Appendix C, have been made in order to estimate the magnitude of the step increase in reactivity which would be required to cause significant damage to the core. These calculations are quite conservative, be-

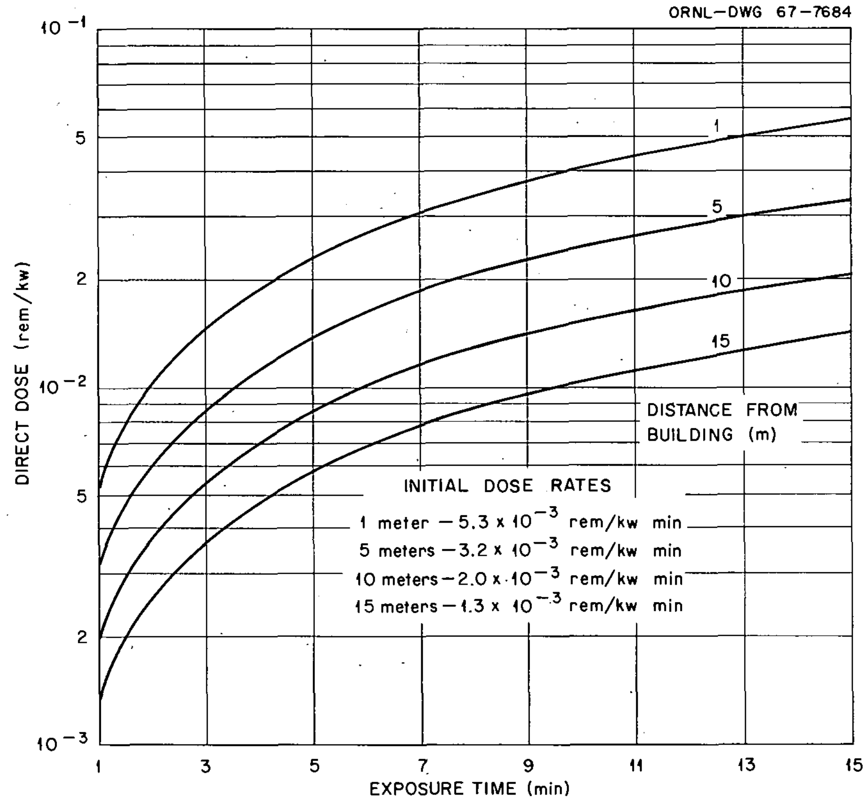

Fig. 4.7. Direct Radiation Dose Delivered at Various Points near the Building. Noble gases.

cause the model used does not include the negative reactivity effects of the moderator-void coefficient. Moreover, the only safety-system action included in the model used is that of the level safeties. It is shown in this study that even under these pessimistic conditions a step insertion of at least 2% in reactivity is required before any melting would occur.

These results are consistent with those obtained by comparison with the SPERT experiments, in which it was demonstrated that a reactor core quite similar to that of the ORR can withstand a 5-msec period without melting.[19] This corresponds in the case of the ORR to a step input of slightly over 2% in reactivity. Again, the results are conservative because no safety action was operative during this test. Tallackson[20] has shown that inclusion of a period scram circuit would permit the reactor to withstand a 3-msec period (~3% reactivity step). Moreover, it must be conceded that a true "step" is not possible, so that any reactivity addition must take place over a finite time. This introduces a further element of conservatism.

Thus it appears from the foregoing that it would require the rapid addition of at least 2% in reactivity to the ORR in order to initiate a transient of sufficient magnitude to cause significant damage to the core.

Aside from deliberate sabotage, there appears to be no credible method for rapidly introducing this amount of reactivity. It is a firm requirement that the reactor be loaded in such a way that it is subcritical when the control rods are less than

[19]M. R. Zeissler, *Non-Destructive and Destructive Tests of the Spert 1-D Fully Enriched Aluminum Plate-Type Core: Data Summary Report*, IDO-17886 (November 1963).

[20]*Instrumentation and Controls Div. Ann. Progr. Rept. July 1, 1960*, ORNL-3001, sect. 16 (Jan. 27, 1961).

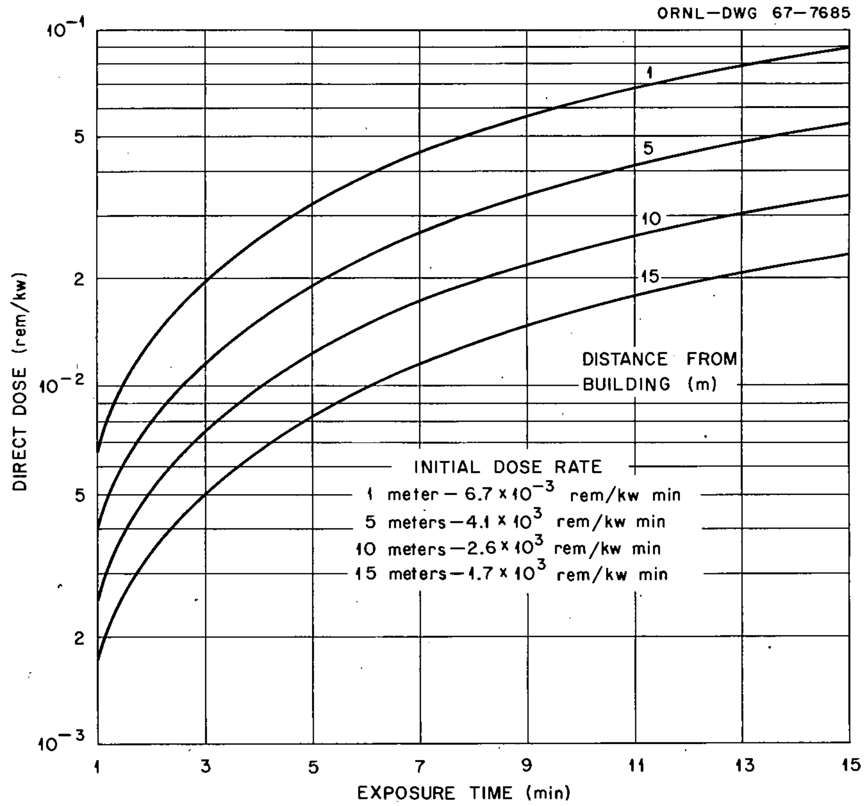

Fig. 4.8. Direct Radiation Dose Delivered at Various Points near the Building. Iodines.

halfway withdrawn. Thus, when shut down, it would require the complete removal of at least two of the control rods to even reach criticality. This cannot be done rapidly. The withdrawal of all of the control rods has been discussed above.

There remains only the possibility that failure or malfunction of an experiment or core component could result in a reactivity increase. This is guarded against by careful review of all experiments[21] to make sure that no experiment is installed which, by failure, could rapidly introduce more reactivity than can be handled by the servo-control system (~0.5%). At present the total available worth of the most reactive experiment is less than 1%; the probability that this could be realized, even slowly, is very small. Each experiment is carefully reviewed prior to its installation to make sure this is so.

Of the core hardware, the fuel elements, beryllium reflector pieces, and control rods possess significant amounts of reactivity. Since the ^{235}U mass coefficient of reactivity is everywhere positive, removal of fuel will reduce reactivity; the same is true of the removal of beryllium. Removal of the control rods has been considered previously; however, it is worth noting that, since the control rods are driven from below the core and are retained by positive stops at the shock absorbers,[22] they cannot fall out of the core. The only possible way in which reactivity could be added to the core by removal of a fuel element or reflector piece is as follows. Assume that a core component is

[21] C. D. Cagle, *General Standards Guide for Experiments in ORNL Research Reactors*, ORNL-TM-281 (Aug. 20, 1962).

[22] T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I), sect. 8.2 (to be published).

moved up out of the core by some mechanical action (which would have to be associated with an experiment or control rod). This would cause a decrease in reactivity, which could be compensated for (up to 0.5% in reactivity) by the servo. If the core piece then fell back into position, it could quickly add the amount of reactivity available to the servo at the time of its removal. This alone presents no problem; however, if the operator manually shimmed the reactor by withdrawing the control rods, the entire worth of the core piece in question could be involved. All of the more reactive core components – those located in the central part of the reactor – are retained in place by the two holddown arms which carry the control rod bearings.[23] Thus, only low-worth components could be involved. The maximum amount of reactivity attributed to any such single core element is less than 1.4%. Moreover, because of the precautions taken during the design and installation of experimental apparatus, the probability of such an occurrence is nil.

5.2 Loss-of-Coolant Accidents

Loss of adequate cooling capacity due to a reactor "walkaway," that is, an undetected gradual increase in power, or due to a gradual loss in cooling are effectively prevented by the level safety scram[24] and by the low-flow scram.[25] In addition, scrams on the reactor ΔT and outlet temperature[26] would also operate to terminate such a situation before dangerous levels were reached. All four of these are elements of the reactor safety system and are characterized by redundancy and maximum reliability. In addition, operation at a power level significantly above normal is guarded against by the inclusion of a "setback" and a "reverse" between the normal power level and the scram trip point.

[23] T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I), sect. 5.2 (to be published).

[24] T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I), sect. 8.4 (to be published).

[25] T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I), sect. 8.32 (to be published).

[26] T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I), sect. 8.5.2 (to be published).

As is shown in Appendix B, the reactor can be operated quite safely at a power level just below the power scram point and at a flow just above the low-flow scram point. Operation at power levels greater than about 1 Mw with flow less than 14,000 gpm is prevented by the automatic reduction of the power-level scram trip point when this flow is reached.

The ORR is provided with afterheat protection by dc motor drives mounted coaxially with the main ac motors which drive the primary circulating pumps.[27] As shown in Appendix B, any one of these is more than adequate to prevent boiling in the core. Moreover, since the maximum heat flux to be expected from a shutdown ORR core is 75,000 Btu ft^{-2} hr^{-1} and the natural circulation burnout heat flux is estimated to be 125,000 Btu ft^{-2} hr^{-1}, it appears that forced convection cooling is not necessary to prevent significant core damage from afterheat. Nevertheless, the afterheat provisions represent a redundant maximum reliability system and are always required to be operable when the reactor is at power.

One situation which could cause loss of forced convection afterheat protection would be a primary-coolant system break of sufficient magnitude to essentially uncouple the reactor from the pumps. Except in the pipe chase below the reactor pool, in the vicinity of the pump house and heat exchangers, and a short span in the pool, the primary water lines are buried, either in concrete or underground, so that there is no possibility of a break of this magnitude – except in the locations cited. Moreover, because the primary system operates at low temperature and pressure, it is not subjected to high stresses, and thus a major rupture is highly unlikely. Because of the design of the system and the elevations involved, the reactor vessel would not be drained by such a break; water from the pool would enter the vessel through the equalizer leg.[28] Under these conditions it is unlikely, but possible, that some melting could occur.

Probably the most likely cause for core damage is as a result of a local flow blockage in one or more fuel elements. Such blockages have occurred

[27] T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I), sect. 6.2.1 (to be published).

[28] T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I), sect. 6.4.1 (to be published).

at several reactors,[29,30] including the ORR,[31] and have caused fuel melting. In none of these cases were the results catastrophic or even particularly serious, except in terms of lost time required for cleanup and investigation.

Occurrences of this type have been due to the accidental introduction of foreign objects into the reactor core, usually plastic or other soft material which blocks the flow of water through the cooling channels or which coats their surfaces, thus inhibiting heat transfer. It is conceivable that failure of a fuel element as a result of faulty manufacture or mishandling could cause sufficient flow blockage to result in overheating; however, no such incidents have been reported.

The only real protection against accidents of this type is the establishment of procedures designed to prevent the introduction of foreign material into the system, coupled with careful surveillance to ensure that these procedures are carried out. This has been done at the ORR and includes a visual inspection of the core following startup but prior to operation at any substantial power level.[32] Fuel elements receive three inspections: once at the fabricator's plant; once upon being received at ORNL; and, finally, immediately before insertion into the reactor.

Usually, flow-blockage situations occur during reactor startup and may be detected by the practiced observer before any damage is done. As the power level of the reactor is increased, boiling will begin in a blocked channel and make itself evident by a distinct increase in the fluctuations in the output of the power-level instruments. In the case of the ORR meltdown cited above, an inspection of the charts indicates that detectable fluctuations began at a power level of about 9 Mw. Burnout occurred at approximately 24 Mw. Thus, had the operators known how to interpret the noisy output signal, the incident could have been avoided.

The lesson learned here has been incorporated in the operator and supervisor training program and is now thoroughly understood. Moreover, a high-sensitivity instrument which magnifies these fluctuations is now routinely used during startup.

While it is believed that the flow-blockage problem is now sufficiently understood to prevent such an accident during startup, a sudden flow blockage during full-power operation would be likely to cause damage before the reactor could be shut down.

It follows that the flow-blockage accident must be considered the most likely cause for a fission product release. However, it is highly unlikely that more than a small fraction of the core would be involved; in the cases experienced so far, there has been only a very minor release of fission products from the primary coolant system.

6. THE MAXIMUM HYPOTHETICAL ACCIDENT

Based on the discussion in Sect. 5, it appears highly unlikely that any malfunction of the ORR could result in a major release of fission products. In fact, it is difficult realistically to postulate a meltdown which involves more than very small fractions of the fuel. Nevertheless, in order to illustrate the results of a very severe accident, the consequences of a "maximum hypothetical accident" have been calculated for the ORR.

In this accident, *which is not only incredible but nearly an order of magnitude worse than any credible accident to the ORR*, it is assumed that the ORR core operating at 45 Mw suffers a 100% meltdown in which 100% of the noble gases and 50% of the iodines are released. It is also assumed that the noble-gas daughters of 100% of the iodines are also released.[33] The escape of nonvolatile fission products from the vicinity of the fuel is so small as to be negligible, particularly since the release takes place under water.[34] Experience indicates[31] that virtually all of the iodine will remain in the water. Parker[35] has estimated

[29]J. W. Dykes, J. D. Ford, and K. R. Hoopingarner, *A Summary of the 1962 Fuel Element Fission Break in the MTR*, IDO-17064 (February 1965).

[30]F. R. Keller, W. S. Little, and J. H. Ronsick, "Fuel Element Flow Blockage in the Engineering Test Reactor," *Conference on Light-Water-Moderated Research Reactors, Gatlinburg, Tennessee*, Book 3, TID-7663 (June 1962).

[31]T. M. Sims and W. H. Tabor, *Report on Fuel-Plate Melting at the Oak Ridge Research Reactor July 1, 1963*, ORNL-TM-627 (October 1964).

[32]Standard procedure calls for a core inspection when the reactor has reached 3 Mw and once per shift thereafter.

[33]In this treatment the iodine daughters are actually counted twice; this lends a slight additional element of conservatism but simplifies the calculation.

[34]F. T. Binford, T. E. Cole, and E. N. Cramer, *The High Flux Isotope Reactor, Accident Analysis*, ORNL-3573, Appendix A.

[35]G. W. Parker, ORNL Reactor Chemistry Division, private communication (February 1965).

that between 65 and 90% retention appears conservative. Consequently, it is assumed that the reactor pool provides a decontamination factor of 3 for iodine. The iodine which remains in the pool is available neither to the stack plume nor as a source of direct radiation. In addition to this, the building air is directed through filters before being passed up the ORR stack. These filters are designed to have a minimum decontamination factor for iodine of 10. They are normally operated with a decontamination factor of 100 or more, which is verified by semiannual testing. The filter factor is therefore taken to be 100.

Although the noble gases do have a finite solubility in water and are retained to some extent on the charcoal filters, it will be assumed that they reach the stack plume undiminished except by decay.

The results of these computations are given in Figs. 6.1 through 6.3. Both "most representative" and "inversion" meteorological conditions are shown, with the latter being in general the most severe. It is at once obvious from Fig. 6.1 that the iodine release presents no problem. Even for an infinite exposure time the maximum dose is more than an order of magnitude below the 300 rems specified in the guidelines of 10 CFR 100. Since the doses delivered in finite time are all less than that for infinite time, they have not been shown in the figure. The maximum dose delivered is about 16 rems, and from this fact it is possible to deduce that a filter decontamination factor of 5.3 is all that is required to keep this peak dose below 300 rems.

As can be seen from Figs. 6.2 and 6.3, the whole-body doses from gamma radiation do not exceed the 25 rems specified in 10 CFR 100 at any point outside the site boundary, even under the assumption of infinite exposure. Within 150 m of the building, the 25-rem guideline is exceeded for periods of exposure in excess of 15 min; but, at the most densely populated location in the Laboratory, the 4500-area complex, the exposure time required to exceed 25 rems is at least $\frac{1}{2}$ hr. This gives more than adequate time for evacuation.

It should be borne in mind that, at distances within 150 to 200 m of the building, the controlling external dose is that delivered by direct radiation from the building. The direct doses delivered for various exposure times following the maximum hypothetical accident are plotted in Fig. 6.4 as a function of distance. It should also be recalled

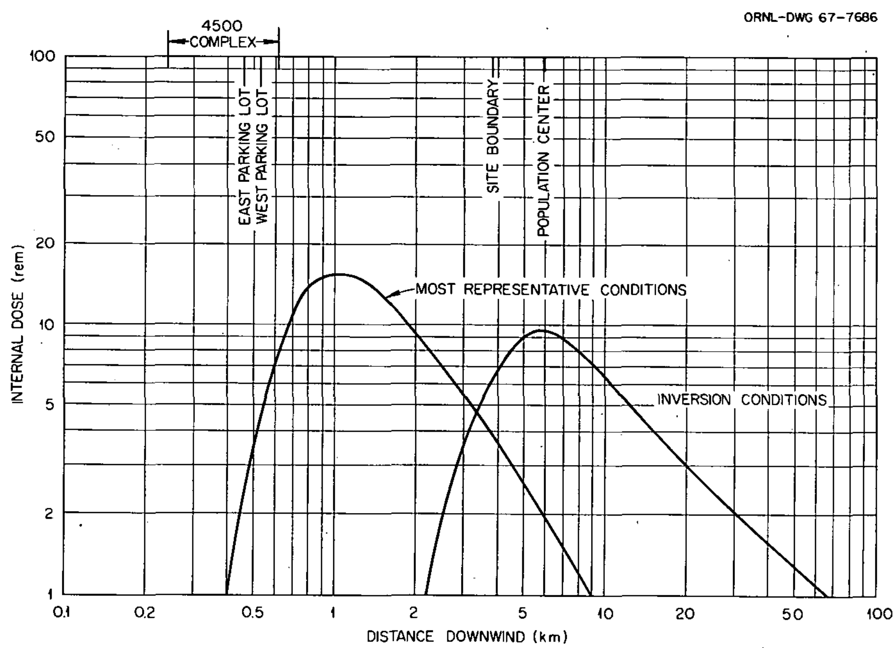

Fig. 6.1. Internal Iodine Dose, Infinite Exposure. Maximum hypothetical accident: 100% meltdown; 50% iodine release; 0.666 retained in water; decontamination factor of filters, 100.

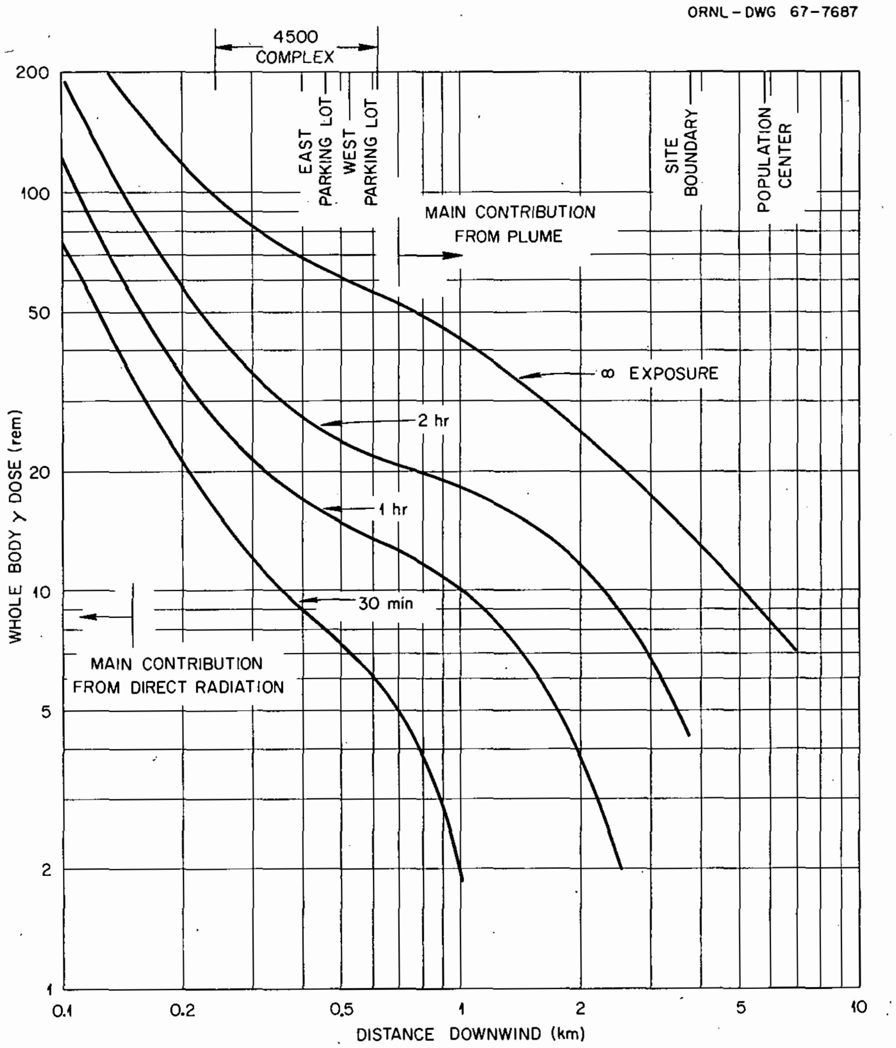

Fig. 6.2. Whole-Body Gamma Dose vs Downwind Distance Under Inversion Conditions. Maximum hypothetical accident.

that these doses are based on the extremely conservative assumptions: (1) that all the released fission products appear in the building instantly, (2) that the building and its contents afford no shielding, and (3) that there is no shielding of any kind between the receptor and the building. Thus, the values given are a gross upper bound.

The initial dose rate 1 m from the building wall following this accident is 300 rems/min. This has fallen to 7 rems/min at 100 m. Thus, an individual standing just outside the reactor building could escape very serious consequences if he took swift action to leave the scene. The initial whole-body dose rate within the building would be about 700 rems/min; however, the initial internal dose rate from ^{131}I would exceed 7500 rems/min. Thus, persons in the reactor bay would very likely become casualties unless they escaped before the fission gases became mixed with the atmosphere in the building.

In summary, it appears clear that even an accident of this incredible magnitude would have no serious off-site environmental consequences. Moreover, prompt evacuation of the Laboratory – which can be accomplished in 15 to 20 min, as demonstrated by unannounced practice drills –

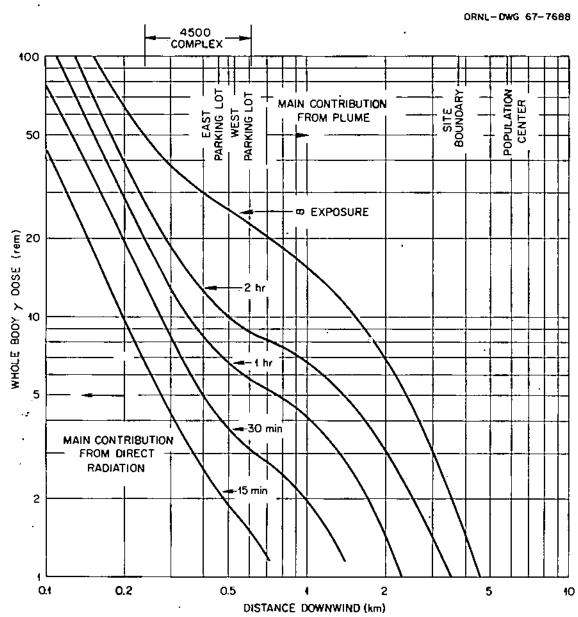

Fig. 6.3. Whole-Body Gamma Dose vs Distance Downwind Under Most Representative Conditions. Maximum hypothetical accident.

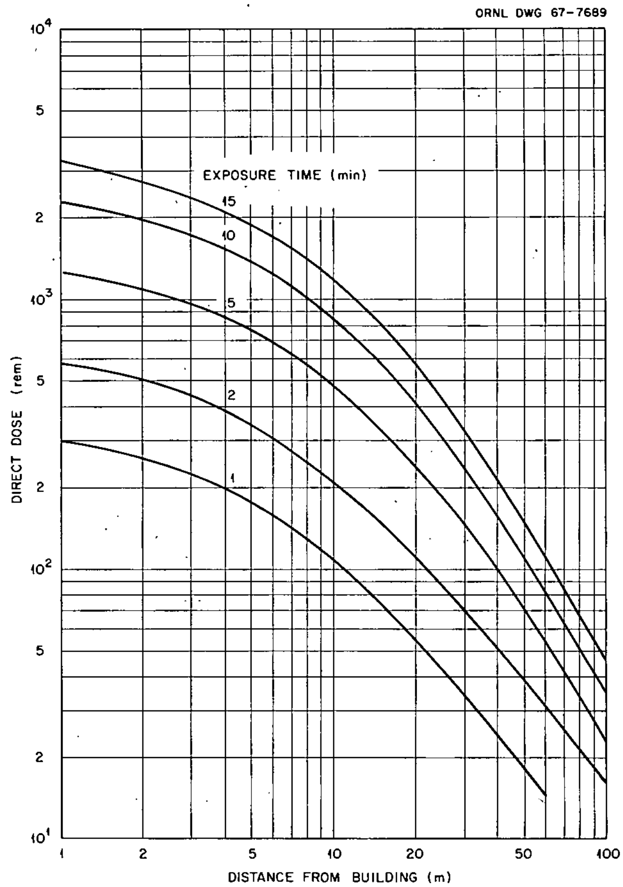

Fig. 6.4. Direct Radiation Dose near the Building Following the Maximum Hypothetical Accident.

would ensure that a minimum number of persons were exposed to the risk of receiving radiation doses in excess of the guidelines of 10 CFR 100. Those in most danger would be staff personnel whose place of work is located near the reactor. This involves approximately 490 people located within a 100-m radius of the ORR building. The location affected is shown in Fig. 6.5, and the estimated area population is given in Table 6.1. Despite the fact that considerable shielding may be interposed between their work location and the reactor building, persons working within the radius indicated must evacuate at once following the postulated hypothetical accident. It should be noted that the figure of 490 people is the normal daytime complement. On weekends and night shifts, the number is reduced to a maximum of about 15. Finally, it is emphasized again that the accident discussed above represents a gross upper bound and does not represent a credible occurrence.

7. POTENTIAL ENERGY RELEASE

None of the credible accidents discussed in Sect. 5 involve a significant release of nuclear energy. Based upon SPERT results, a 2% step addition of reactivity will result in a burst of only 17.5 Mw-sec and only minor damage to the fuel elements. Even the destructive 3.2-msec SPERT test released only 30.7 Mw-sec of nuclear energy.[36]

[36] M. R. Zeissler, *Non-Destructive and Destructive Tests of the Spert 1-D Fully Enriched Aluminum Plate-Type Core: Data Summary Report*, IDO-16886 (November 1963).

ORNL–DWG 67–7690

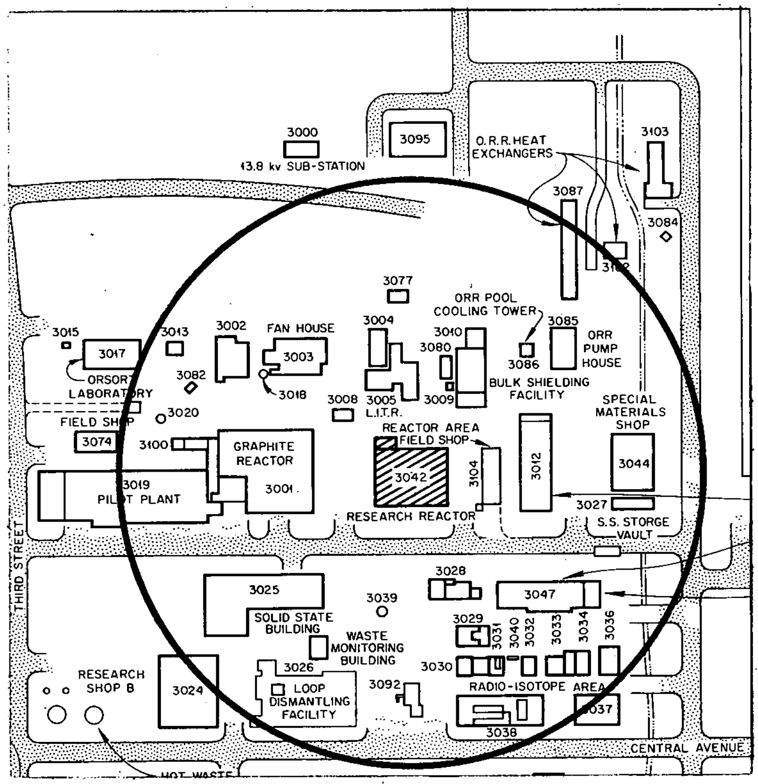

Fig. 6.5. Buildings Located Within a 100-m Radius of the ORR.

The only other potential source of energy is the aluminum-water reaction. This requires an ignition temperature of about 2140°F, which is considerably in excess of the melting point of aluminum (1220°F) and thus much higher than the incipient melting temperatures which might be reached following a 2% step increase in reactivity. Even in the SPERT destructive test it is estimated that only about 4 Mw-sec of chemical energy was released.[37]

It is extremely doubtful that a flow-blockage-type accident could develop sufficient temperature to cause any appreciable metal-water reaction. Moreover, the heating rate would be relatively slow in this case.

It is concluded that the likelihood of an appreciable metal-water reaction is small enough to be considered incredible.

8. ACCIDENTS NOT INVOLVING THE REACTOR CORE

All experiments in ORNL reactors are required to meet two basic criteria: (1) they must not present an unacceptable hazard to personnel and (2) they must not interfere with the orderly operation of the reactor. To ensure that these criteria are satisfied, all experiments must undergo an internal safety review[38] before they are installed. In many cases, this review also extends throughout the design period. Initially, the review is carried out by the Technical Assistance Department of the ORNL Operations Division. It is attempted at this level to resolve any problems regarding safety and to produce a design which meets the necessary requirements. Once agreement has been reached,

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[37] R. W. Miller, A. Sola, and R. K. McCardell, *Report on the Spert I Destructive Test Program on an Aluminum, Plate-Type, Water-Moderated Reactor*, IDO-16883 (June 1964).

[38] C. D. Cagle, *Considerations Involved in the Safety Review of Experiments to Be Operated in Nuclear Reactors*, ORNL-TM-745 (1964).

Table 6.1. Number of People Located Within 100 m of the ORR

Building Number	Number Present During Day Shift
3001	27
3009	0
3010	8
3012	4
3019	42
3024	34
3025	79
3026	18
3028	1
3029	4
3030	1
3032	0
3033	2
3034	0
3036	0
3037	45
3038	12
3042	47
3044	15
3047	52
3080	1
3104	95
3105	4
Total	491

the experiment may be approved for insertion in the reactor by the Technical Assistance Department. If any significant hazard had existed, even though it was corrected by the design, the experiment is submitted with appropriate recommendations to the Experiment Review Committee for further review. When this committee concurs that the experiment is safe, it may be inserted in the reactor. The committee also periodically reviews the operation of all experiments to make sure that its recommendations are being carried out.

Despite the careful design and review procedures utilized to prevent malfunction or misoperation of experimental apparatus, experience indicates that the most likely radiation accidents are minor releases from this source. Such incidents do not involve the reactor itself and in most instances occur after the experimental apparatus has been withdrawn from the core. Because of the containment features of the building and the small amount of activity involved, they do not produce environmental contamination. They do, however, constitute a hazard to the operating personnel.

In general, such accidents include leaking capsules, broken off-gas lines, loss of shielding, etc. Protection against these accidents is afforded by constant anticipation and planning and by the utilization of various special procedures, such as the use of temporary off-gas lines, shielding, etc. While such incidents are annoying and potentially dangerous, to date no excessive radiation exposures have been experienced; the frequency of the incidents is decreasing.

Perhaps the only type of waste-handling accident which might be of any significance is a large spill of contaminated water, which could eventually find its way to the White Oak Creek watershed and thence to the Clinch River. The only way in which any quantity of highly contaminated water could be accumulated at the ORR would be as a result of some previous major accident, in which case special emergency procedures would be put into effect.

The fission products of most significance with respect to contamination of watershed are the long-lived isotopes ^{90}Sr, ^{106}Ru, ^{137}Cs, and ^{144}Ce. These, together with their characteristics, are listed in Table 8.1. Clearly the controlling factor here is ^{90}Sr.

In a study of the ORNL waste-disposal pits undertaken in 1960,[39] it was found that with the

[39] F. T. Binford, *An Analysis of the Potential Hazards Associated with the Disposal of Radioactive Waste in Open Pits at ORNL*, internal correspondence (May 17, 1963).

Table 8.1. Characteristics of the Long-Lived Products

Isotope	Half-Life	MPC[a] (μc/cm^3)	Critical Organ	ORR Inventory[b] (curies)
^{90}Sr-^{90}Y	28 years	1×10^{-7}	Bone	2.6×10^3
^{106}Ru-^{106}Rh	1 year	1×10^{-5}	Kidney	4.5×10^3
^{137}Cs	26.6 years	2×10^{-5}	Muscle	2.6×10^3
^{144}Ce-^{144}Pr	290 days	1×10^{-5}	Bone	1×10^5

[a] Nonoccupational MPC for water from *Report of Committee II on Permissible Dose for Internal Radiation*, Pergamon, New York, 1959.

[b] Based on average burnup of 20% in a 5.5-kg core.

exception of ^{106}Ru virtually 100% of the long-lived activity is removed from solution by sorption as the contaminated water percolates through soil of the type found in the vicinity of the ORR site. The ^{106}Ru is removed to the extent of about 93%. This, coupled with the fact that probably only a small percentage of the inventory listed in Table 8.1 would escape from the fuel, makes it appear unlikely that any sizable quantity of activity would reach the watershed.

Should the release postulated in the maximum hypothetical accident actually occur and should the entire inventory of long-lived fission products reach the river undiminished, the computational procedures cited in the referenced report indicate that the result would be equivalent to the discharge of about 100 curies of ^{90}Sr to the Clinch River. Under the assumptions that the average flow rate of the river is 5000 cfs and that there is little or no reduction in the concentration due to dispersion and settling, but only by dilution, the integrated dose received at the Oak Ridge Gaseous Diffusion Plant, the nearest significant user of water, located $\sim$7.4 miles downstream, is equivalent to that received from approximately 1.5 weeks at continuous nonoccupational MPC. At locations farther downstream, the integrated dose would be considerably less because of additional dilution and dispersion in the Watts Bar Reservoir.

Spent and partially spent fuel elements are stored routinely in racks in the ORR pool. At times the inventory of these elements may exceed 100. The element location and fuel content of stored elements are known at all times, since burnup calculations[40] are made immediately after a core change, and new weights and locations are posted.

Fuel manufacturing procedures are such that it is not possible to fabricate by current techniques fuel elements containing more than a few percent ^{235}U above that called for in ORNL specifications. That the fuel elements actually meet ORR fuel-content experiments has been conclusively demonstrated by both measurements on new elements and analysis of depleted elements.[40]

Criticality safety is maintained during fuel moves through strict administrative procedures, which require that not more than two elements be moved at a time and that each element be either in the reactor, in a storage rack, or in transit. The storage racks, which are located under approximately 25 ft of water, are designed in such a manner that they constitute an always critically safe array. Although criticality safety depends very heavily on administrative procedure, there has never been a criticality accident or near accident at the ORNL reactors. All such administrative procedures are constantly reviewed in order to incorporate the most recent information available concerning fuel handling.

All core loadings are prepared by one qualified technical person and checked by another. During the past nine years, many cores have been assembled with not one case of inadvertent assembling of a core which formed a critical array unexpectedly. The operating criteria for the ORR specify that the "excess reactivity loading above clean-cold-critical will not exceed that which will permit achievement of criticality with the control rods withdrawn less than half their reactivity worths." However, when the rods are on seat, it would require approximately 1500 g of additional ^{235}U distributed over the core to achieve criticality.[41] This is assuming the present configuration and control rod worths. It may be concluded that the entire core could be loaded with new 240-g elements with the rods on seat and the reactor would not be critical.

It must be emphasized that fuel may be moved into and out of the core only upon the written instructions of the reactor supervisor or his representative and that during such refueling the subcritical multiplication is monitored audibly in order that the reloading crew will be immediately aware of any sudden increase of multiplication. Therefore, inadvertent assembly of a critical array during reloading appears to be impossible.

9. ADEQUACY OF THE REACTOR SHIELDING

The maximum permissible exposure limits specified for personnel at ORNL are given in Table 9.1.

For administrative purposes the Operations Division works to one-half of these values. Moreover,

[40] T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I), Appendix C (to be published).

[41] T. P. Hamrick, "An Experimental Evaluation of Errors Arising from the Burnup Calculation Used at the ORNL Research Reactor," American Nuclear Society Conference on Reactor Operating Experience, Jackson, Wyo., July 28–29, 1965.

Table 9.1. Personnel Exposure Limits to Direct Radiation

Organ	Recommended Maximum Weekly Dose (millirems)	Quarterly Dose (13 weeks) (rems)	Annual Dose (rems)
Total body, head, and trunk, eye lens, gonads, blood-forming organs	100	3	12
Skin of whole body	600	10	30
Hands	1500	25	75
Other organs		5	15

each employee of the Division receives as part of his training extensive instruction in health physics techniques. That the program of minimizing exposure has given satisfactory results is illustrated in Fig. 9.1, which presents a distribution of the annual whole-body dose of penetrating radiation received by all currently employed persons associated with ORR operation. These have been averaged over their respective periods of employment, which represents a total of over 600 man-years. As can be seen, the average for all ORR employees is less than 1 rem/year, and only one employee has received an average of more than 2.5 rems/year. Of the two peaks in the distribution, the one at the lower dose level represents administrative and technical personnel, while the other peak is associated with the doses received by the direct operating personnel. It should be realized that all of the individuals represented in this diagram are from time to time assigned to other reactors or areas which may or may not involve radiation, so that

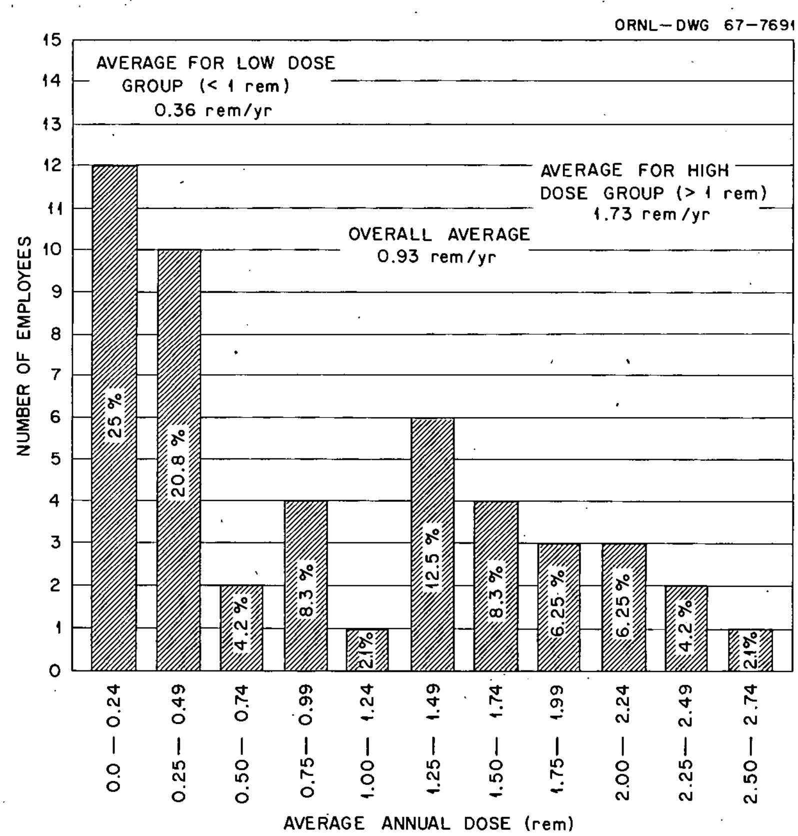

Fig. 9.1. Distribution of the Average Annual Dose of Penetrating Radiation Received by Employees Working at the ORR Through 1966.

Table 9.2. Background Survey (30-Mw Operation)

PERFORMED BY	REASON	DATE	TIME
H.G and H. A.	Weekly Check	12-10-66	1:00 p.m.

NOTE: This Survey is to be performed each Saturday and after the startup of the reactor at the beginning of a cycle. A survey meter (G-M Counter) shall be used for this Survey, with a Cutie Pie used when survey meter goes off scale.

AREA		mr/hr MAXIMUM BACKGROUND	mr/hr PRE-SET LOCATION
1. POOLS:			
a. East	TAKEN AT SURFACE NEAR POOL WALLS	2	2
b. Center		2	1.5
c. West		5	2
d. Top of Bridge		5	2.5
e. Above Hydraulic Tube		2.5	2.5
2. HOT CELL:			
a. North Cell			
1. Window		0.1	0.1
2. Access Holes (Not Including Top of Cell)		0.2	
3. Walls (Up To Height of 6 Feet)		0.5	0.5
b. South Cell			
1. Window		0.1	0.1
2. Access Holes (Not Including Top of Cell)		0.6	
3. Walls (Up To Height of 6 Feet)		3	3
3. WALKWAYS:			
a. Third Level, South		1	0.5
b. Third Level, North		0.5	0.1
c. Second Level, South		1	1
d. Second Level, North		1	1
e. Second Level, West		0.8	0.3
f. First Level, South		0.5	0.5
g. First Level, East		0.8	0.8
h. First Level, North		1.2	1.2
i. First Level, West		0.4	0.4
4. EXPOSED EXPERIMENTAL EQUIPMENT (PIPING, ETC.) ON 2nd. LEVEL BALCONY:	NORTH	1	0.5
	SOUTH	0.5	0.5
5. FACILITIES:			
a. North Facility		7	7
b. South Facility		2	0.5
c. HB-1		15	1
d. HB-2		4	1
e. HB-3		3	0.8
f. HB-4	1	5	1.5
	2	2	2
g. HB-5		5	0.5
h. HB-6		10	10
6. BASEMENT:			
a. Pipe Tunnel		1.5	1
b. Sub-Pile Room Door		0.4	0.4
1. Near Reactor Bottom Plug		25	2
c. North Walkway		1	1
d. South Walkway		12	2
e. Pool Heat Exchanger Area		7	4
f. Beam Hole Plug Storage Area		24	1

UCN-2276
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Table 9.2 (Continued)

AREA	mr/hr	
	MAXIMUM BACKGROUND	PRE-SET LOCATION
6. BASEMENT (Continued)		
g. Degasifier		
1. Air Separator	70	70
2. Water Tank	90	50
h. Pneumatic Tube Laboratory	2	0.8
i. Reactor North Demineralizer		
1. Anion	15	15
2. Cation	25	20
3. Mixed Bed	600	300
j. Reactor South Demineralizer		
1. Anion	450	300
2. Cation	600	300
3. Mixed Bed		18
k. Pool Demineralizer		
1. Anion	12	2
2. Cation	18	1
7. PUMP HOUSE:		
a. #1 Pump Cell	200	170
b. #2 Pump Cell	180	220
c. #3 Pump Cell	180	200
d. Shutdown Pump (Electric) Cell	20	20
e. Emergency Pump (Gasoline) Cell	20	20
8. REACTOR HEAT EXCHANGER PIT:		
a. Entrance	10	5
b. North Side of Pit	20	15
c. South Side of Pit	20	15
9. REACTOR WATER ACTIVITY	57	
10. POOL WATER ACTIVITY	3.8	
11. SUMP COUNTS		
a. #5 (Decay Tank)	230 c/m/ml	
b. #4E (Expansion Pit)	1326 c/m/ml	
c. #4W (Expansion Pit)	680 c/m/ml	
d. #6 (3042)	0 c/m/ml	

the doses indicated are not exclusively due to work at the ORR. The data presented reflect the actual operating history of the ORR. It is perhaps worthy of note that no employee of the ORR operating organization has ever recorded a dose greater than 1 rem from a radiation incident. The highest single dose recorded is 900 mr, which occurred during the handling of a piece of experimental apparatus following its removal from the reactor.

Although the radiation-dose history presented above is ample evidence that the ORR shielding together with the administrative controls utilized to minimize radiation exposure is more than adequate to permit safe operation at power levels well in excess of 30 Mw, it is to be expected that, all other things being equal, the average exposure would increase somewhat at 45 Mw. The existing background levels (30 Mw) near the reactor are given in Table 9.2, which reproduces the results of a typical background survey. The locations referred to in this survey are shown in Figs. 9.2 through 9.6. In interpreting these data, it should be understood that the high-background regions such as the subpile room, degasifier cell, demineralizer cells, pump cells, and the vicinity of the experiment facilities are restricted access areas and not normally occupied during operation.

As a first approximation, it can be assumed that at the 45-Mw power level none of the background levels will increase by a factor of more than 1.5.

Actually it is likely that in many cases the increase would be less than this. Where necessary or desirable, additional local shielding can be provided and access further limited in order to minimize the risk of exposure.

ORNL–DWG 67–7692

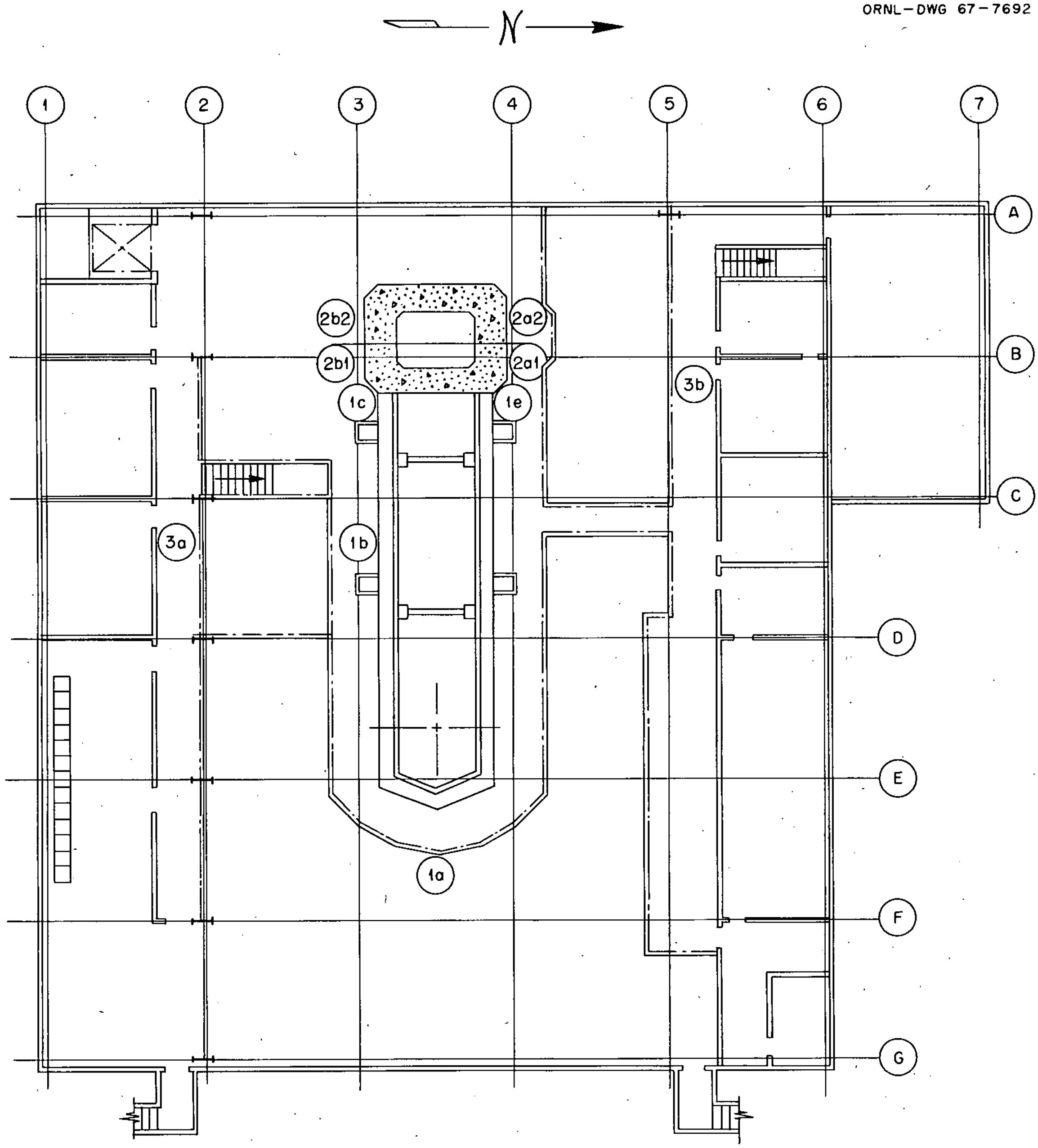

Fig. 9.2. Radiation Survey Points. Third level, ORR.

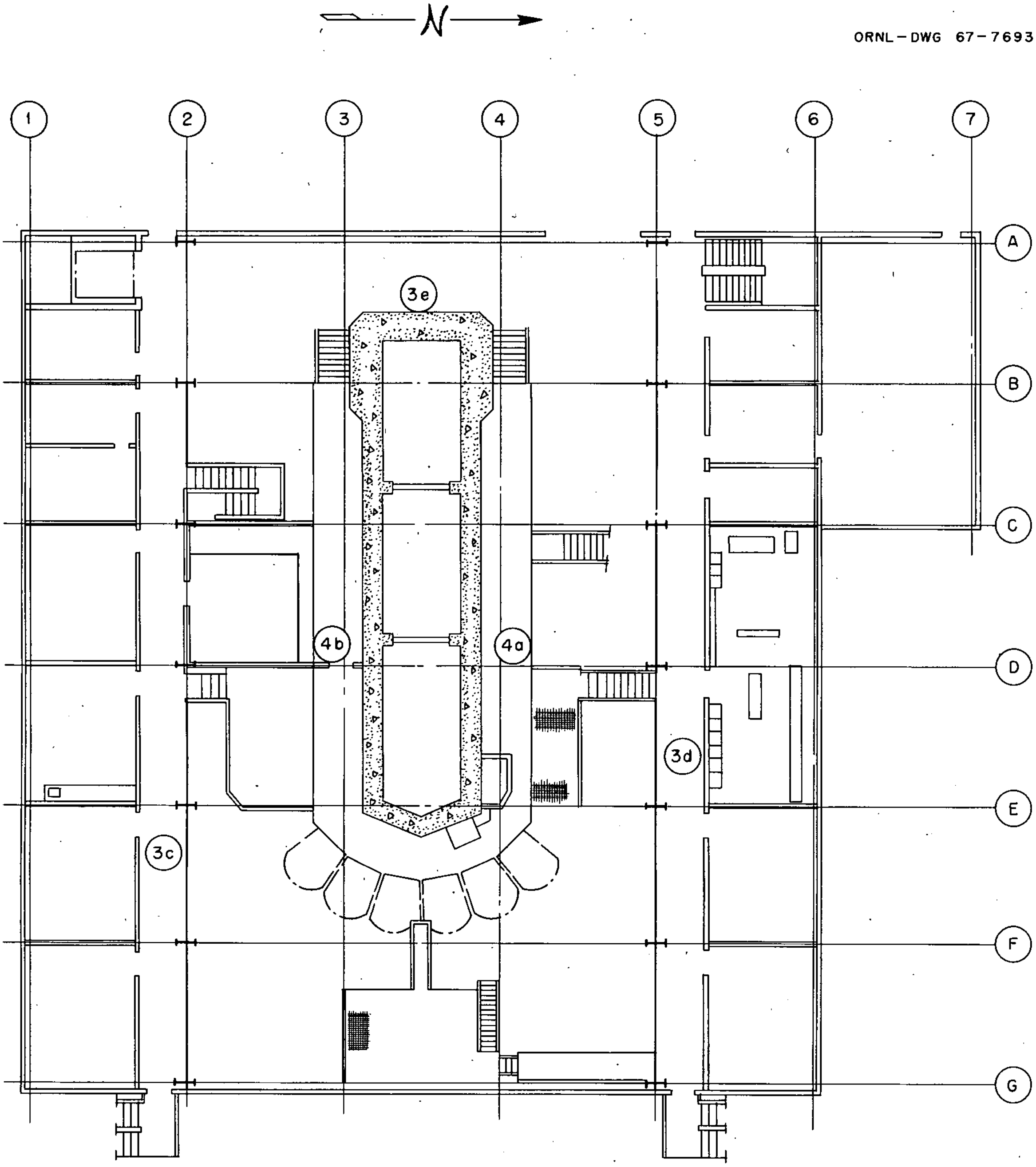

Fig. 9.3. Radiation Survey Points. Second level, ORR.

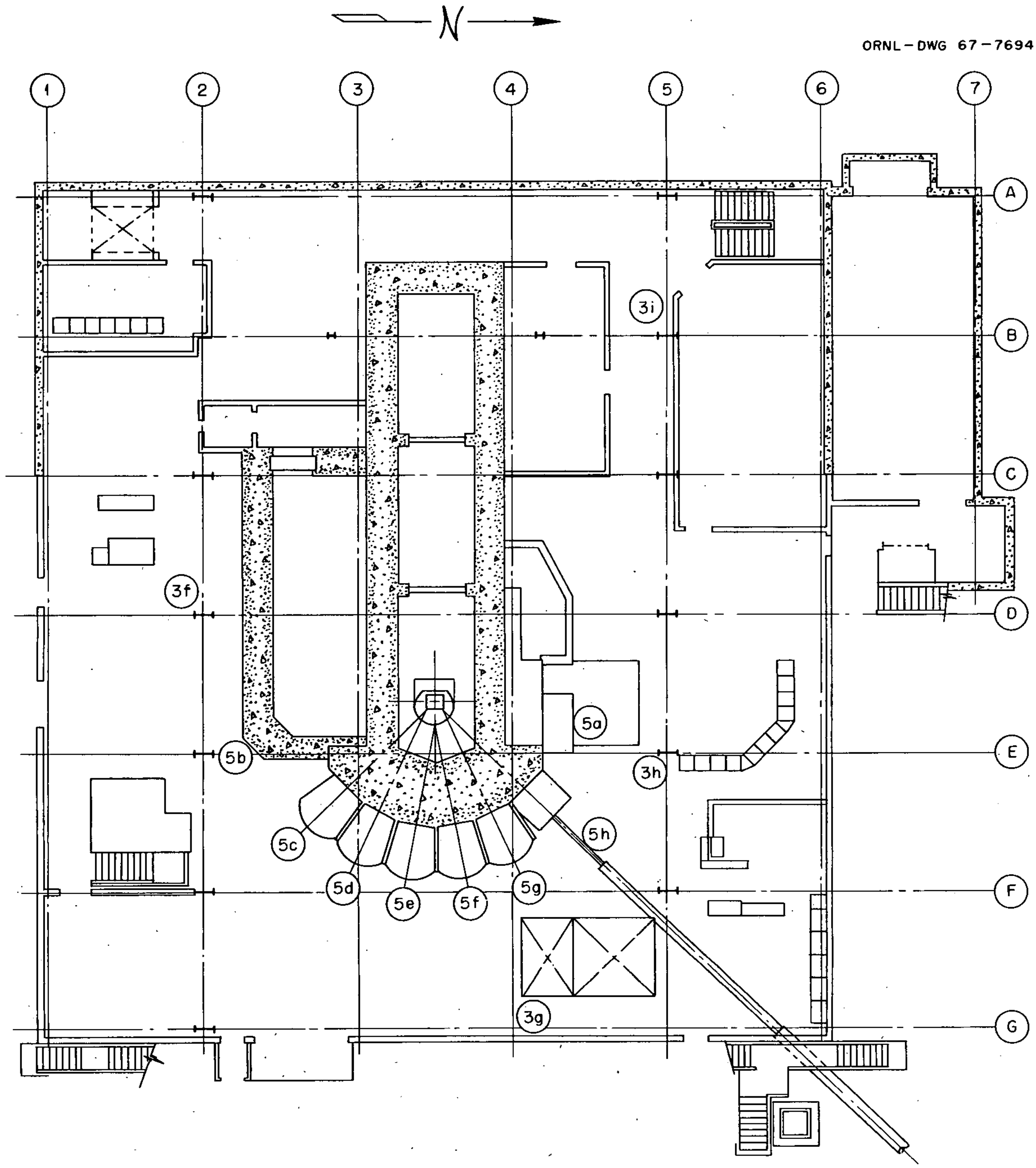

Fig. 9.4. Radiation Survey Points. First level, ORR.

ORNL-DWG 67-7695

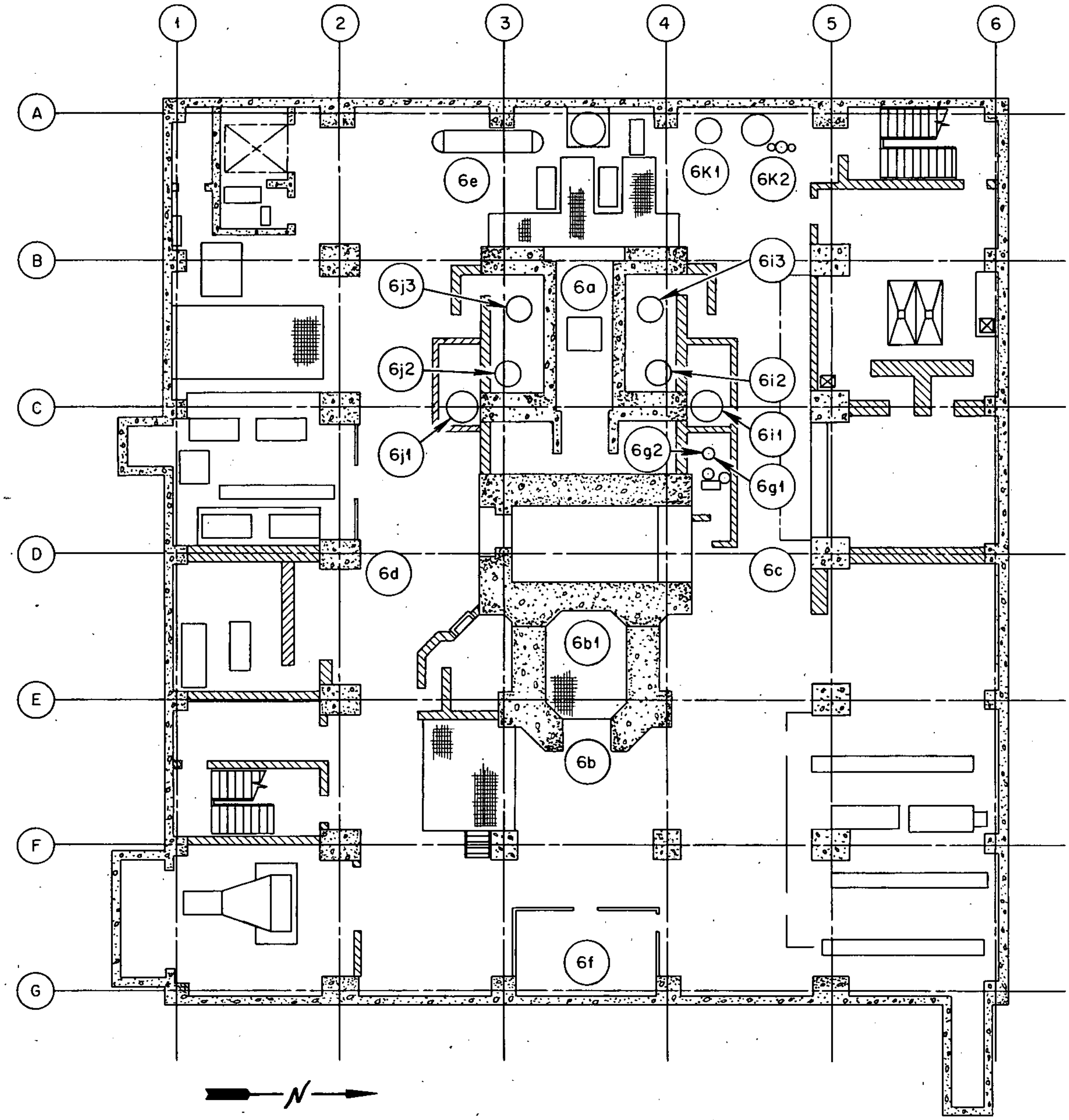

Fig. 9.5. Radiation Survey Points. Basement, ORR.

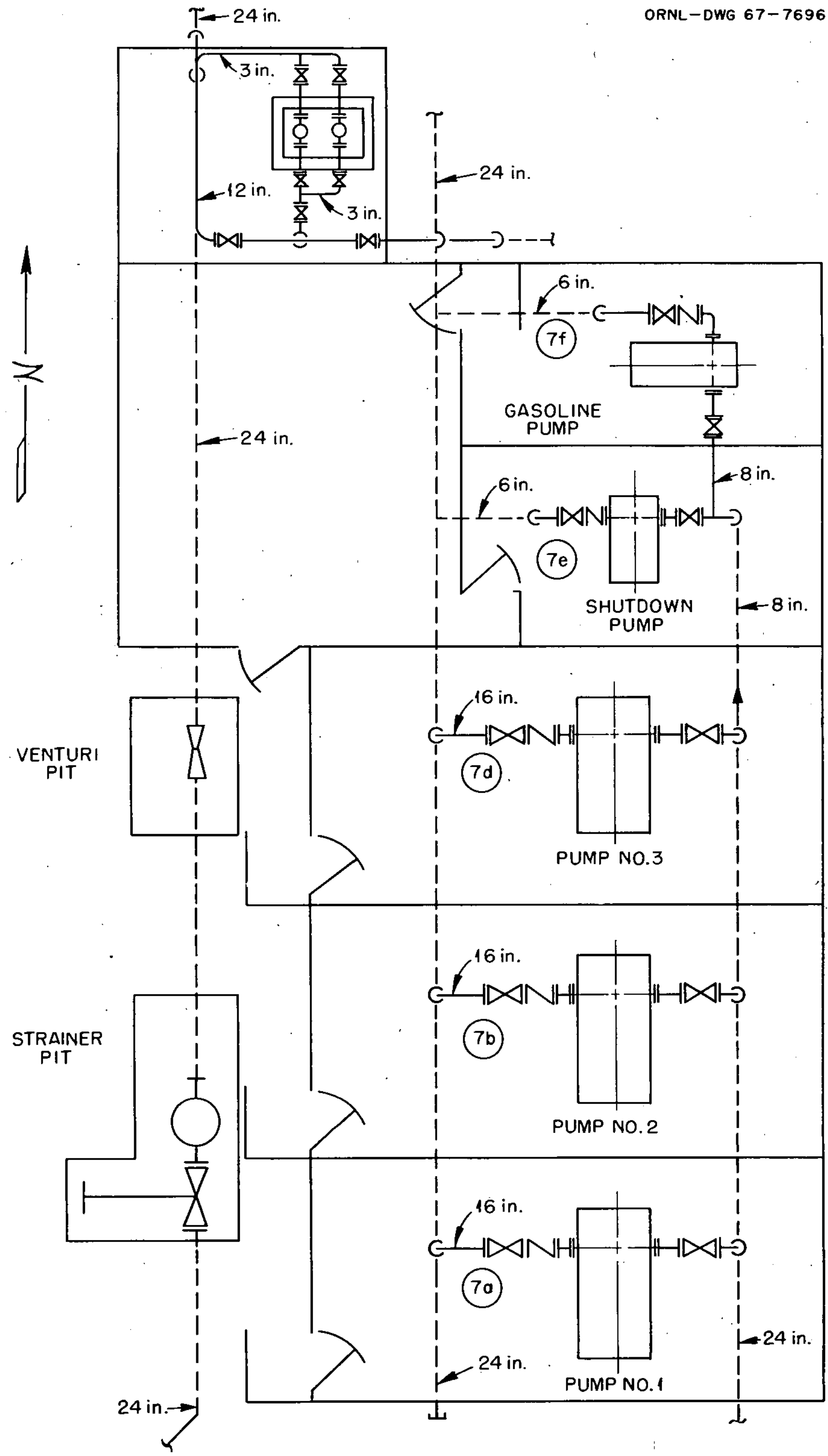

Fig. 9.6. Radiation Survey Points. Primary pump house, ORR.

10. HISTORY OF OPERATING DIFFICULTIES

Operation of the ORR has proceeded in an orderly and uninterrupted fashion during the nearly nine years since the reactor was first brought critical. During this time the ORR has been operated at power approximately 80% of the time, and since January 1, 1964, this figure has increased to more than 85%.

Operating difficulties have been relatively minor and have constantly decreased in frequency and severity. This is due largely to two factors: (1) increasing familiarity with the characteristics of the reactor, which permits developing troubles to be recognized and corrected before they become problems, and (2) constant improvement and modification of components, systems, and procedures which are found lacking in performance.

During the first five years of operation, probably the most vexing problem involved numerous unscheduled control rod drops. The frequency of these occurrences, which in each case shut the reactor down, is shown in Fig. 10.1. In general, the specific cause for a given rod drop has been difficult to determine; however, inspection of the equipment has revealed that in some cases they were caused by faulty components, such as defective magnetic clutches and magnet amplifiers, electrical short circuits, leaks in the bellows seal, and gaps between magnets and armatures. The most frequent cause appears to have been nonuniform thermal expansion of the various components.

The decline in rod drops that occurred from 1959 to 1964 is attributed to a systematic inspection and preventive maintenance program instituted following the onset of the difficulties.

Improved rod-drive mechanisms were installed in June 1964, and since that time there has been only one rod drop; this was due to a human error which occurred during the replacement of a defective magnet amplifier.

Of considerably more significance is the fact that on a number of occasions prior to 1964 a control rod failed to drop upon receipt of a scram signal. The frequency of these occurrences is shown in Fig. 10.2. Of the nine incidents of this type, seven occurred during performance tests of the mechanism. In each of the two which occurred during reactor operation, only a single rod failed to scram; the others behaved in a normal fashion. Each time a "fail-to-drop" incident occurred, the rod drive in question was replaced by an acceptable unit. Since the new rod-drive mechanisms were installed in 1964, no "fail-to-drop" incidents have occurred.

A second type of difficulty has been the frequency with which the safety or control system circuits have been tripped due to some reason other than "real cause."[42] Since 1958 system action has occurred 308 times. Of these, 185 (60%) were

[42] Real cause – a reactor power reduction was initiated by an instrument because an operating limit was exceeded.

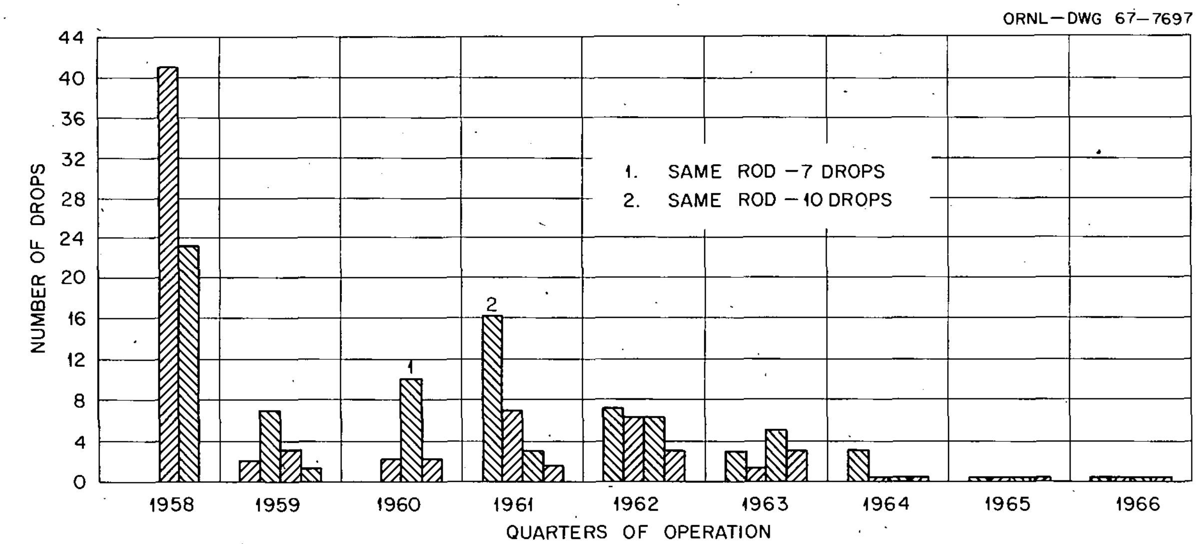

Fig. 10.1. ORR Distribution of Unscheduled Rod Drops.

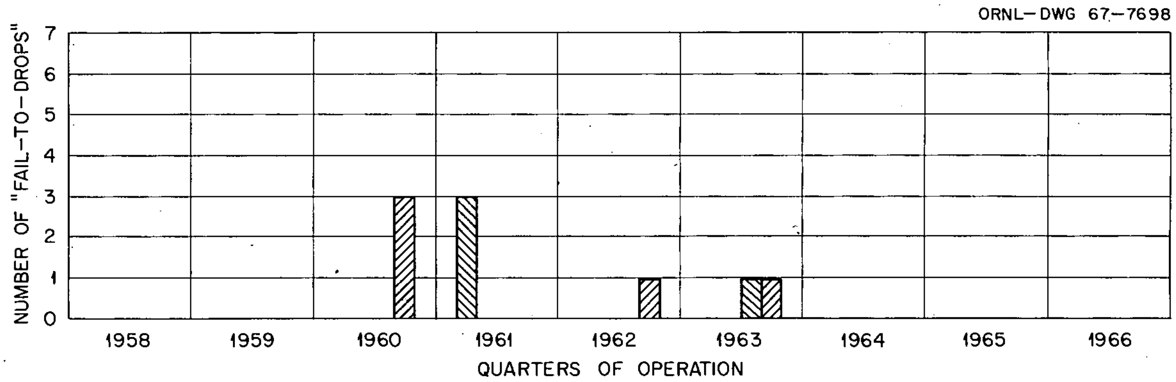

Fig. 10.2. ORR Distribution of "Fail-to-Drop" of Shim Rods.

caused by signals originating in experiment equipment and 123 (40%) by signals originating with the reactor. Of the experiment-induced scrams, only 9 (4.86%) were due to a real cause. The remainder occurred because of component failure, human error, electrical power fluctuation, etc. Of the reactor-induced scrams, only 5 (4.06%) were due to a real cause. A tabulation of these system actions together with their cause is given in Table 10.1.

A breakdown of the incidents included in the "real cause" category is as follows:

1966 – On June 25, when the reactor was returned to power, a setback resulted when the high-temperature set point was exceeded in the P4-B4 experiment.

1962 – On March 14 a setback resulted when the flow in the hydraulic tube dropped below the low-flow set point. This facility is connected to the reactor control circuit only when samples containing fissionable materials are irradiated.

1961 – On January 3 two setbacks were received when the high-temperature set points on the Loop 1 experiment were exceeded.

Table 10.1. Occasions When ORR Safety or Control System Circuits Have Been Tripped

Cause	1959	1960	1961	1962	1963	1964	1965	1966	Total
				Experiments					
Instrument failure	1	22	19	8	0	0	0	3	52
Mechanical or component failure	3	17	5	14	24	3	4	5	76
Real cause	3	1	4	0	0	0	0	1	9
Human error	2	0	3	0	6	0	3	2	16
Electrical power fluctuation	2	0	6	0	7	0	1	0	16
Other	0	0	15	0	1	0	0	0	16
Subtotal	11	40	52	22	38	3	8	11	185
				Reactor					
Instrument failure	5	2	2	0	0	8	1	3	21
Mechanical or component failure	0	1	3	12	5	2	4	4	31
Real cause	1	1	2	1	0	0	0	0	5
Human error	5	5	0	4	2	6	1	5	28
Electrical power outage	5	4	5	5	6	5	3	5	38
Subtotal	16	13	12	22	13	21	9	17	123
Total	27	53	64	44	51	24	17	28	308

On January 22 a setback resulted when the temperature of the reactor exit water exceeded the high-temperature set point.

On September 2 a reverse resulted when the temperature of the reactor exit water exceeded the high-temperature set point.

On September 5 a reverse resulted when the low-flow set point of the Loop 1 experiment was actuated due to the low flow in the experiment.

On September 14 a scram resulted when the high-pressure set point was exceeded when the B-8 experiment encountered excessively high pressure.

1960 – On March 30 a reverse resulted when the temperature differential of the primary water across the reactor exceeded the set point.

On April 23 a setback resulted when the temperature of the GCR capsule experiment exceeded the high-temperature set point.

1959 – On April 12 the high-temperature set point of the GCR capsule experiment was exceeded, which produced a setback.

On May 8 the high-temperature set point of a GCR capsule experiment was exceeded, which produced a setback.

On May 17 a setback was experienced when a safety recorder exceeded the 110% of full power set point.

On May 25 low water flow was experienced in the HN-1 loop experiment, which produced a setback of the reactor power.

Finally it is worth while to list the incidents of higher-than-normal radiation, surface contamination, and high air activity which have occurred since initial reactor startup. These are given in Table 10.2.

The maximum radiation level encountered during any abnormal event was 75 r/hr at 4 ft from the experiment. This occurred when an experiment tube was accidentally removed from the pool water during the desegmenting operation. The maximum exposure received during this incident was 900 mr when personnel who were performing the operation were inadvertently subjected to the high dose rate.

The maximum surface contamination resulted from a "spill" when an experiment was being removed from the reactor facility. This activity was identified as ^{187}W and was particulate matter. The radioactivity level of the highest smear was 400,000 dis/min.

The maximum level of air contamination resulted from a malfunctioning UO_2 experiment ($\sim$0.6 g of ^{235}U). The activities which escaped to the building were identified as ^{138}Xe and ^{138}Cs. All other fission products were retained in the experiment or trapped in the water through which the gases traveled. Although no estimate was made of the total release to the building, a filter from a continuous air monitor gave a reading of $\sim$300 mr/hr. The background radiation level due to ^{138}Xe, ^{138}Cs, and ^{88}Rb at the northwest personnel door of the ORR building was $\sim$60 mr/hr, while the general background due to the same radionuclides in the vicinity of the pool was $\sim$100 mr/hr. It is interesting to note that during the melting of a fuel plate of an ORR fuel element[40] a general background due to the same radionuclides in the vicinity of the pool was $\sim$25 mr/hr, while a filter from a continuous air monitor gave a reading of $\sim$65 mr/hr.

It is important to note that during the above conditions no personnel received either external or internal exposure in excess of the MPC.

When evaluating the data which appear in the table, some additional information should be considered. For example:

1. All of the four building evacuations charged to the reactor were a direct result of the partial fuel-element plate melting.
2. Four of the five instances of high air activity charged to the reactor can be attributed to the fission products in the water system resulting from the partial fuel-element plate melting.
3. The incidents encountered with experiments can be categorically grouped as follows: $\sim$47% due to component failures, and $\sim$53% due to inadequate procedure or human error.

Considering the very large number of experiments handled routinely in the ORR and the operational activities routinely performed, the number and severity of the abnormal events which have been experienced are quite low.

Table 10.2. Abnormal Incidents Involving Radiation

Cause	Reactor	Experiments	Total
High radiation	3	8	11
Surface contamination	2	7	9
High air activity	5	5	10
Total	10	20	30
Building evacuations	4	9	13

11. CONCLUSIONS

It may be concluded from the analysis presented above that the Oak Ridge Research Reactor can be operated at 30 Mw in an entirely safe and orderly fashion. That this is true has been amply demonstrated by the excellent operation record compiled over the past nine years. Moreover, because of experience gained during this time, the operating procedures have been continually improved, and the incidence of component failure has been reduced to a minimum. In addition, this experience has served to increase the skill and knowledge of the operating and technical personnel.

The analysis has shown that even a 100% meltdown of the ORR core would not produce off-area environmental contamination in excess of the guidelines given in 10 CFR 100 and would not produce catastrophic results within the Laboratory.

The foregoing conclusions are equally valid for operation at a power level of 45 Mw, which could easily be made feasible by relatively minor modifications to the external heat-removal system.

APPENDIX A – STATUS OF SPECIFIC ACRS AND AEC RECOMMENDATIONS WITH RESPECT TO ORR OPERATIONS AS OF JANUARY 1, 1962[1]

The following list of documents includes all those which may be considered to constitute the original authorization for operation of the ORR.

March 25, 1955
S. R. Sapirie (USAEC) to C. E. Center (CCCC)
Subject: ACRS and Budget Approval for 20 Mw ORR

This letter gives authorization to proceed with construction. It asserts that ACRS has given approval for 20 Mw power level in accordance with the descriptions given in ORNL-1794, *ORR Safeguard Report*; CF-55-3-8 (later issued as ORNL-2086), *A Method for the Disposal of Volatile Fission Products from an Accident in the ORR*; and CF-54-11-115, *Advisory Committee on Reactor Safeguards ORR* (this is a letter from C. E. Larson summarizing the answers to some questions brought up in an ACRS meeting held at Oak Ridge).

In general, construction and operation at 20 Mw were approved subject to the provisions that containment be provided by means of an emergency ventilation system and scrubber and that the excess reactivity be minimized.

December 30, 1955
S. R. Sapirie (USAEC) to H. L. Price (USAEC)

This is an intra-office memorandum in reply to a request from Price, dated November 8, 1955, requesting information on recommendations made by RSC and ACRS in connection with reactor safeguards at Oak Ridge.

It is asserted that "nearly all existing limitations on reactor operations necessitated by safety are self-originated at ORNL. In some cases these limitations developed as a result of informal review by ACRS Committee members."

A list of ACRS recommendations is included. These are the same ones contained in the letter of March 25, 1955. They are as follows (quoted from the December 30, 1955, letter):

"Containment is to be provided to the extent that an accident releasing the volatile fission product gases from the reactor core will not constitute a wide-spread hazard."

"Available excess reactivity is to be minimized insofar as practical."

"That experimental installations in the engineering test facilities be restricted such that upon failure or malfunction of the installation no more than 1.4% reactivity will be added to the reactor."

December 3, 1957
J. A. Swartout (UCNC) to H. M. Roth (USAEC)

This letter is a request to permit testing of the ORR at power levels in excess of 20 Mw. No limit on power is suggested, and the request is for temporary authorization.

February 17, 1958
H. M. Roth (USAEC) to J. A. Swartout (UCNC)

This is in reply to the December 3, 1957, letter and advises that the "available information is

[1]Internal correspondence dated Feb. 8, 1962.

insufficient for Headquarters' review and action." A memorandum from H. L. Price to P. W. McDaniel listing the required information is enclosed. This memorandum, dated February 6, 1958, points out that the AEC has no information concerning the incorporation of the ACRS suggestions into the ORR as built. It is requested that quantitative information on the features of the reactor which relate to safety be supplied. Certain specific information is requested.

February 28, 1958
J. A. Swartout (UCNC) to H. M. Roth (USAEC)

This letter is in reply to the letter of February 17, 1958, and the memorandum from Price to McDaniel of February 6, 1958. It is pointed out that the design has changed little from that set forth in ORNL-1794, *ORR Safeguards Report.* The specific questions asked in the February 6, 1958, memorandum are answered in most instances by reference to various documents. These include:

ORNL-2200, *The Oak Ridge Research Reactor,* which updates the description in ORNL-1794.

CF-57-5-31, *Two Group Calculations for Flux Distribution and Critical Mass in Clean, Cold ORR Cores.*

CF-54-11-115, *Advisory Committee on Reactor Safeguards ORR,* which discusses the minimization of excess reactivity.

CF-58-2-11, *Preliminary Report on the Results of the Oak Ridge Research Reactor Hydraulic Test,* which verifies the hydraulic design basis experimentally.

CF-55-3-8 (ORNL-2086), *A Method for the Disposal of Volatile Fission Products in the ORR.* It is pointed out that this report overestimates the maximum credible accident by a factor of at least 100.

A listing of the experiments scheduled for installation in the ORR as of that time is included.

In this letter it is asserted with respect to testing the building containment: "The first test conducted using the scrubber to withdraw air from the ORR Building resulted in a building pressure of –0.3 in. of H_2O at a flow of 6000 cfm. This pressure and flow are considered quite satisfactory." In this letter the scrubber specifications are set forth. These include a decontamination factor of 1000 for iodine. Also at this time the emergency ventilation system and scrubber were described as being actuated either manually from the ORR control room or by signal from a shielded ion chamber near the control room.

March 18, 1958
H. M. Roth (USAEC) to J. A. Swartout (UCNC)

This letter permits interim operation of the ORR at levels not to exceed 1 Mw pending review at Headquarters of the information contained in the letter of February 28, 1958. It contains certain restrictions which are termed "temporary limitations."

March 28, 1958
H. M. Roth (USAEC) to J. A. Swartout (UCNC)

This letter supplements the letter of March 18, 1958, and modifies the "temporary limitations" for operation below 1 Mw. It further tacitly permits operations above 1 Mw "for the initial exploratory operation." These are as follows:

a) "The core will not be operated with less than four or more than six control rods."

b) "The excess reactivity loading above clean cold critical will not exceed that which will permit achievement of criticality with the rods withdrawn less than $\frac{1}{2}$ their reactivity worths."

c) "No experiments, voids or plastic strip poisons will be loaded in or adjacent to the core."

d) "Short exploratory and check-out power experiments may be performed to levels of 40 Mw."

e) "Average power levels will not exceed 20 MWD/day."

It appears from the context of this letter that these restrictions were meant to apply only during the testing period and were to be superseded by other limitations once the testing program was completed.

May 22, 1958
H. M. Roth (USAEC) to J. A. Swartout (UCNC)

This letter notifies ORNL that routine operation of the ORR at 30 Mw is authorized subject to the "satisfactory completion of the initial exploratory experimental program within the limitations outlined in my March 28 letter." This communication does nothing specific to resolve the question concerning the permanent or temporary nature of the restrictions set forth in the letter of March 28, 1958.

April 27, 1959
M. E. Ramsey (UCNC) to H. M. Roth (USAEC)

In this letter a basis for reducing the required iodine decontamination factor in the scrubber from 1000 to 10 is presented. It is further stated: "Measurements of the differential building-to-atmosphere pressure were also made, which showed that more than 0.3 inch differential pressure is maintained under operating conditions. This is quite adequate to insure the necessary in-leakage."

May 8, 1959
S. R. Sapirie (USAEC) to E. J. Bloch (USAEC)

This letter transmits Ramsey's letter of April 27, 1959, to AEC Headquarters. It asserts that a decontamination factor of 100 was measured at the scrubber and recommends that a reconsideration of the scrubber system factor is not necessary. No mention of the factor of 10 is made in the transmittal letter.

On the basis of the foregoing correspondence, it appears clear that the specific ACRS and AEC recommendations with respect to ORR operation include the following:

1. Containment is to be provided to the extent that an accident releasing the volatile fission product gases from the reactor core will not constitute a widespread hazard.
2. Available excess reactivity is to be minimized insofar as practical.
3. Experimental installations in the engineering test facilities are to be restricted so that upon failure or malfunction of the installation no more than 1.4% reactivity will be added to the reactor.
4. "Routine" operation of the ORR may be conducted at power levels up to 30 Mw.

APPENDIX B – THERMAL ANALYSIS OF THE ORR CORE FOR 45-Mw OPERATION

C. C. Webster

1. Introduction

The following thermal analysis of 45-Mw operation of the ORR core is based partially upon calculations using conventional and well-substantiated methods and partially upon direct extrapolation of observations made at power levels up to 30 Mw. Because the core configurations and fuel distribution at 45 Mw will differ little from those employed at 30 Mw, the excellent agreement in the values obtained by the two methods permits the results to be viewed with considerable confidence.

2. Power Density and Neutron Flux Distribution

The ORR core contains 63 core-component positions. Of these, six are capable of accepting any core component, including a shim-safety rod. The remaining 57 are capable of accepting any core component except a shim-safety rod. Under current practice, a normal core loading contains either four or six shim-safety rods and 25 or 26 fuel elements. The remaining positions are occupied by beryllium reflector pieces, solid aluminum spacers, or experimental apparatus of one kind or another. Four typical core loadings are shown in Figs. B.1 through B.4. Although there is no intention that the loading patterns be restricted to those shown, they are representative of the configurations which have been in use for several years and will be employed as reference configurations for this study.

The fuel content of the individual elements will, of course, decrease with time during the fuel cycle, which lasts two to three weeks depending upon the initial loading.[1] Moreover, the beginning-of-cycle fuel loading is not uniform but is composed of a combination of new and partially depleted fuel elements ranging in fuel content from about 135 to 240 g of ^{235}U. The weight of fuel in the shim-rod fuel followers may vary from 0 to 154 g of ^{235}U.

[1]The amount of fuel initially loaded may vary from 5 to 6 kg of ^{235}U. It is governed by the requirement that the reactor shall be subcritical when the shim-safety rods are withdrawn less than one-half of the total reactivity worth of the rod complement.

ORR CORE

ORNL-DWG 67-7699

POOL
W

A-1	A-2	A-3	A-4 (188) 0.770	A-5 (240) 0.812	A-6 (181) 0.575	A-7	A-8	A-9 EX
B-1 EX	B-2	B-3 (201) 1.223	B-4 (104) 0.671	B-5 (189) 1.272	B-6 (99) 0.536	B-7 (240) 0.873	B-8 EX	B-9 EX
C-1 EX	C-2	C-3 (157) 1.336	C-4 (146) 1.305	C-5 (146) 1.303	C-6 (147) 1.263	C-7 (145) 0.947	C-8 (154) 0.719	C-9
D-1 (133) 0.505	D-2 (240) 1.244	D-3 (168) 1.448	D-4 (142) 1.050	D-5 (140) 1.316	D-6 (146) 0.829	D-7 (148) 1.008	D-8	D-9
E-1 (188) 0.602	E-2 (181) 0.895	E-3 (240) 1.424	E-4 (172) 1.224	E-5 (201) 1.498	E-6 (180) 1.220	E-7 (240) 1.211	E-8 (184) 0.927	E-9
F-1 EX	F-2	F-3 EX	F-4	F-5	F-6	F-7	F-8	F-9
G-1	G-2	G-3	G-4	G-5	G-6	G-7	G-8	G-9

S (left) — N (right)

E

$$\bar{P}_c = \frac{M_i \, \bar{\Phi}_i / \bar{\Phi}_c}{\sum_{i=1}^{29} M_i \, \bar{\Phi}_i / \bar{\Phi}_c} \, (30)$$

(M) — MASS ^{235}U

--- — Mw/ FUEL ELEMENT

$\bar{\Phi}_i/\bar{\Phi}_c$ — MEASURED NEUTRON FLUX

EX — IN CORE EXPERIMENTS

Fig. B.1. Power Output at Start of Full-Power Operation. Cycle 59-C.

ORR CORE

ORNL-DWG 67-7700

POOL
W

A-1	A-2 EX	A-3	A-4 (194) 0.665	A-5 (240) 0.687	A-6 (192) 0.709	A-7	A-8 (149) 0.349	A-9 EX
B-1 EX	B-2 EX	B-3 (240) 1.038	B-4 (93) 0.451	B-5 (181) 1.010	B-6 (102) 0.596	B-7 (220) 1.000	B-8 EX	B-9 EX
C-1 EX	C-2	C-3 * (119) 1.023	C-4 (170) 1.490	C-5 (151) 1.228	C-6 (154) 1.116	C-7 (163) 0.940	C-8 (178) 0.888	C-9
D-1 (218) 0.919	D-2 (240) 1.207	D-3 (200) 1.390	D-4 (141) 0.860	D-5 (180) 1.444	D-6 (141) 0.881	D-7 (205) 1.183	D-8	D-9
E-1 (221) 0.625	E-2 (191) 0.926	E-3 (240) 1.305	E-4 (183) 1.245	E-5 (222) 1.494	E-6 (184) 1.245	E-7 (240) 1.224	E-8 (218) 0.862	E-9
F-1 EX	F-2 EX	F-3 EX	F-4	F-5	F-6	F-7	F-8	F-9 EX
G-1	G-2	G-3	G-4	G-5	G-6	G-7	G-8	G-9

S (left) — N (right)

E

C-3* PARTIAL ELEMENT WITH EXPERIMENT

$$\bar{P}_i = \frac{M_i \, \bar{\Phi}_i / \bar{\Phi}_c}{\sum_{i=1}^{30} M_i \, \bar{\Phi}_i / \bar{\Phi}_c} \, (30)$$

(M) — MASS ^{235}U

--- — Mw/ FUEL ELEMENT

$\bar{\Phi}_i/\bar{\Phi}_c$ — MEASURED NEUTRON FLUX

EX — IN CORE EXPERIMENTS

Fig. B-2. Power Output at Start of Full-Power Operation. Cycle 62-A.

ORR CORE

ORNL–DWG 67–7701

POOL
W

A-1	A-2	A-3	A-4 206 0.873	A-5 240 0.922	A-6 209 0.747	A-7	A-8 141 0.374	A-9 EX
B-1 EX	B-2 EX	B-3 240 1.565	B-4 88 0.635	B-5 198 1.339	B-6 97 0.572	B-7 216 0.840	B-8 EX	B-9 EX
C-1 EX	C-2 156 0.896	C-3 175 1.010	C-4 233 1.343	C-5 235 1.351	C-6 211 1.217	C-7 181 1.044	C-8 130 0.751	C-9
D-1 197 0.530	D-2 EX	D-3 176 1.265	D-4 141 1.038	D-5 178 1.521	D-6 141 1.062	D-7 178 1.055	D-8	D-9
E-1 212 0.497	E-2 188 0.743	E-3 240 1.178	E-4 168 1.172	E-5 224 1.583	E-6 169 1.075	E-7 240 1.078	E-8 196 0.726	E-9
F-1 EX	F-2 EX	F-3	F-4	F-5	F-6	F-7 EX	F-8	F-9 EX
G-1	G-2	G-3	G-4	G-5	G-6	G-7	G-8	G-9

S (left) N (right)

E

$$\bar{P}_i = \frac{M_i\,\bar{\Phi}_i/\bar{\Phi}_c}{\sum_{i=1}^{30} M_i\,\bar{\Phi}_i/\bar{\Phi}_c}\,(30)$$

(M) — MASS ^{235}U

--- — Mw/FUEL ELEMENT

$\bar{\Phi}_i/\bar{\Phi}_c$ — MEASURED NEUTRON FLUX

EX — IN CORE EXPERIMENTS

Fig. B.3. Power Output at Start of Full-Power Operation. Cycle 63-A.

ORR CORE

ORNL–DWG 67–7702

POOL
W

A-1	A-2	A-3	A-4 183 0.783	A-5 201 0.831	A-6 180 0.687	A-7	A-8 140 0.450	A-9 EX
B-1 EX	B-2	B-3 201 1.059	B-4 78 0.444	B-5 192 1.308	B-6 77 0.546	B-7 195 0.924	B-8 EX	B-9 EX
C-1 EX	C-2 201 0.903	C-3 148 1.104	C-4 171 1.335	C-5 168 1.449	C-6 166 1.350	C-7 170 1.026	C-8 171 0.828	C-9
D-1 201 0.594	D-2	D-3 170 1.152	D-4 121 0.433	D-5 169 1.428	D-6 121 1.038	D-7 188 1.197	D-8	D-9
E-1 201 0.585	E-2 186 0.828	E-3 196 1.170	E-4 181 1.356	E-5 197 1.533	E-6 184 1.242	E-7 195 1.101	E-8 202 0.828	E-9
F-1	F-2	F-3 EX	F-4	F-5	F-6	F-7	F-8	F-9 EX
G-1	G-2	G-3	G-4	G-5	G-6	G-7	G-8	G-9

S (left) N (right)

E

$$\bar{P}_c = \frac{\bar{\Phi}_i/\bar{\Phi}_c}{\sum_{i=1}^{30} M_i\,\bar{\Phi}_i/\bar{\Phi}_c}\,(30)$$

(M) — MASS ^{235}U

--- — Mw/FUEL ELEMENT

$\bar{\Phi}_i/\bar{\Phi}_c$ MEASURED NEUTRON FLUX

EX — IN CORE EXPERIMENTS

Fig. B.4. Power Output at Start of Full-Power Operation. Cycle 64-A.

Although the beginning-of-cycle fuel distribution for any given configuration is not always exactly the same, in general the loading procedure is repetitive so that the loading pattern is similar from cycle to cycle. It follows that changes in the neutron flux pattern from cycle to cycle are also small. This has been frequently confirmed by measurements. Since the power density is proportional to the product of the fuel density and the local neutron flux, and since for a particular location changes in the local neutron flux vary inversely as changes in the fuel density, any local changes in power density are quite small.

The experiment load in the reactor is reasonably static. From time to time, however, some experiments are removed and others are added. These changes, together with changes in the fuel distribution, can be expected to alter the neutron flux distribution and to change the power output of the individual fuel locations. These effects have been carefully studied over the past several years. The power output of the individual fuel elements is shown in Figs. B.1 to B.4. The average power of the ith fuel element is determined by the equation on the right of the figure, where $\overline{P}_i$ = the average power of the element, M_i = mass of ^{235}U in the element, $\overline{\phi}_i$ = the average neutron flux in the ith element, $\overline{\phi}_c$ = the average neutron flux in the core, $\sum_{i=1}^{m}$ means to sum the product $M_i\overline{\phi}_i/\overline{\phi}_c$ over the number (m) of elements in the core, and the number (30) represents the 30-Mw power level. As can be seen, the variation in the power output in any given position is quite small, despite the rather significant differences in fuel loading. Perhaps the most striking change is that which occurred in position D-3, where replacement of a fuel element in position D-2 by a highly absorbing experiment drastically reduced the power output in position D-3. It is the practice to make a careful analysis of all such changes and, when deemed necessary, to perform low-power tests on the changed configuration in order to ensure that the power densities developed are not excessively high.

On the basis of past investigations, positions E-5 and D-4 have been selected as typical core positions upon which to base the thermal analysis: E-5 because it has consistently exhibited the highest power level of any core position and D-4 because it has the highest power output among the shim-safety rod fuel followers. Since beginning-of-cycle conditions are most severe, this condition has been chosen as a basis for the analysis. At that time the four control rods are banked at about 15 in. withdrawn. The thermal neutron flux in each of these positions has been measured on numerous occasions and with a number of variations of the core configuration shown, and the axial distribution of the flux is given in Fig. B.5. It can be seen that the three distributions shown for each position differ little from one another, despite the fact that they were measured for different core configurations and different fuel concentrations. The axial distribution of the fuel density in partially depleted fuel elements has been calculated by Colomb[2] based upon fission-product distribution measure-

[2] A. L. Colomb, *Fission Product Distribution in ORR Fuel Elements*, ORNL-2897 (May 5, 1960).

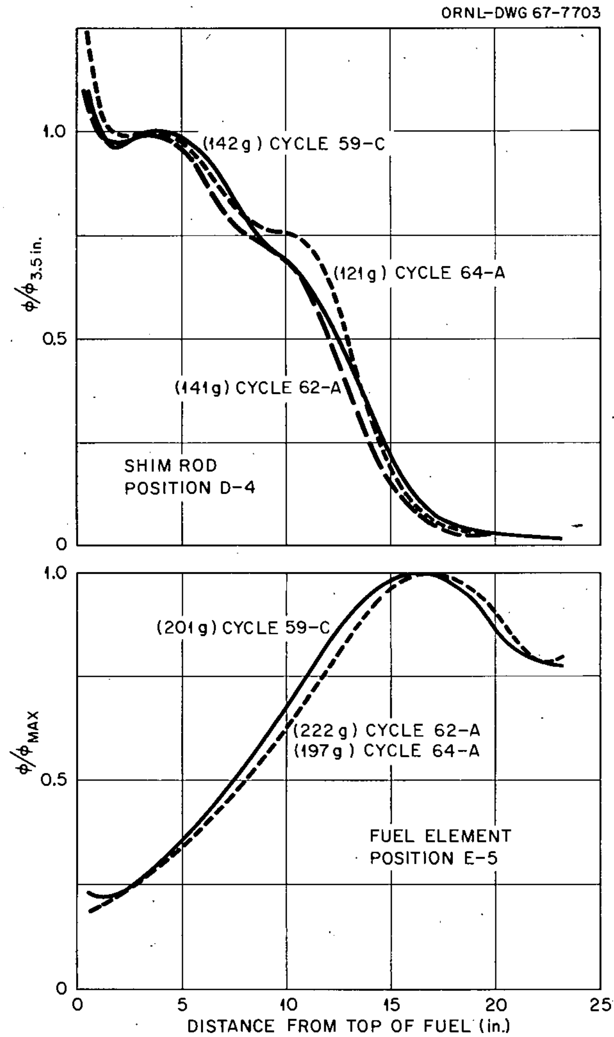

Fig. B.5. Axial Neutron Flux Distribution in ORR Core Positions E-5 and D-4.

ments. Upon utilizing the macroscopic fission cross section as a function of position deduced from Colomb's work (see Fig. B.6) and the measured flux distribution from Fig. B.5, the axial power distribution in the reference positions was computed. The results are shown in Fig. B.7. Extrapolation from 30 to 45 Mw is accomplished by merely multiplying the ratio of the power levels.

It is worth noting that the ratio of peak to average power density in the fuel element is only about 1.7; over that portion of the shim-rod follower within the reactor (~16 in.), it is only 1.6. Moreover, these ratios change little with burnup. Because, in general, lower-weight fuel elements are loaded near the center of the core and the newer or higher-weight elements are loaded on the periphery of the core, the radial power density distribution is also flattened.

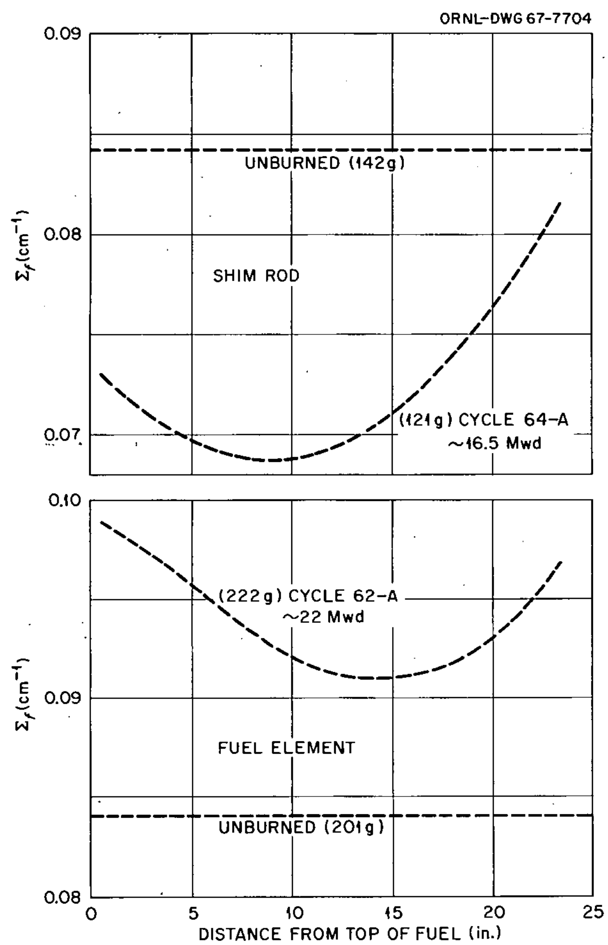

Fig. B.6. Macroscopic Fission Cross Section in ORR Fuel.

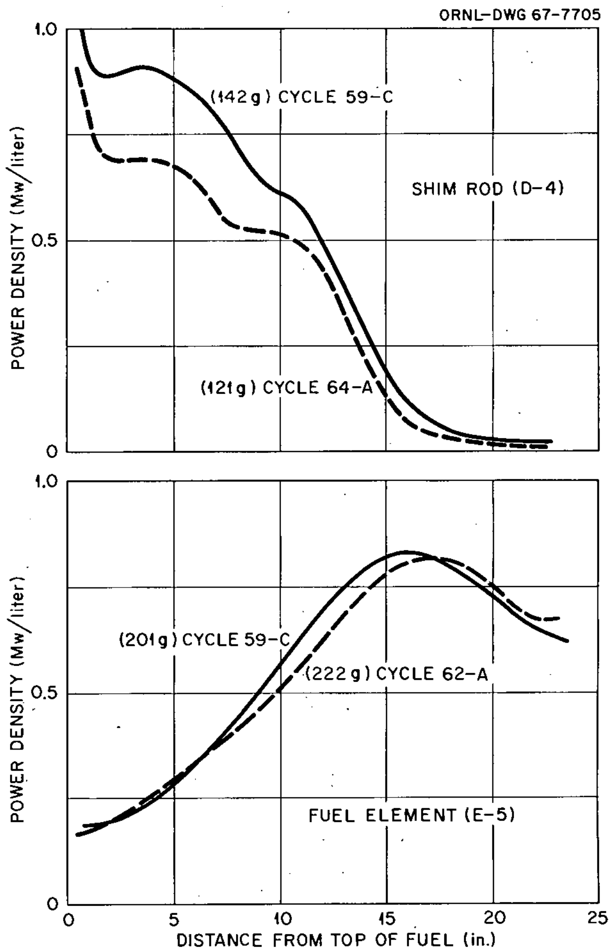

Fig. B.7. Power Density Profile of ORR at 30 Mw.

3. Fuel-Plate Surface Temperatures

The fuel-plate surface temperatures for the positions of interest were determined using heat fluxes obtained from the power-density distributions computed as described above and from forced-convection heat-transfer coefficients derived from a series of experimental observations made using rectangular cooling channels of the type found in ORR fuel.[3] The results for 30- and 45-Mw operations are given in Figs. B.8 and B.9 respectively.

[3]W. R. Gambill and R. D. Bundy, *HFIR Heat Transfer Studies of Turbulent Water Flow in Thin Rectangular Channels*, ORNL-3079 (June 5, 1961).

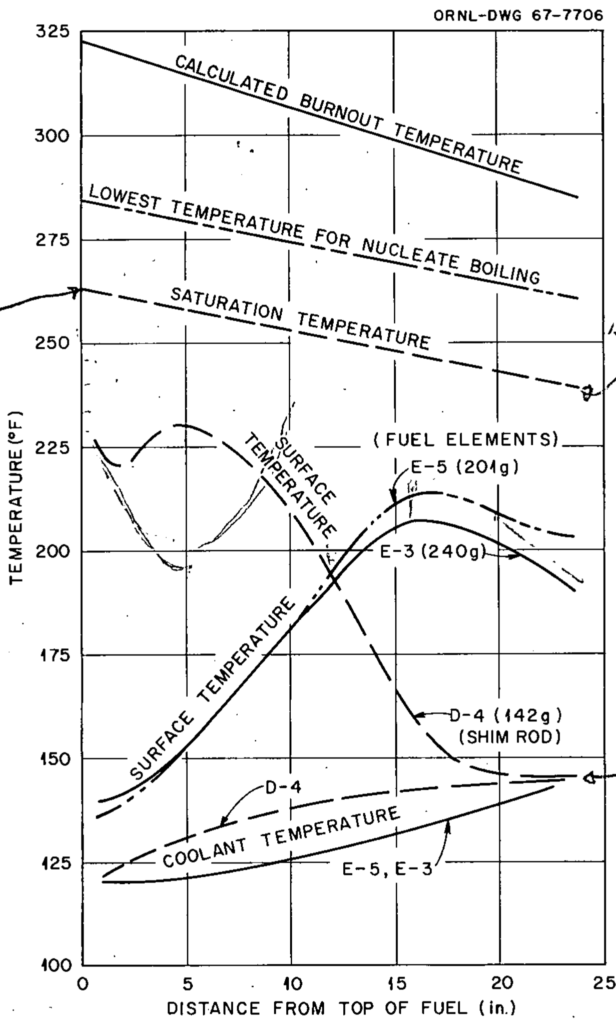

Fig. B.8. Temperature Distribution in Hottest ORR Channels at 30-Mw Operation. Inlet temperature, 120°F; flow, 17,300 gpm (28.8 fps).

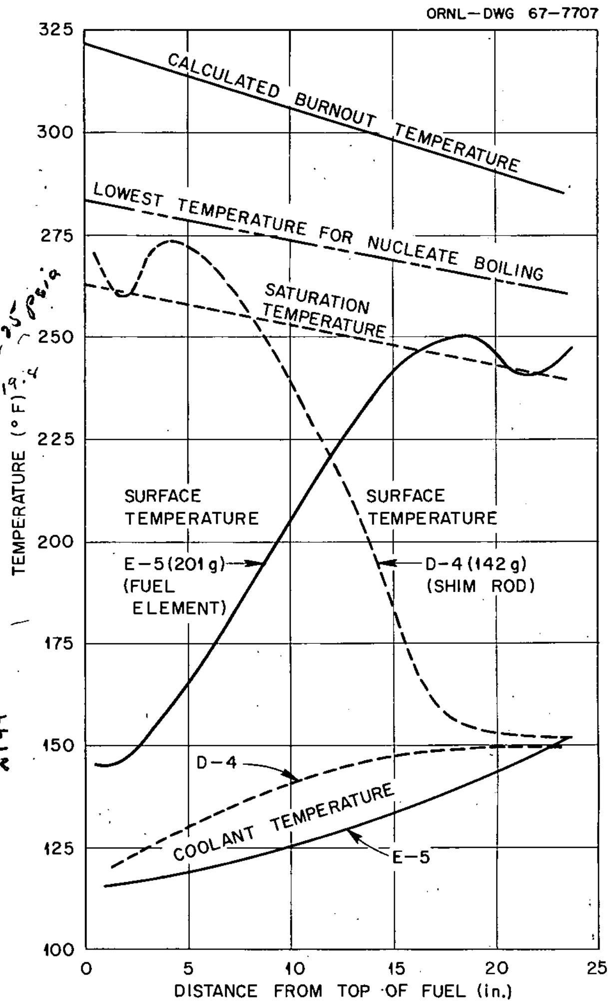

Fig. B.9. Temperature Distribution in Hottest ORR Channel at 45-Mw Operation. Inlet temperature, 115°F; flow, 17,300 gpm (28.8 fps).

These figures also show the calculated burnout wall temperature, the coolant saturation temperature, the estimated minimum wall temperature at the onset of nucleate boiling, and the bulk coolant temperature. The range of values of the heat-transfer coefficients used is also shown.

4. Boiling Experiments – Calculations and Experimental Measurements – Comparison with Earlier Experiments and MIT Data

On May 7, 1965, a boiling experiment was performed in the ORR. In this experiment a new 200-g ^{235}U fuel element was equipped with an orifice to reduce coolant flow and three thermocouples to permit observation of the temperature rise in the fuel element. Boiling was observed to occur at a power level of 5.6 Mw and a coolant velocity of 1.88 fps. The conditions to be expected under these circumstances were calculated using the methods outlined above. As can be seen from the results shown in Fig. B.10, the calculated values agree quite well with those actually observed. The computed maximum surface temperature of 269°F is within the boiling range, and the computed bulk water temperature agrees with the measured values.

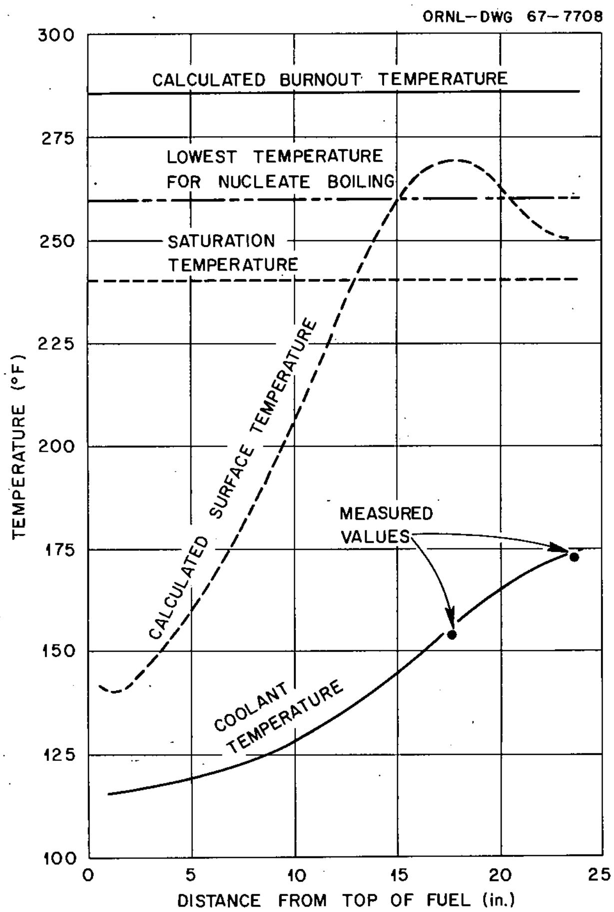

Fig. B.10. Temperature Distribution in Hottest Channel During ORR Boiling Test May 7, 1965, at 5.6 Mw. Flow, 1.88 fps.

During the ORR approach-to-power tests in 1958, a similar series of boiling experiments was conducted.[4] From the results of these tests, a curve of power at which boiling begins vs coolant flow was developed. This curve is shown in Fig. B.11, and it is interesting to note that, since a coolant velocity of 1.88 fps corresponds to a total reactor coolant flow of ~1200 gpm, the values of power level and flow found in the most recent test fall on the curve developed in 1958.

At flows above 1500 gpm, the relation between the power at which boiling begins, P (Mw), and reactor coolant flow, F (gpm), is

$$P = 0.0244F^{0.8} .$$

[4]J. A. Cox *et al.*, *ORR Operations for Period April 1958 to April 1959*, ORNL-CF-59-8-39 (1960).

Substitution of the normal reactor flow of 17,300 gpm into this relation reveals that under normal operating conditions boiling would be expected to begin at about 60 Mw. If the reactor were operated just above the low-flow scram point, 14,000 gpm, boiling would begin at about 51 Mw.

Curves were developed from experimental data at the MIT Heat-Transfer Laboratory.[5] The inception of boiling is considered to occur at the $\Delta T = T_w - T_s$ when the term $[(Q/A) - (Q/A)_{FC}]/(Q/A)$ becomes greater than zero. The correlation curve relating the dimensionless term Re to $(Nu_b/Pr_b^{0.4})$ is used to determine $(Q/A)_{FC}$ (forced-convection heat-transfer term); Q/A is determined from the measured power input to the test section and the surface area. From the experimental data obtained with different coolant flow rates and power input, a curve was prepared which gives the relationship of heat flux to ΔT (wall temperature minus coolant saturation temperature) when incipient boiling occurs for specific tube diameters and pressures. With an inlet water temperature of 115° and bulk water temperature of 147° in the fuel element, it is determined that the fuel-plate surface will reach the temperature for incipient boiling at 30°F above the saturation temperature. Using this ΔT and a pressure of 27.8 psia, the heat flux at which boiling will begin, as determined from the curve described above, is 10.12×10^5 Btu hr^{-1} ft^{-2}, which

[5]A. E. Bergles and W. M. Rohsensow, *Forced Convection Surface Boiling Heat Transfer and Burnout in Tubes of Small Diameter*, NP-11831 (May 25, 1962).

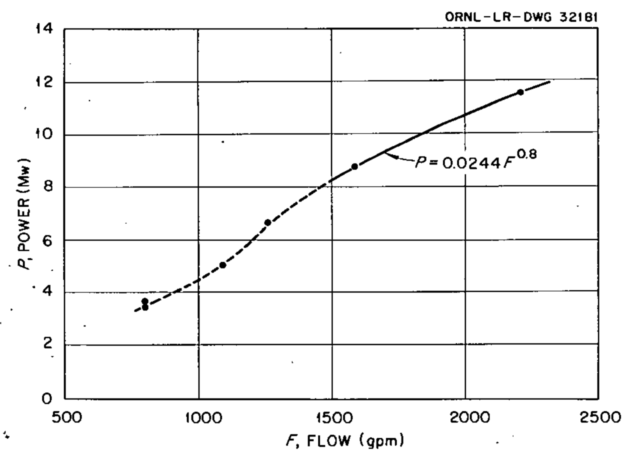

Fig. B.11. Power Level vs Flow at Which Boiling Commences from 1958 Experiment Compared with 1965 Boiling Test.

is equivalent to the heat flux at the point of maximum temperature in E-5 at a reactor power of 58.5 Mw.

The MIT data were gathered from experiments performed with round tubes instead of rectangular channels but with the same L/D ratio, equivalent diameter, flow rate, and inlet temperature as the ORR channels. Using the $\Delta T = T_w - T_s$ of 30°F and the same method of calculation used for the rest of this thermal analysis,[6] the reactor power level at which boiling can be expected to occur in position E-5 with normal operating conditions is calculated to be 58 Mw.

These results agree quite well with the value of 60 Mw obtained by extrapolation of the experimental curve.

5. Calculation of the Burnout Ratio

The burnout ratio is calculated by dividing the maximum heat flux to be expected during normal operation into the heat flux required to produce film blanketing and subsequent burnout of the fuel plates. This important quantity is a measure of the margin which exists between normal operating conditions and those which could lead to melting of the fuel.

The heat flux to be expected under normal operating conditions is readily computed from the power distribution data previously discussed. The Labuntsov correlation[7] for subcooled forced-convection burnout for water-cooled channels was used to compute the burnout heat flux. This correlation has the form

$$\phi_{bo} = 4.61 \times 10^5 \, \theta(P) \times \left[1 + \frac{0.232V^2}{\theta(P)}\right]^{1/4} \left(1 + \frac{15C_p\Delta t_{sub}}{P^{1/2}\lambda}\right),$$

where

$\theta(P) = P^{1/3}(1 - P_r)^{4/3}$,

P = pressure (abs atm),

P_r = reduced pressure (P/P_c),

P_c = critical pressure (abs atm),

V = velocity (fps),

C_p = specific heat (Btu $\text{lb}^{-1}\,{}^\circ\text{F}^{-1}$),

Δt_{sub} = saturation temperature minus bulk water temperature (°F),

λ = heat of vaporization (Btu/lb).

The burnout heat flux was obtained for the reference core positions and for both 30- and 45-Mw operating conditions.

Hot-channel factors were obtained for both core positions by utilizing the following factors:

Factor	Position D-4	Position E-5
Error in flux determination	1.10	1.10
Variation of isotopic ratio	1.03	1.03
Error in ^{235}U content	1.04	1.08[a]
Geometric variation	1.05	1.05
Nonuniformity of flux within element	1.40	1.40
Hot-channel factor	1.73	1.80

[a]Core position E-5 contained a 240-g element which had been depleted to 200 g.

The power level corresponding to the burnout heat flux was obtained in each case by multiplying the ratio of the burnout heat flux to the maximum operating heat flux by the corresponding operating power level. An equivalent burnout power level was then arrived at by dividing this value by the hot-channel factor. The burnout ratio was obtained by dividing the normal operating power level into the equivalent burnout power level. The results, given in Table B.1 below, show the values for the case of full flow (17,300 gpm) and for operation just above the low-flow scram set point (14,000 gpm) for both 30- and 45-Mw operation.

It can be seen from these results that the burnout margin is quite large. It is easy to deduce that even at low flow and at a power level as high as 60 Mw the burnout margin would still be nearly 2.

6. Afterheat Removal

Tests conducted in 1958 revealed that detectable boiling would not occur in the ORR following a shutdown from operation at power levels up to 17 Mw, even if no forced convection cooling were

[6]W. R. Gambill and R. D. Bundy, *HFIR Heat Transfer Studies of Turbulent Water Flow in Thin Rectangular Channels*, ORNL-3079 (June 5, 1961).

[7]D. A. Labuntsov, *Soviet J. At. Energy* (English Transl.) 10, 516–18 (Nov. 1961).

Table B.1. Heat Flux and Burnout Ratio

Position	Power[a] (Mw)	Flow	Burnout Heat Flux (Btu ft^{-2} hr^{-1})	Maximum Heat Flux (Btu ft^{-2} hr^{-1})	Burnout Power (Mw)	Burnout Ratio
			$\times 10^6$	$\times 10^6$	1.44 @ 60	
E-5	30	Normal	4.54	0.55	138	4.60
D-4	30	Normal	4.99	0.72	120	4.01
~~E-5~~	~~30~~	~~Low~~	3.98	0.55	121	4.02
D-4	30	Low	4.42	0.72	107	3.55
E-5	45	Normal	4.48	0.82	136	3.04
D-4	45	Normal	5.02	1.07	122	2.71
E-5	45	Low	3.88	0.82	118	2.63
D-4	45	Low	4.44	1.07	108	2.40

[a]Note that at 30 Mw the inlet coolant temperature is 120°F and at 45 Mw it is 115°F.

supplied.[8] More recently, experiments by Gambill and Bundy[9,10] have permitted predictions of a natural-circulation burnout heat flux in the ORR of approximately 125,000 Btu ft^{-2} hr^{-1}. Immediately following a shutdown from 45-Mw operation, the maximum heat flux in the ORR core will not exceed 75,000 Btu ft^{-2} hr^{-1}, and this will decrease rapidly with time.

Despite the fact that the available shutdown heat flux is considerably lower than that required for burnout, the reactor is provided with positive afterheat protection in the form of dc motors coaxial with the main primary pump motors.[11] Any one of the pumps so powered is capable of circulating 1000 gpm through the primary coolant system.

The maximum temperature of the hottest fuel-plate surface has been calculated under the following assumptions: (1) the reactor is shut down by the low-flow trip following loss of ac power to the primary pumps; (2) the flow from those pumps coasts down according to the experimental curve shown in Fig. B.12; (3) the heat flux drops according to a Way-Wigner-type relation; and (4) only one dc-driven pump is in operation and it delivers not its rated capacity of 1000 gpm but only 500 gpm. The results, shown in Fig. B.13, show that no excessive temperatures are developed following shutdown from 45 Mw.

7. Conclusions

It is clear from the foregoing that although operation at 45 Mw reduces the margin of safety between the normal operating conditions and burnout

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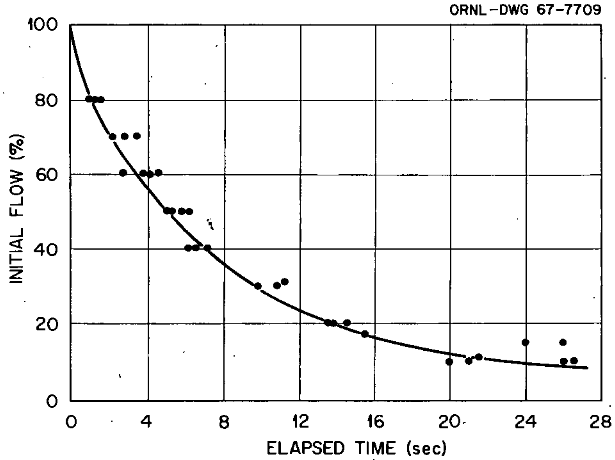

Fig. B.12. ORR Primary Coolant Pump Coastdown Curve.

[8]J. A. Cox *et al.*, *ORR Operations for Period April 1958 to April 1959*, ORNL-CF-59-8-39 (1960).

[9]W. R. Gambill and R. D. Bundy, private communication, Mar. 31, 1960.

[10]W. R. Gambill and R. D. Bundy, *Burnout Heat Fluxes for Low Pressure Water in Natural Circulation*, ORNL-3026 (1960).

[11]T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I), sect. 6.2.1 (to be published).

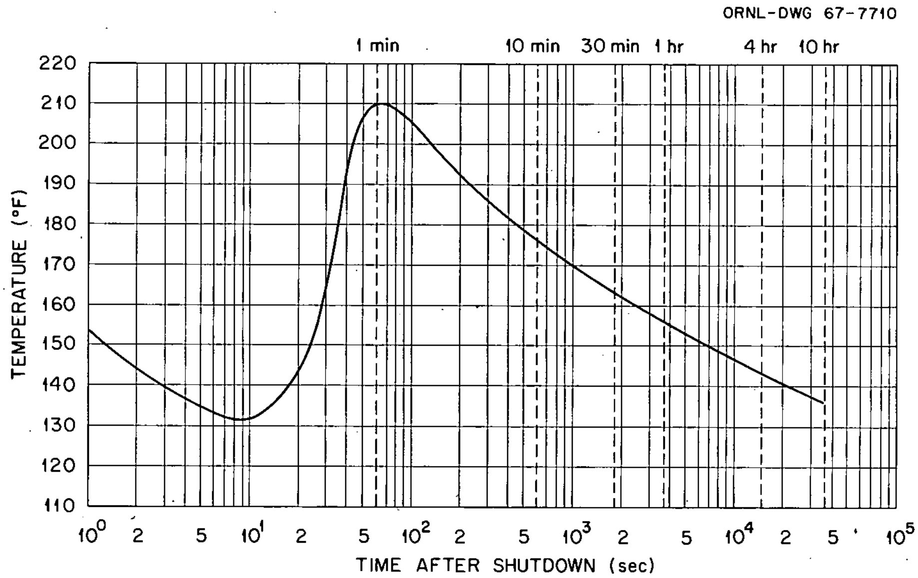

Fig. B.13. ORR Fuel Plate Surface Temperature vs Time with One Cooling Pump Operating on Auxiliary Power (One Pony Motor).

conditions there is still ample headroom. In fact, it appears, as was originally contended following the approach-to-power tests in 1958, that insofar as internal heat removal is concerned the ORR could be successfully and safely operated at power levels up to 60 Mw.

APPENDIX C – ANALOG ANALYSIS OF ORR WITH REFERENCE TO 45-Mw FULL-POWER OPERATION

R. S. Stone

1. Introduction

As a precautionary measure before raising the power of the ORR to 45 Mw, it was felt that an analysis of the "startup accident" should be done with the safety level trip set at 55 Mw.

A "startup accident" is defined as a situation in which the control rods are withdrawn from the reactor at their maximum speed and cannot be stopped. The period meter and/or the period trips do not work, so that the only protection left in the system comes from the safety level trips and such self-limiting effects as temperature coefficient, void coefficient, etc.

As part of the same program, it was suggested that an analysis of reactor response to the failure of the primary cooling circuit pumps would give valuable information.

These analyses were carried out on the ORNL Reactor Control Analog Facility.

2. ORR Simulator Block Diagram

The ORR[1] was simulated on the Analog Computer by the system shown in Fig. C.1.

The reactor network solves the reactor kinetic equations for power, with the various reactivity inputs summed by the reactivity network. The reactivity network is controlled by the control rod simulator and by the metal and water temperature

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[1]T. P. Hamrick and J. H. Swanks, *The Oak Ridge Research Reactor, A Functional Description*, ORNL-4169 (vol. I) (to be published).

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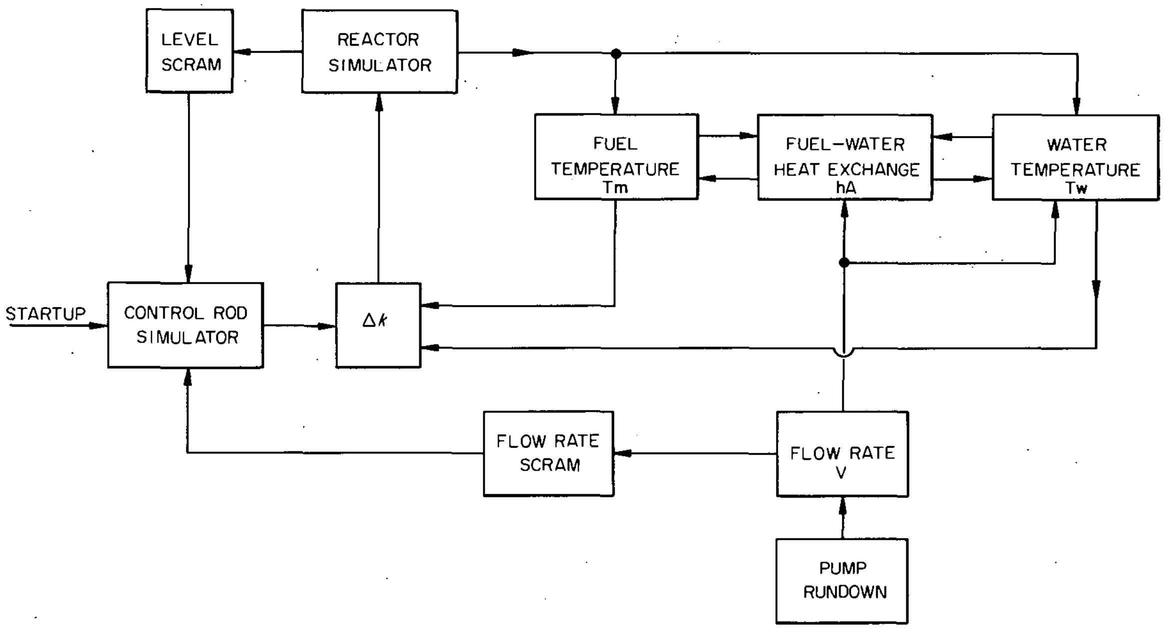

Fig. C.1. ORR Simulator Block Diagram.

coefficients. The control rod simulator generates rod position and from this introduces reactivity as a function of the S-shaped control rod calibration curve. In a scram situation this block simulates the delay due to the magnet release time and then generates the rod's accelerated motion.

3. Methods of Calculation

The reactor kinetics and heat-transfer equations were written for the reactor core assuming a flat temperature profile in the fuel and a uniform temperature in the bulk water. The heat-transfer coefficient across the metal-water film is given by the formula[2]

$$h = 170\,(1 + 0.010T_w - T_w^2 \times 10^{-5}) \times (V^{0.8}/D^{0.2})\ \text{Btu hr}^{-1}\ \text{ft}^{-2}\ {}^{\circ}\text{F}^{-1}\ .$$

This was approximated as

$$hA = (35.2 + 0.232T_w)V^{0.8}\ \text{Btu sec}^{-1}\ {}^{\circ}\text{F}^{-1}\ .$$

The boiling of the moderator was not simulated.

[2]After J. A. Lane; see S. Glasstone, *Principles of Nuclear Engineering*, Van Nostrand, p. 678.

4. ORR Data Used in Simulation[3]

Neutron generation time in the beryllium-reflected core

$$\ell^* = 6.4 \times 10^{-5}\ \text{sec}$$

Effective delayed-neutron fraction

$$\beta = 0.0080$$

Temperature effect[4]

$$-\Delta k = 0.18 \times 7.6 \times 10^{-5}\ T_m + 0.82 \times 7.6 \times 10^{-5}\ T_w$$

Fraction of power generated in fuel

90%

Fraction of power generated in moderator

4.9%

[3]T. P. Hamrick, private communication.

[4]Experimental net temperature coefficient of $7.6 \times 10^{-5}/^{\circ}$F was divided between metal and water proportionally to their relative thermal expansions (expulsion of moderator).

Heat capacity of fuel

0.21 Btu lb^{-1} $^{\circ}F^{-1}$ or 40.1 Btu/$^{\circ}$F for the core

Heat capacity of moderator

1 Btu lb^{-1} $^{\circ}F^{-1}$ or 108 Btu/$^{\circ}$F for the core

Period scram

It is assumed that the period safety is inoperative

Safety level trip

55 Mw for full coolant flow

1.35 Mw for low coolant flow, except as noted

Magnet release time

25 msec, except as noted

Control rod acceleration

0.6 *g*

Initial level for "shutdown" cases

1.1 w

Initial negative reactivity for "shutdown" cases

−1%

Rate of rod withdrawal used for startup accident

0.156%/sec for maximum value portion of rod

0.031%/sec for minimum value portion of rod

Negative reactivity vs time after start of scram

Shown in Fig. C.2

5. Startup Accident

Uncontrolled rod withdrawals were investigated under a variety of simulated conditions. Table C.1 lists these initial conditions and their effects upon reactor response. Runs with the reactor shut down were made at initial temperatures of both 70 and 95°F, but results were so nearly identical that only the 95° data are presented. In all runs, including those where scram did not occur, rod withdrawal was stopped when the safety trip point was reached. In those runs which used a scram, a 25-msec time delay was assumed between attainment of trip level and start of rod insertion.

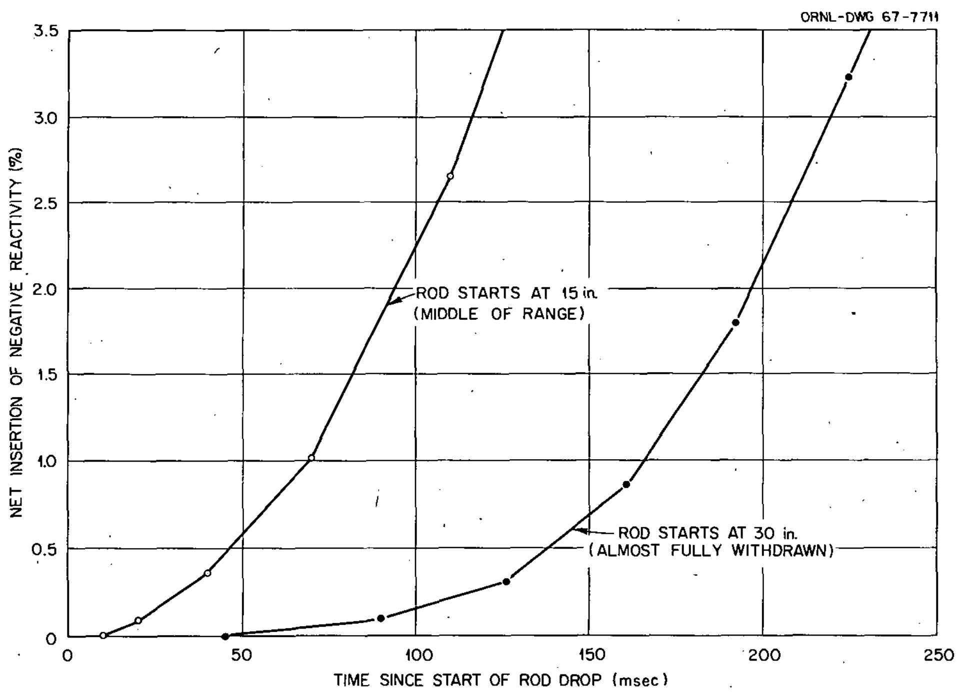

Fig. C.2. Time Response of Safety Rods as Used in Simulation.

Table C.1. Startup Accident Results for Various Initial Conditions

A 25-msec delay is assumed between trip point and start of rod drop.
Figures in parentheses are for runs with scram disconnected.

Run No.	Initial Conditions	Total Positive Reactivity Inserted (%)	Minimum Period Reached	Maximum Power Level (Mw)	Maximum Mean Fuel Temperature (°F)	Maximum Bulk Water Temperature (°F)
49 (48)	Shut down, no pumps, maximum rod worth, trip point = 1.35 Mw	0.85 (0.85)	84 msec (84 msec)	1.35 (17)	101 (350)[a]	97 (160)[a]
57	Shut down, no pumps, minimum rod worth, trip point = 1.35 Mw	0.32	0.975 sec	1.35	109	101
46 (47)	Shut down, full flow, maximum rod worth, trip point = 55 Mw	0.95 (0.95)	75 msec (75 msec)	57.5 (155)	205 (370)[b]	110 (139)[b]
45b	Shut down, full flow, minimum rod worth, trip point = 55 Mw	0.52	1 sec	56.0	210	110
43c (54b)	Full power, full flow, maximum rod worth, trip point = 55 Mw	0.16 (0.16)	4 sec (4 sec)	55.4 (70)	220 (250)	128 (135)
44	Full power, full flow, minimum rod worth, trip point = 55 Mw	0.18	27 sec	55.0	220	128
50	Shut down, no pumps, maximum rod worth, trip point = 55 Mw	3.87	84 msec	55.0		c
52	Shut down, no pumps, maximum rod worth, trip point = 30 Mw	0.90	84 msec	32.0	198	105

[a]These values were read 18.6 sec after start of rod withdrawal (7.4 sec after start of power excursion). Temperatures at that time were still rising but showed signs of leveling off. Power at that time was steady at about 8 Mw.

[b]These values were read 16.5 sec after start of rod withdrawal (5.2 sec after start of power excursion). Temperatures at that time were still rising and looked like they would get 50% higher before leveling off at about 155 Mw.

[c]Bulk water temperature reaches 270°F (boiling) 19 sec after start of withdrawal (7 sec after start of power transient).

Terms used in Table C.1 are defined as follows:

Shutdown: Reactor initially in equilibrium at 95°F and −1% ΔK.

No pumps: Flow limited to 1 fps through core. This is a reasonable value for convection cooling.

Full flow: 30 fps through the core.

Full power: Reactor initially in equilibrium at 45 Mw. Full flow is assumed with an inlet temperature of 115°F, mean fuel temperature of 201.7°F, and mean coolant temperature of 127.5°F.

Maximum rod worth: Rod withdrawal is assumed to take place with the rods half inserted, that is, at the location where they have their maximum worth per inch. This creates a faster moving accident but also makes the rods more effective when a scram begins.

Minimum rod worth: Rod withdrawal is assumed to take place with the rods almost fully withdrawn, that is, at the location where they have their minimum worth per inch, about 20% of the maximum value. This creates a slower moving excursion but also makes the rods less effective when a scram begins.

Trip point: This is the power level at which a scram is initiated. In the analog model, trip point at full flow is set at 55 Mw. In the ORR, when coolant circulation drops below 77% of full flow, the trip point is automatically reduced to 3 N_L. When full power is 45 Mw, this low-flow trip point becomes 1.35 Mw and is so used in the simulated "no pumps" runs. Other trip points of interest were used as shown.

5.1 Discussion of Results. – The data in Table C.1 show that a level safety system set to trip at 55 Mw for full flow and 1.35 Mw for low flow is capable of halting a startup accident with virtually no power overshoot and with minimal increases in fuel and coolant temperatures (Figs. C.3 to C.6). In order to check the necessity for reducing the trip point when the pumps are off, a no-pumps startup accident was run with the trip point at 55 Mw (as used at full flow). Results are shown in Fig. C.7 and summarized in Table C.1. The main point to be seen here is that the temperature coefficients turn the excursion well before the power level reaches the trip point, and the reactor runs at 10 Mw or more for 18 sec before the still-withdrawing rods work the power up to the 55-Mw trip point. With no coolant flow, the reactor cannot stand such power levels; boiling and burnout may possibly occur. (In an actual reactor incident, the coolant temperature interlocks would protect the core from a slow temperature rise of this kind. However, in determining the proper set point for the neutron level safeties, credit should not be taken for "non-safety" systems.) It should be noted that the continued withdrawal of rods is a worse feature of this run than the failure to scram (see no-scram runs). These results show that the 55-Mw trip

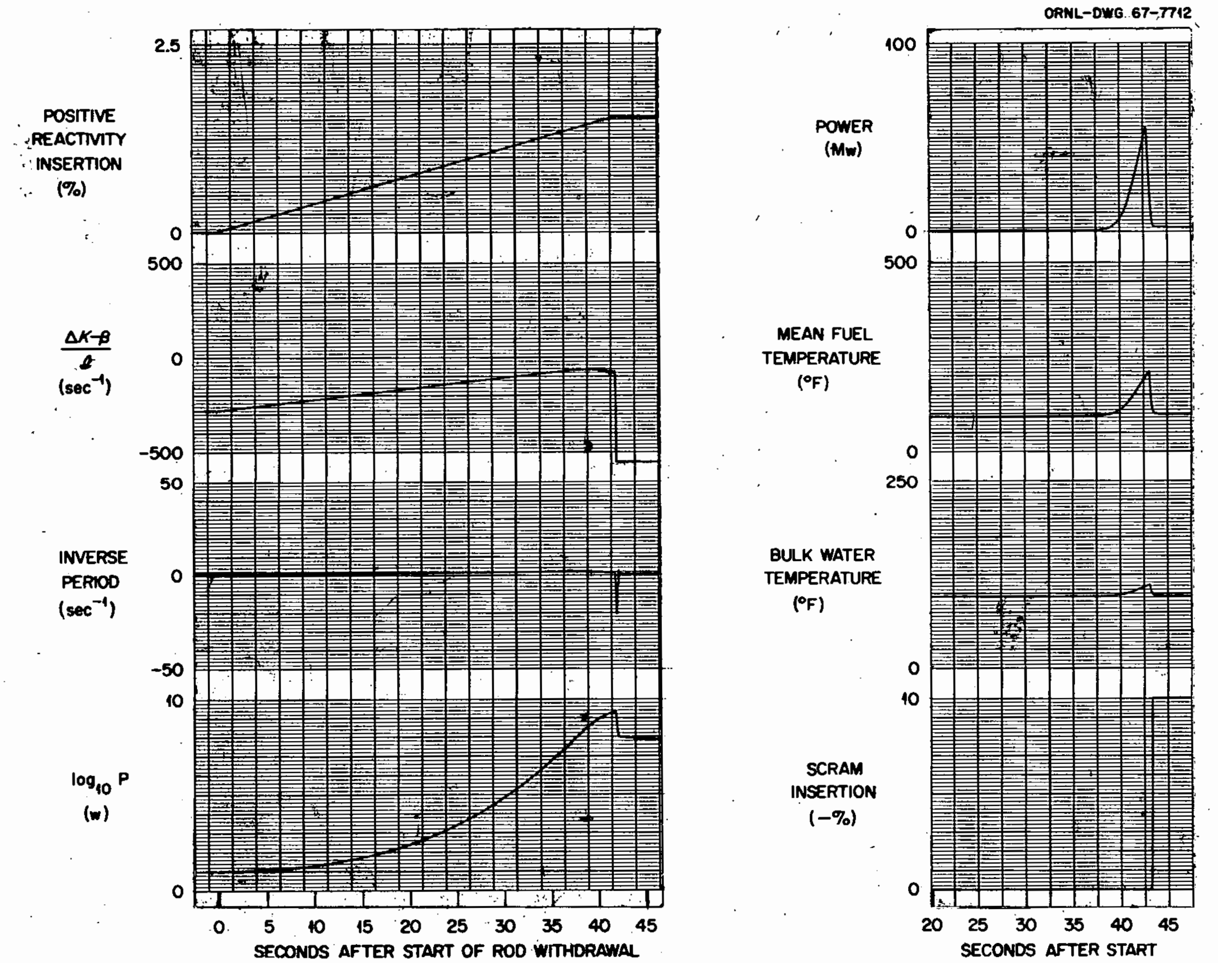

Fig. C.3. Startup Accident from Shutdown Condition (−1% ΔK). Full coolant flow; initial temperature, 95°F; rods at 30 in. (position of least worth/in.); trip point, 55 Mw. Run 45b.

point is perfectly adequate under full-flow conditions but should be, and is, reduced for critical runs with no flow.

Since for low flow the temperature coefficient turns the power transient at about 33 Mw (see Fig. C.7), another no-flow run was made with the trip point set at 30 Mw to determine the effect of scramming just before the temperature coefficient reduces the power level. The results are shown in Fig. C.8 and summarized in Table C.1. The outcome is a relatively minor excursion wherein the fuel temperature reaches 198°F and the bulk coolant temperature goes up 10°F. This indicates that, with respect to the startup accident, the 3 N_L (1.35-Mw) low-flow trip point setting is extremely conservative. The maximum allowable setting will depend not on the startup accident but on the maximum power the reactor can safely and continuously sustain without coolant flow. This is probably as high as 5 Mw.[5]

In those cases when rod withdrawal is cut off upon reaching trip point but when the rods do not drop (no scram), the fuel reaches temperatures such that the contacting coolant will boil (Figs. C.9 and C.10 and Table C.1), but it is not at all likely that damage will result to the core. Temperatures reached are moderate, even though the simulation takes no credit for the negative Δk introduced by steam void formation. The case starting at full power is particularly innocuous. Here only 0.16% Δk is inserted before trip point is reached, and the fuel temperature barely reaches the coolant boiling point.

[5]C. C. Webster, private communication (January 1967).

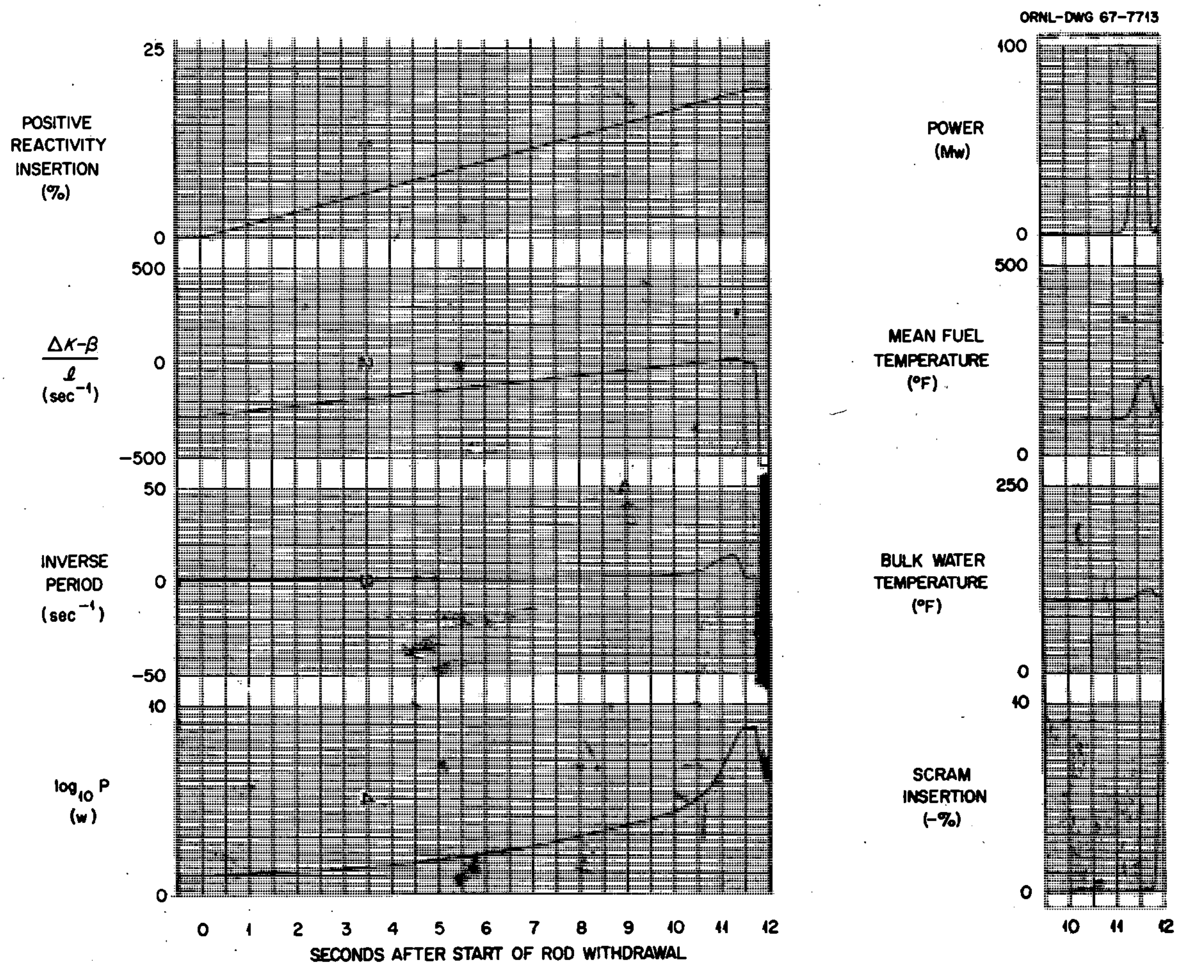

Fig. C.4. Startup Accident from Shutdown Condition (−1% ΔK). Full coolant flow; initial temperature, 95°F; rods at 15 in. (maximum worth/in.); trip point, 55 Mw. Run 46.

In light of the above, the startup accident does not constitute a significant challenge to the safety system; if the rods' removal is halted at trip point, nothing very drastic happens, even if no scram occurs. The one requirement is that the trip point be dropped somewhere below 30 Mw for low-flow operation.

6. Pump Rundown

Loss of main pumps with consequent coolant flow coastdown was simulated for the case where the pony motors are inoperative and the reactor bypass valve closed.[6] For the first minute following pump failure, the bypass valve and/or the pony motors have only a small effect upon coastdown of the coolant flow. Consequently, only the one coastdown case was simulated. Initial power was assumed to be 45 Mw, inlet temperature constant at 115°F, and all reactor temperatures and delayed-neutron precursors at equilibrium values.

[6]Values of flow vs time after pump failure were obtained from data taken by W. H. Tabor *et al.* on Nov. 17, 1966.

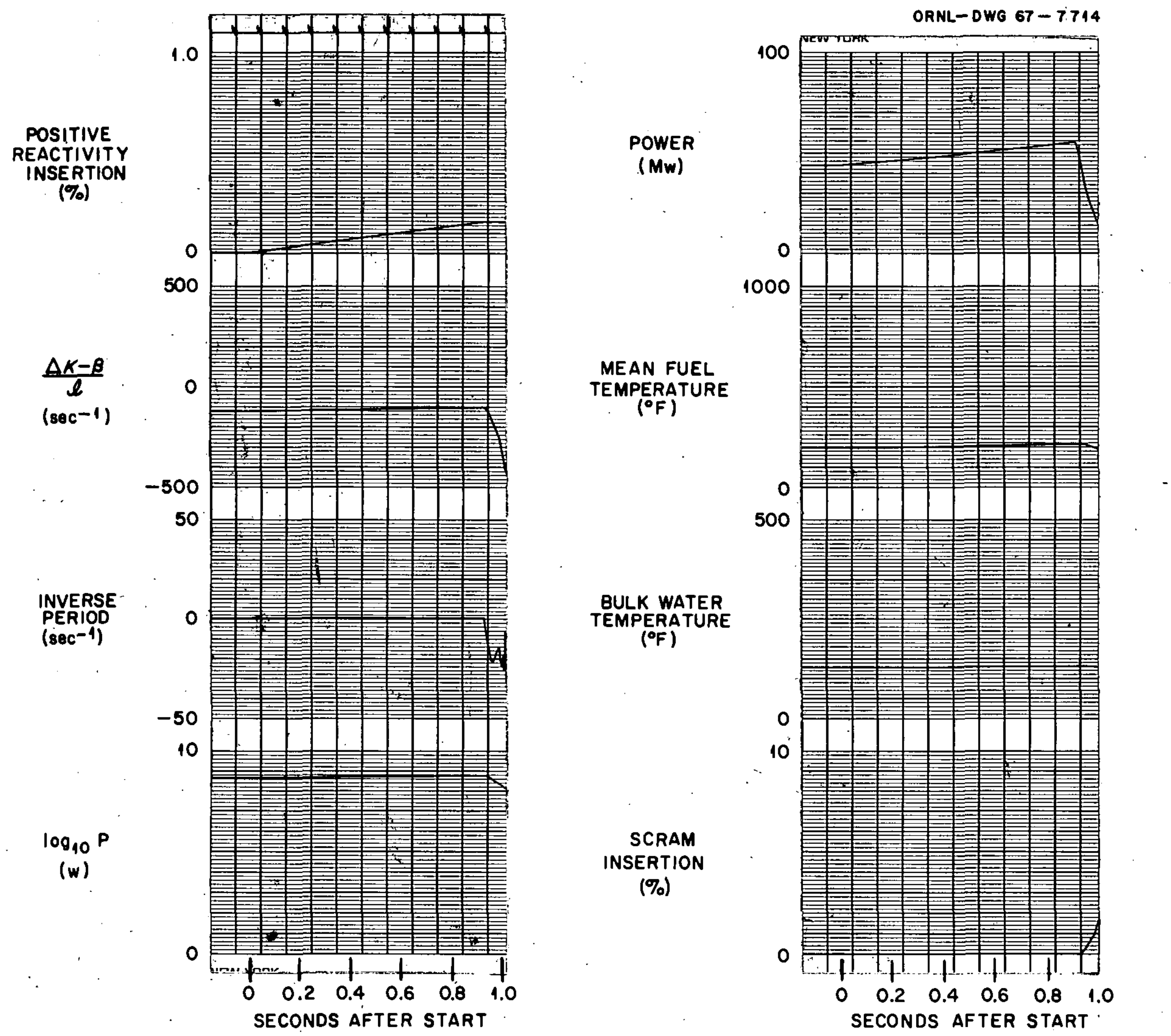

Fig. C.5. Startup Accident from Full Power (45 Mw). Full coolant flow; initial temperature, 115°F; rods at 15 in. (maximum worth/in.); trip point, 55 Mw. Run 43c.

In the run where low-flow scram was used, trip point was taken to be 80% of full flow, that is, 24 fps through the core. This point was reached about 1.5 sec after pump rundown began. After reaching trip point, a half-second delay was simulated before safety rod insertion began. Rods were assumed to be almost fully withdrawn, that is, in their least effective position, when the emergency began. In the scram-terminated run, the servo was assumed to oppose the temperature coefficient by pulling rods so as to hold the reactor at 45 Mw until rod drop actually began. This is an arbitrary and pessimistic situation which assumes the failure of operational interlocks. Under these extreme conditions, pump failure results in a transient rise of only 20°F in the mean fuel temperature and 5°F in the bulk water temperature. The whole transient was over and all temperatures were down to reactor inlet temperature within 3 sec after the start of pump rundown (see Fig. C.11*a*).

The same transient was run for the case where no scram occurs. Here the only action taken was to turn the servo off at the time when rod drop was supposed to begin. In this case, the temperature coefficient drops reactor power, so that 8 sec after rundown starts the mean fuel temperature has leveled off at 275°F, the mean coolant temperature is holding at 145°F, and power is down to 35 Mw and dropping proportionally with flow (see Fig. C.11*b*).

These results indicate that pump rundown at 45 Mw does not create a dangerous initial tempera-

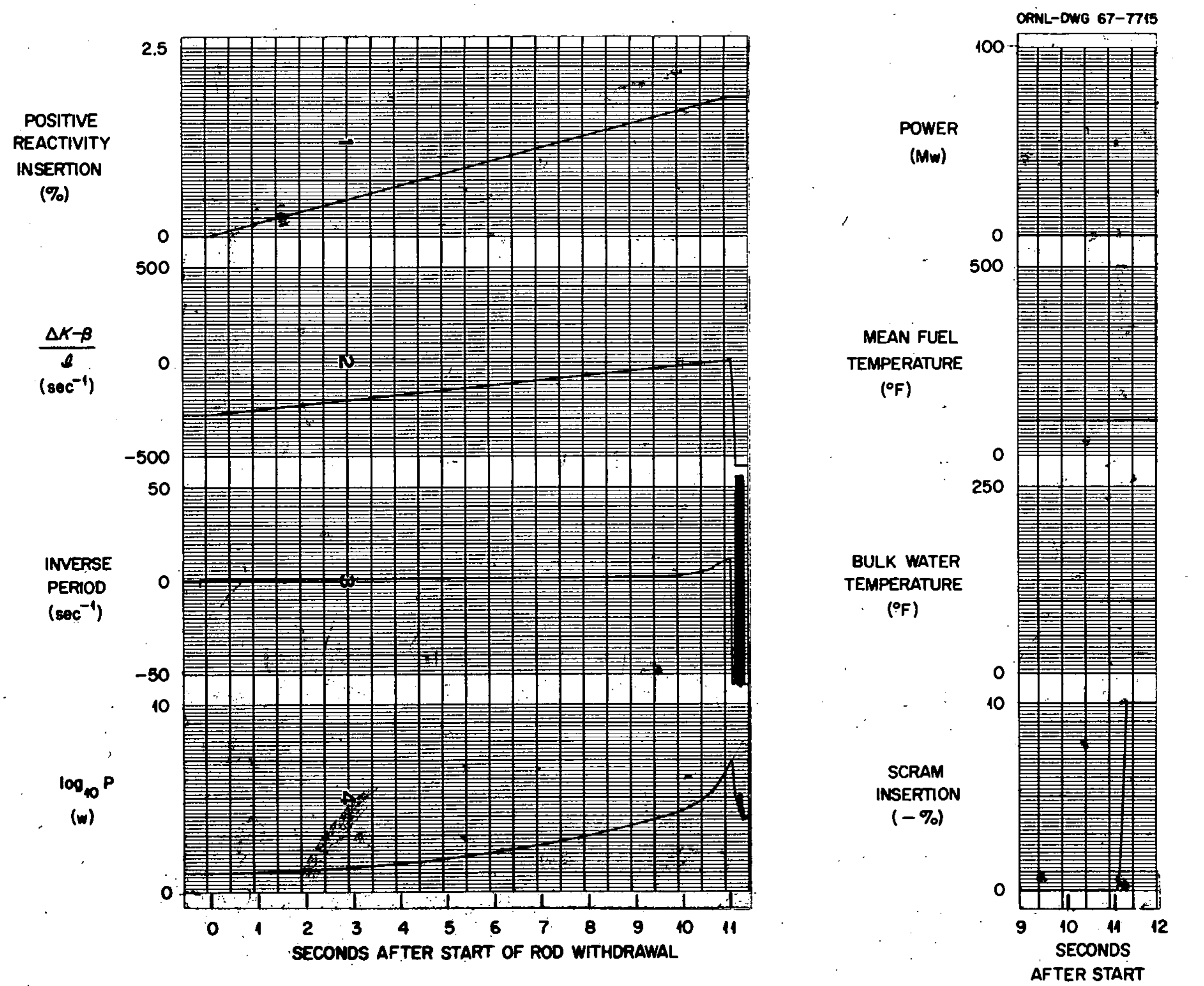

Fig. C.6. Startup Accident from Shutdown Condition (−1% ΔK). No coolant flow; initial temperature, 95°F; rods at 15 in. (maximum worth/in.); trip point, 1.35 Mw. Run 49.

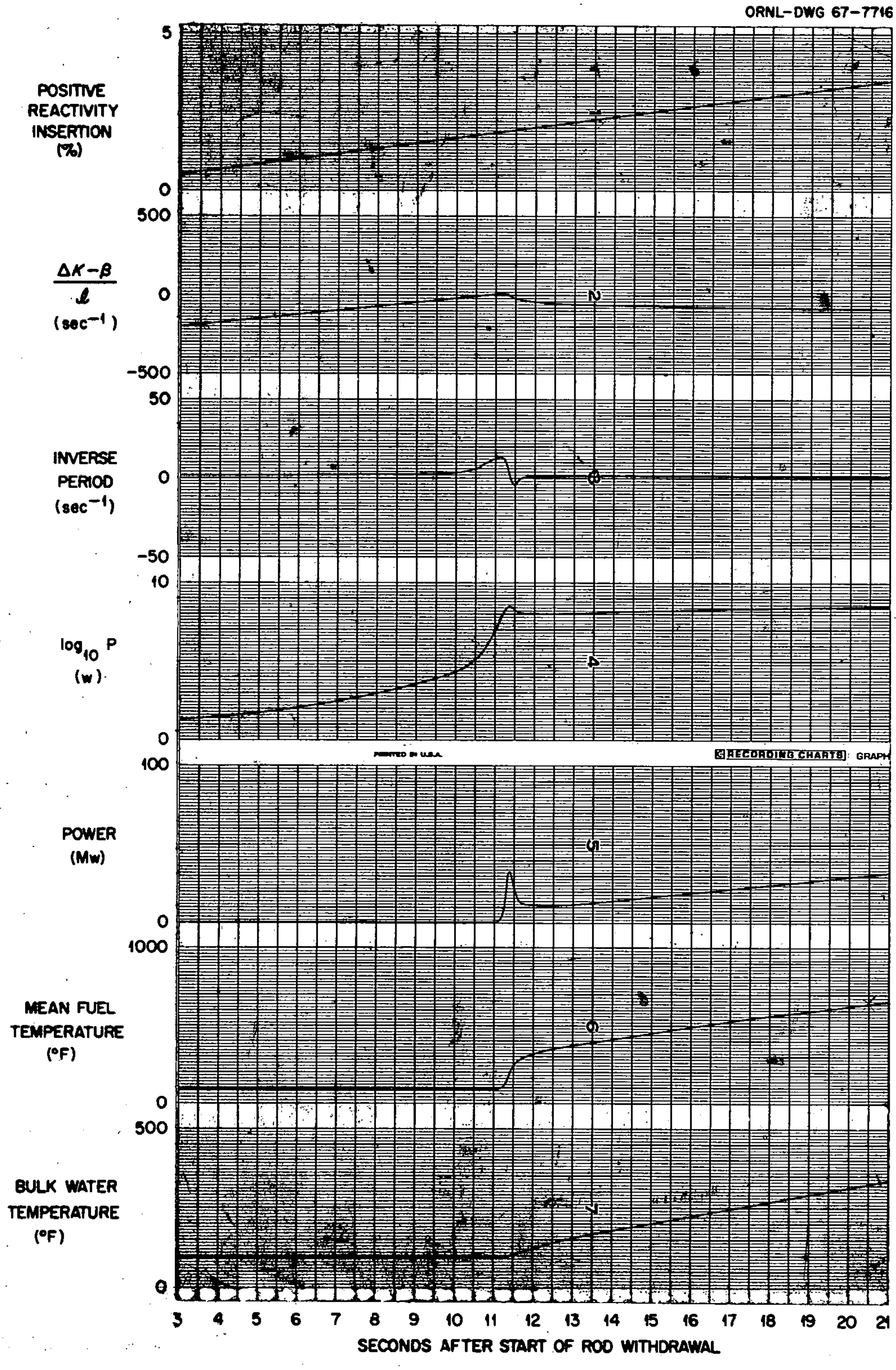

Fig. C.7. Startup Accident from Shutdown Condition (−1% ΔK). No coolant flow; initial temperature, 90°F; rods at 15 in. (maximum worth/in.); trip point, 55 Mw. Run 50.

ture rise, even where no scram occurs. Afterheat considerations require pony motor operation, and the rods obviously should be inserted, but such an occurrence constitutes no burden on the fast safety system.

7. Rapid Insertions of Positive Reactivity

A number of computer runs were made to investigate the response of the reactor model to rapid insertions of positive reactivity. These insertions were made both as steps and as fast ramps, with results as shown in Tables C.2 to C.4. The results are unrealistic for the larger reactivity insertions, since the model makes no provision for negative-reactivity shutdown through steam voiding (steam voids are a major contributor to self-shutdown from short periods, as shown by the SPERT experiments). The temperature at which boiling is initiated and the fraction of thermal energy which goes into steam production appear to be complex functions of a number of variables, including reactor period. For want of a steam-generation model which faithfully reproduces the SPERT results, steam effects have simply been omitted in this study of the ORR. The data shown in Tables C.2 through C.4 are the results of shutdown by temperature coefficient plus safety system only, and are useful only as a general guide to the type of transient which is potentially dangerous.

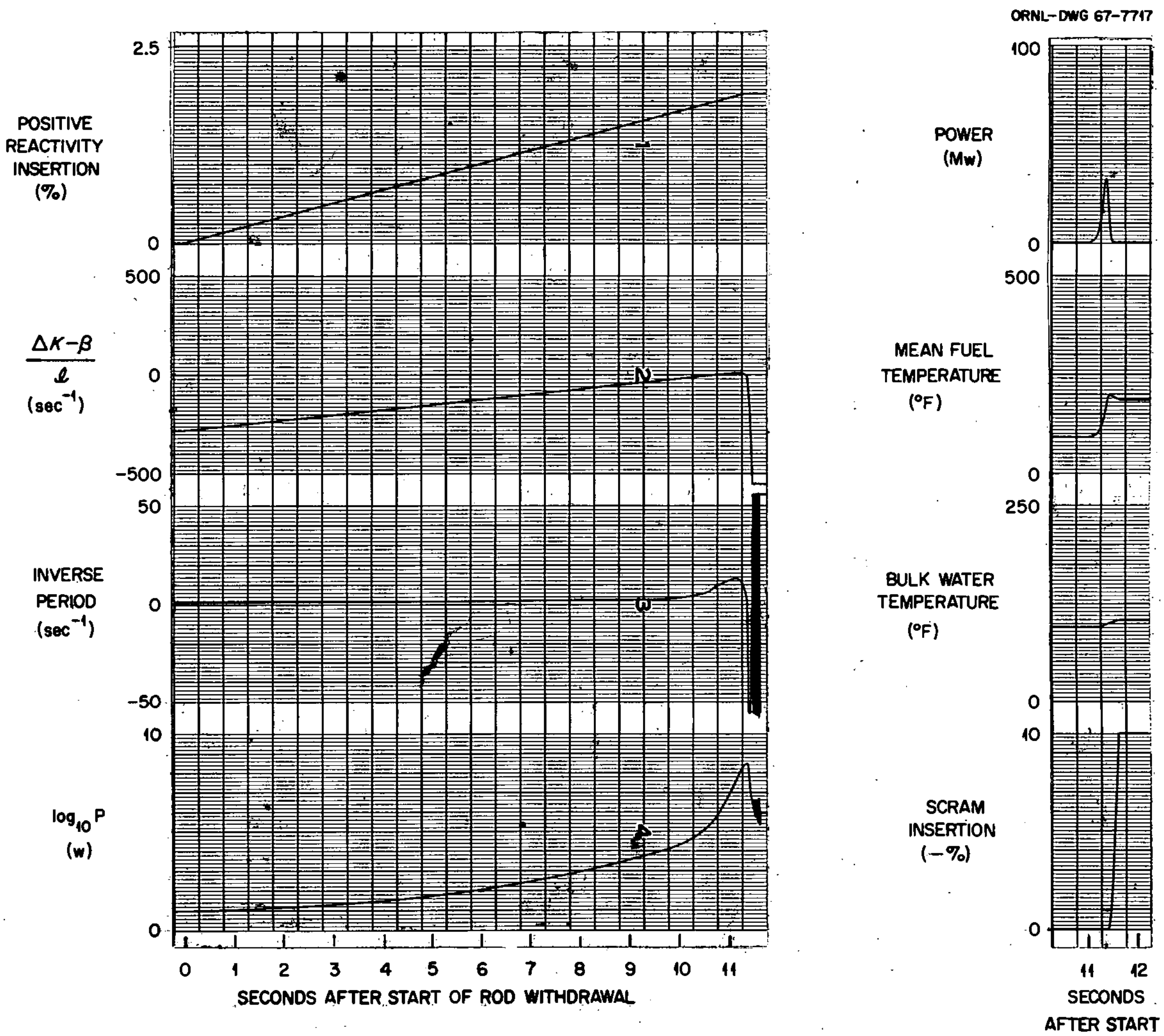

Fig. C.8. Startup Accident from Shutdown Condition (−1% ΔK). No coolant flow; initial temperature, 95°F; rods at 15 in. (maximum worth/in.); trip point, 30 Mw. Run 52.

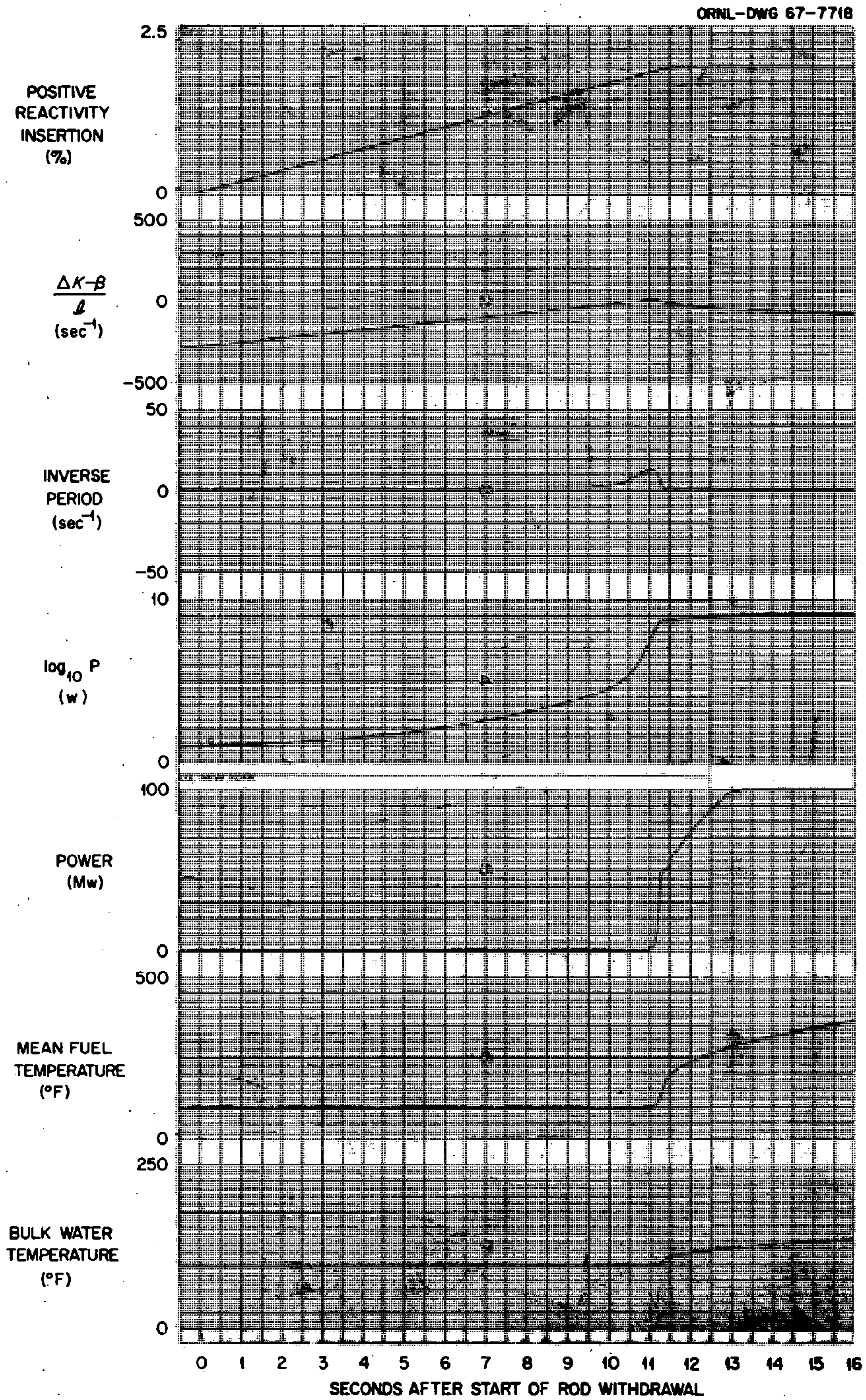

Fig. C.9. Startup Accident from Shutdown Condition (−1% ΔK). Full Coolant flow; initial temperature, 95°F; rods at 15 in. (maximum worth/in.); no scram. Run 47.

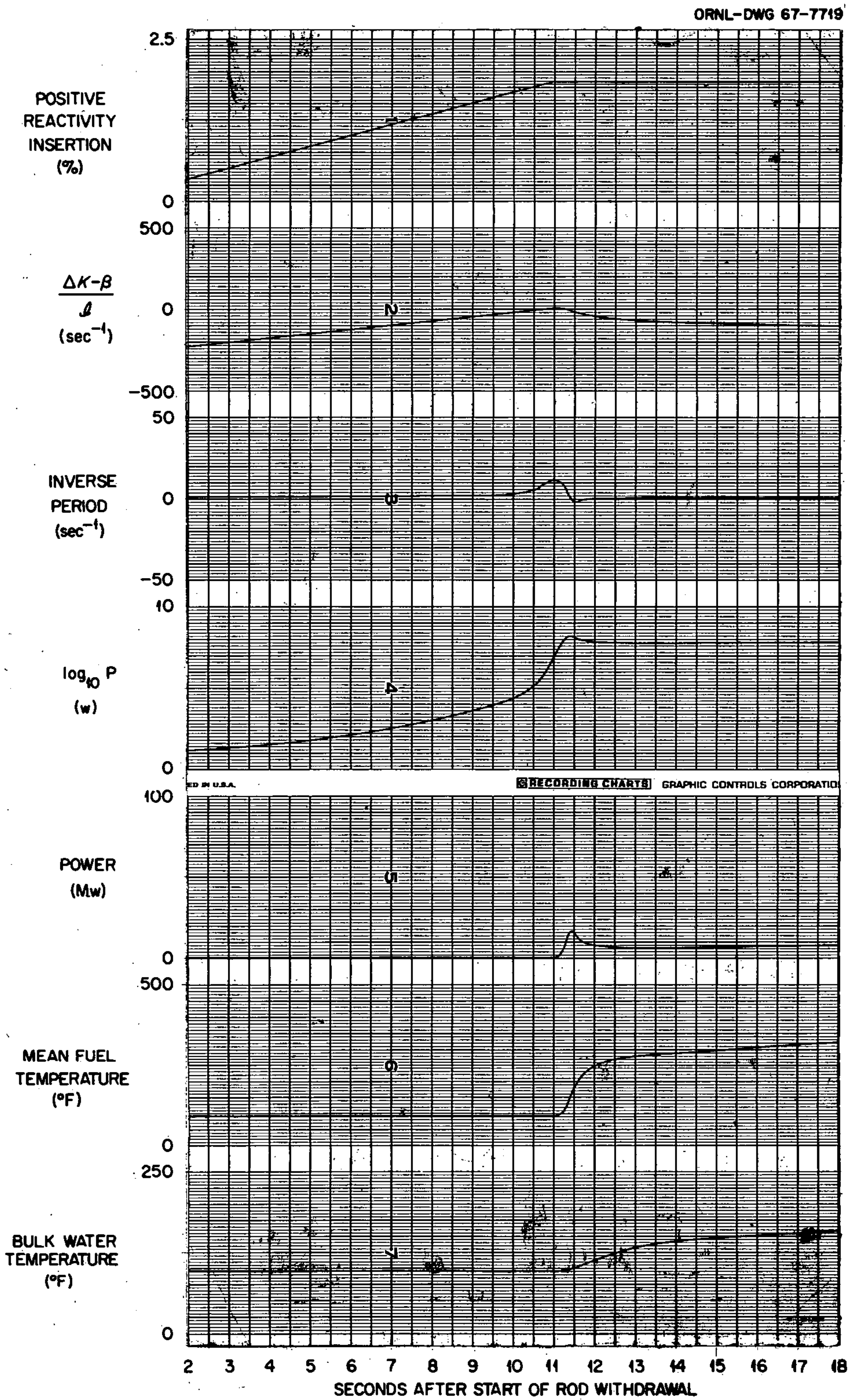

Fig. C.10. Startup Accident from Shutdown Condition (-1% ΔK). Full coolant flow; initial temperature, 95°F; rods at 15 in. (maximum worth/in.); no scram; rod withdrawal stops at 55 Mw. Run 56.

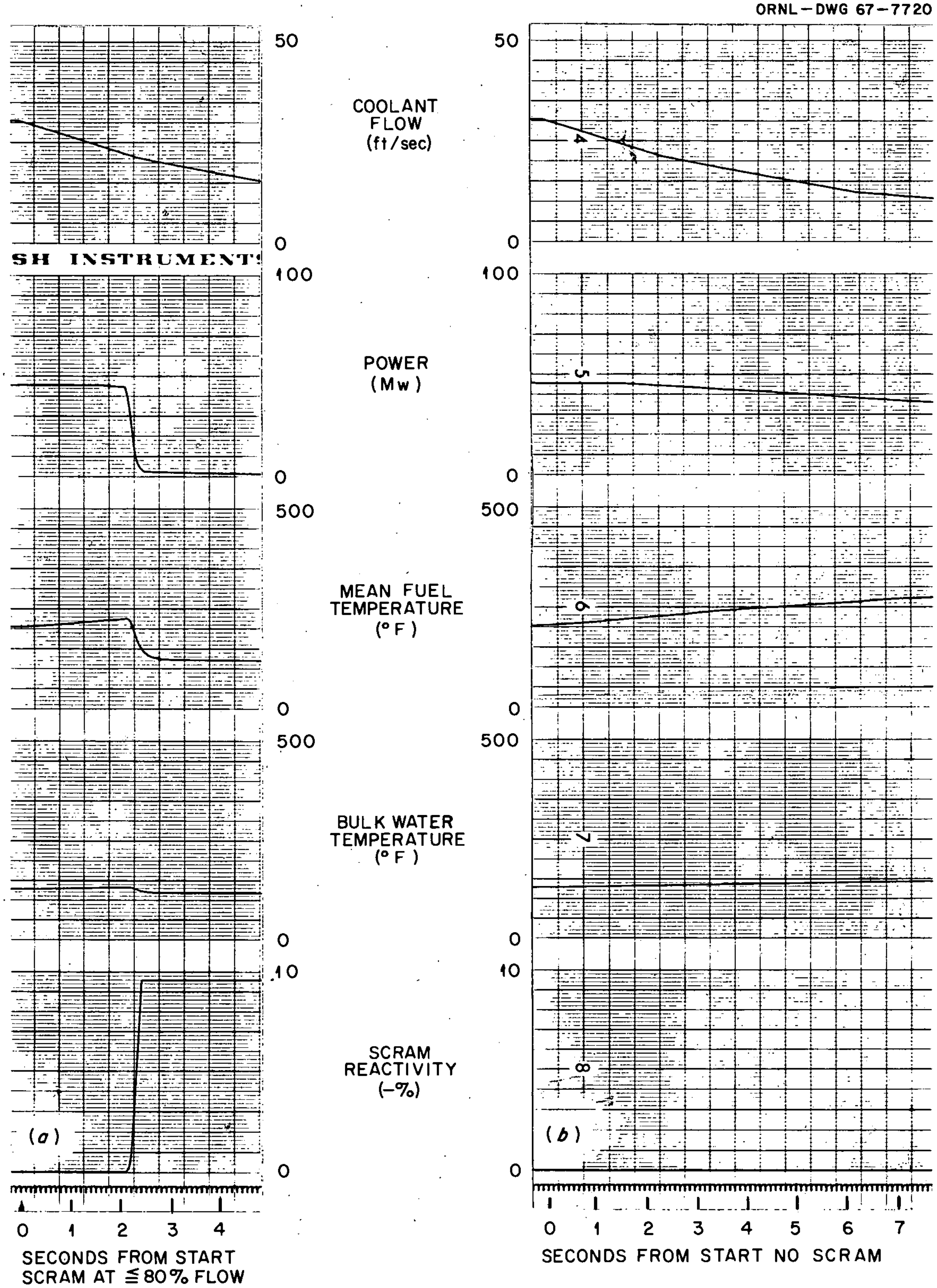

Fig. C.11. Pump Rundown at Full Power. Servo holds power at 45 Mw until flow drops to scram point.

Table C.2. Response of Shutdown Reactor to Rapid Reactivity Insertions

Reactor initially at –1%, 95°F, with no coolant flow; 1.35-Mw trip point

Run	Safety System	Reactivity Insertion	Minimum Period (msec)	Stable Period (msec)	Maximum Power (Mw)	Maximum Mean Fuel Temperature (°F)	Maximum Bulk Water Temperature (°F)
32	Maximum rod worth, 25-msec delay	Step to +1%	3	24	40	135	101
31	Minimum rod worth, 25-msec delay	Step to +1%	3	24	160	400	140
22	Maximum rod worth, 25-msec delay	Step to +1.25%	2.7	13	258	335	112
29	Minimum rod worth, 25-msec delay	Step to +1.25%	2.7	13	600	797	235
30	Minimum rod worth, 15-msec delay	Step to +1.25%	2.7	13	580	762	205
26	Maximum rod worth, 25-msec delay	Step to +1.40%	2.5	10	580	550	160
28	Minimum rod worth, 15-msec delay	Step to +1.40%	2.5	10	990	1000	270 (boiling) after 1 sec
23	Maximum rod worth, 25-msec delay	Step to +1.50%	2.4	8.9	1000	740	170
24	Maximum rod worth, 15-msec delay	Step to +1.50%	2.4	8.9	630	570	>140
25	Maximum rod worth, 40-msec delay	Step to +1.50%	2.4	8.9	1250	940	220

Table C.3. Response of Shutdown Reactor to Rapid Reactivity Insertions

Reactor initially at –1%, 95°F, with full coolant flow; 55-Mw trip point

Run	Safety System	Reactivity Insertion	Minimum Period (msec)	Stable Period (msec)	Maximum Power (Mw)	Maximum Mean Fuel Temperature (°F)	Maximum Bulk Water Temperature (°F)
7e	Maximum rod worth, 25-msec delay	Step to +1%	3	24	172	240	115
8c	Minimum rod worth, 25-msec delay	Step to +1%	3	22	225	350	130
11c	Maximum rod worth, 40-msec delay	Step to +1%	3	24	200	284	119
10	No scram	Step to +1%	3	20	480	760	270 (boiling) after 1.4 sec
15	Maximum rod worth, 25-msec delay	Step to +1.25%	2.7	13	585	470	142
12*b*	Maximum rod worth, 25-msec delay	Step to +1.5%	2.4	8.9	1330	760	185
13	Maximum rod worth, 15-msec delay	Step to +1.5%	2.4	8.9	1190	690	172
21	Maximum rod worth, 25-msec delay	3%/sec to +1.30% (Ramp)	12.2	Mean ≅ 18	670	505	147
20	Maximum rod worth, 25-msec delay	3%/sec to +1.5% (Ramp)	9.7	Mean ≅ 19	1280	740	182

Table C.4. Response of Full-Power Reactor to Rapid Reactivity Insertions

Reactor initially at 45 Mw, full flow, 115°F coolant inlet temperature; 55-Mw trip point, 25-msec delay

Run	Safety System	Reactivity Insertion	Minimum Period (msec)	Average Period (msec)	Maximum Power (Mw)	Maximum Mean Fuel Temperature (°F)	Maximum Bulk Water Temperature (°F)
37	Maximum rod worth	+1.1% step	5.3	16	560	495	170
38	Maximum rod worth	+1.2% step	4.8	15	750	580	185
39	Minimum rod worth	+1.2% step	4.8	15	750	605	193
36	Maximum rod worth	+1.5% step	3.9	12	1600	865	233
33	Maximum rod worth	+2% step	3.0	10	2200	1065	270 (boiling) after 60 msec
41	Maximum rod worth	25%/sec till scram. Gets to 0.95%	18	36	280	340	150
42	Minimum rod worth	25%/sec till scram. Gets to 0.95%	18	36	340	420	165

For an upper limit to the reactivity insertion a reactor can absorb without damage, SPERT results provide more realistic information.

Taking the results shown in Tables C.2 and C.3 at face value, positive reactivity steps in a shutdown reactor show a threshold for damage at about 2% if the rods are in the middle (most effective) portion of their range when the transient begins. This threshold drops to a value of about 1.4% if the rods must come in from near the top of the reactor (see Figs. C.12 and C.13). Most runs included a delay such that the safety rods did not start to drop until 25 msec after the reactor reached trip point. This delay is the upper limit permitted for ORR release times. Actual operational release times are 12 to 15 msec, and excursions simulated with 15 msec delay permitted a few tenths percent more reactivity to be added without damage. The effect of this 10-msec reduction in delay was much less noticeable when the rod started from the 30-in. (nearly withdrawn) position. This was because a drop from that point starts with a "built-in" delay of 45 msec before the rod can be appreciably felt. Cases run with no flow and a trip point of 1.35 Mw and those run with full flow and a trip point of 55 Mw both showed about the same allowable step reactivity insertion. Powers and temperatures reached were somewhat higher in the full-flow case (see Figs. C.12 and C.14).

With the reactor initially shut down, the reactivity damage threshold was the same for a moderately fast ramp (3%/sec) as for a step of the same total amount, since the ramped reactivity was all inserted by the time the neutron level got to the power range (see Table C.3). This behavior did not hold true for reactivity insertions made with the reactor initially at full power. With the reactor at power, a +2% step was about the most that could be tolerated for any initial position of the safety rods. However, a ramped insertion called temperature coefficients into play immediately, so that a 25%/sec ramp of any potential amount was easily tolerated, with shutdown accomplished before +1% reactivity had been reached (see Table C.4 and Figs. C.15 and C.16). Since accidental reactivity insertions are always ramps of one speed or another, these results indicate that the reactor at power has a great deal of self-protection against such additions. Even in the case of the shutdown reactor, the ramp nature of reactivity additions gives the period safety, temperature safeties, and boiling void coefficients time to come into play. For very fast reactivity additions to the shutdown reactor, SPERT data provide the best guidelines available.

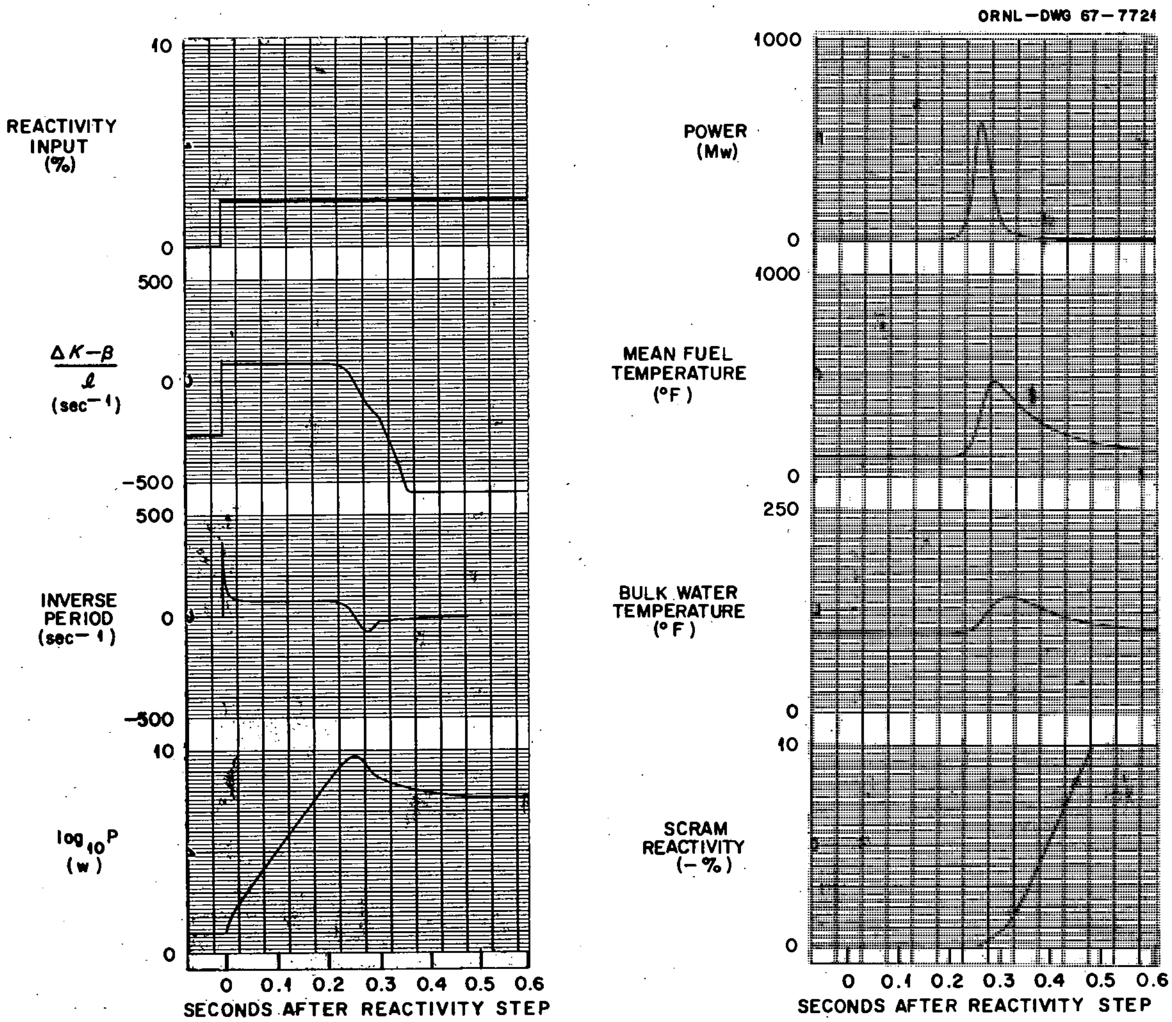

Fig. C.12. 2.25% Step Insertion of Reactivity (from −1% to + 1.25%). Reactor initially shut down at 95°F with no coolant flow. Rods go in from their most effective position (midpoint of core), with 25-msec delay between trip point (1.35 Mw) and start of drop. Run 22.

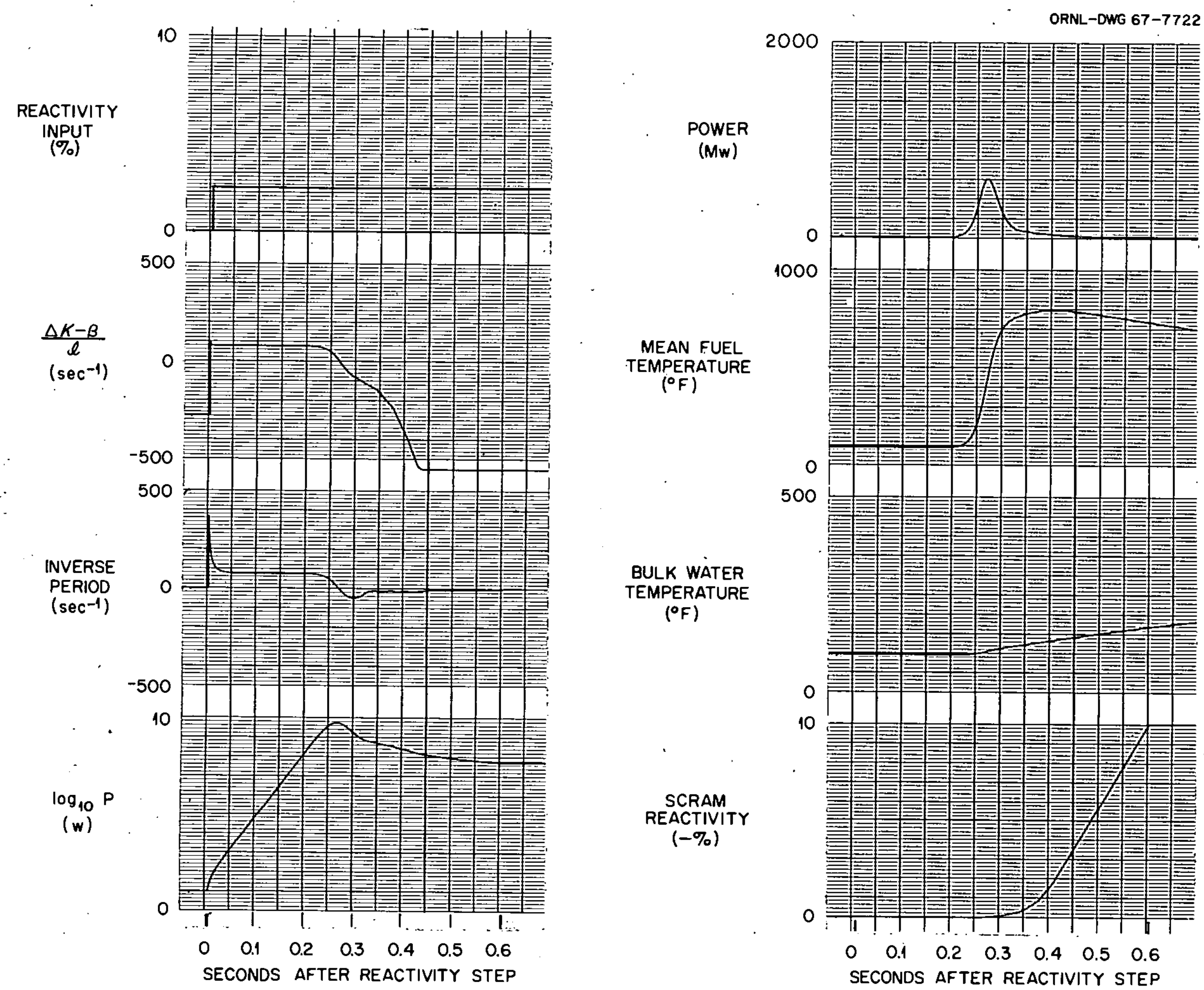

Fig. C.13. 2.25% Step Insertion of Reactivity (from −1% to + 1.25%). Reactor initially shut down at 95°F with full coolant flow. Rods go in from their least effective position (almost fully withdrawn), with 25-msec delay between trip point (55 Mw) and start of drop. Run 29.

ORNL-DWG 67-7723

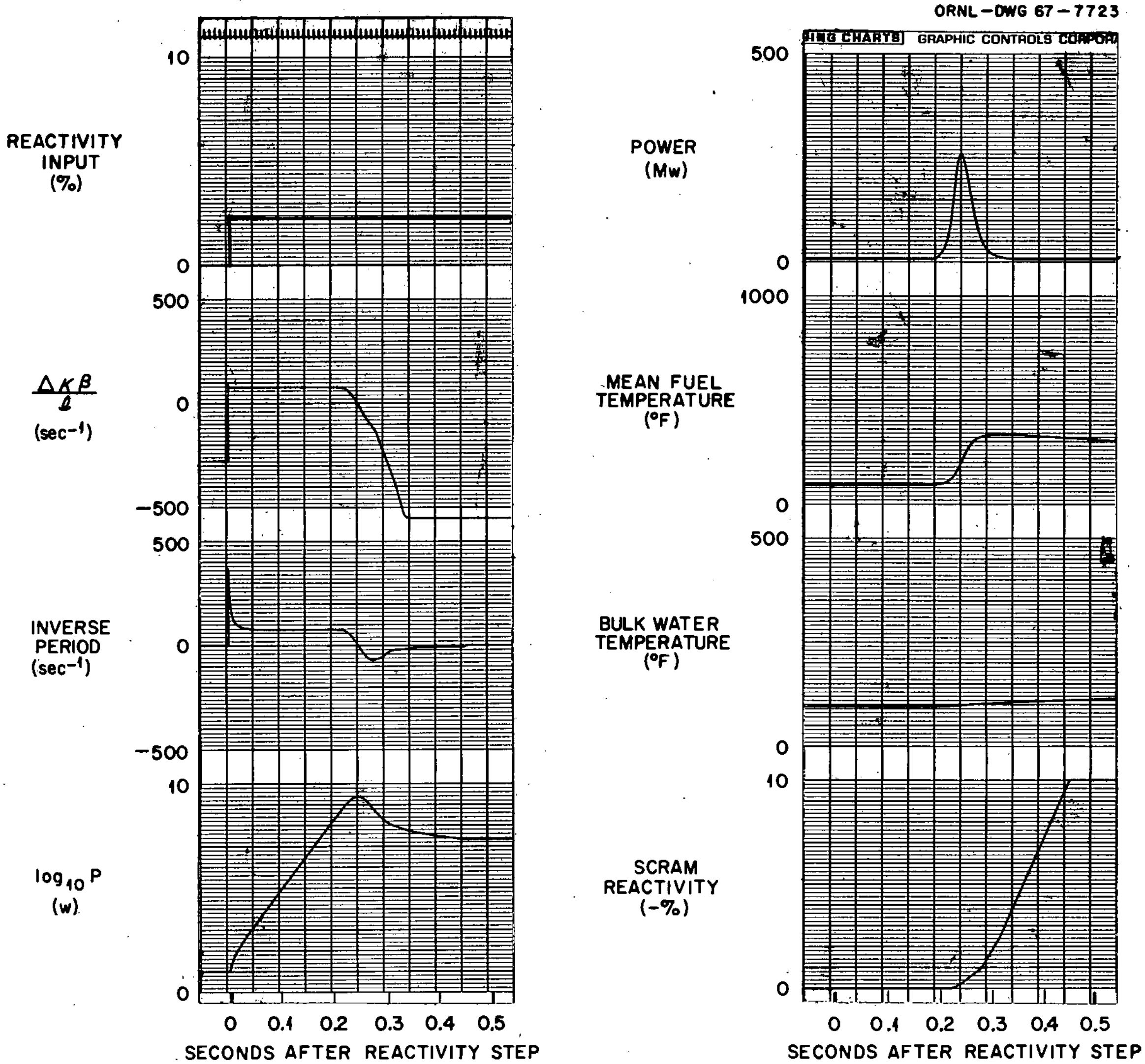

Fig. C.14. 2.25% Step Insertion of Reactivity (from −1% to +1.25%). Reactor initially shut down at 95°F with full coolant flow. Rods go in from their most effective position (midpoint of core), with 25-msec delay between trip point (55 Mw) and start of drop. Run 15.

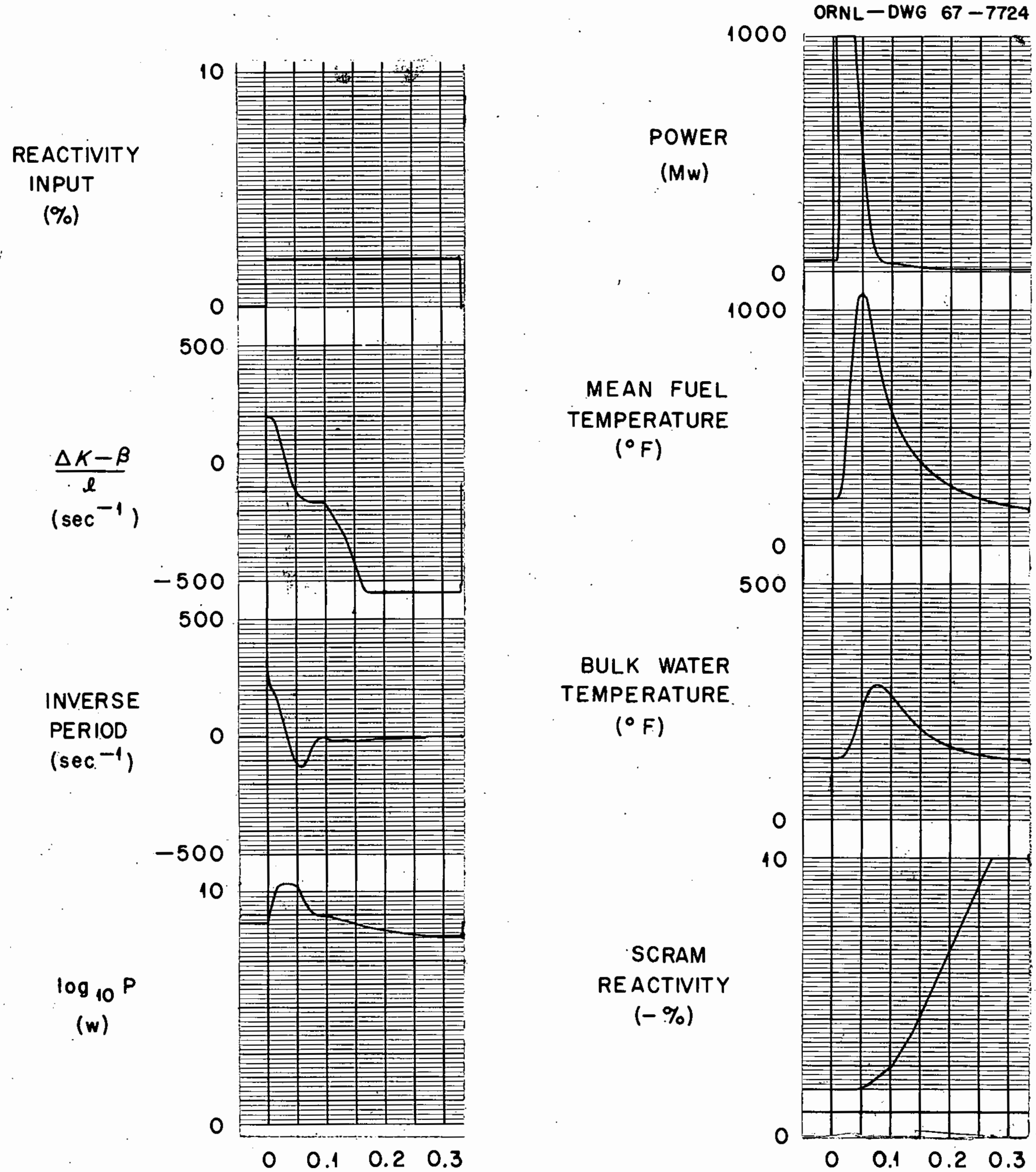

Fig. C.15. 2.0% Insertion of Reactivity (from 0 to +2.0%). Reactor initially at 45 Mw with full coolant flow and 115°F inlet temperature. Rods go in from their most effective position (15 in. withdrawal), with 25-msec delay between trip point (55 Mw) and start of drop. Run 33.

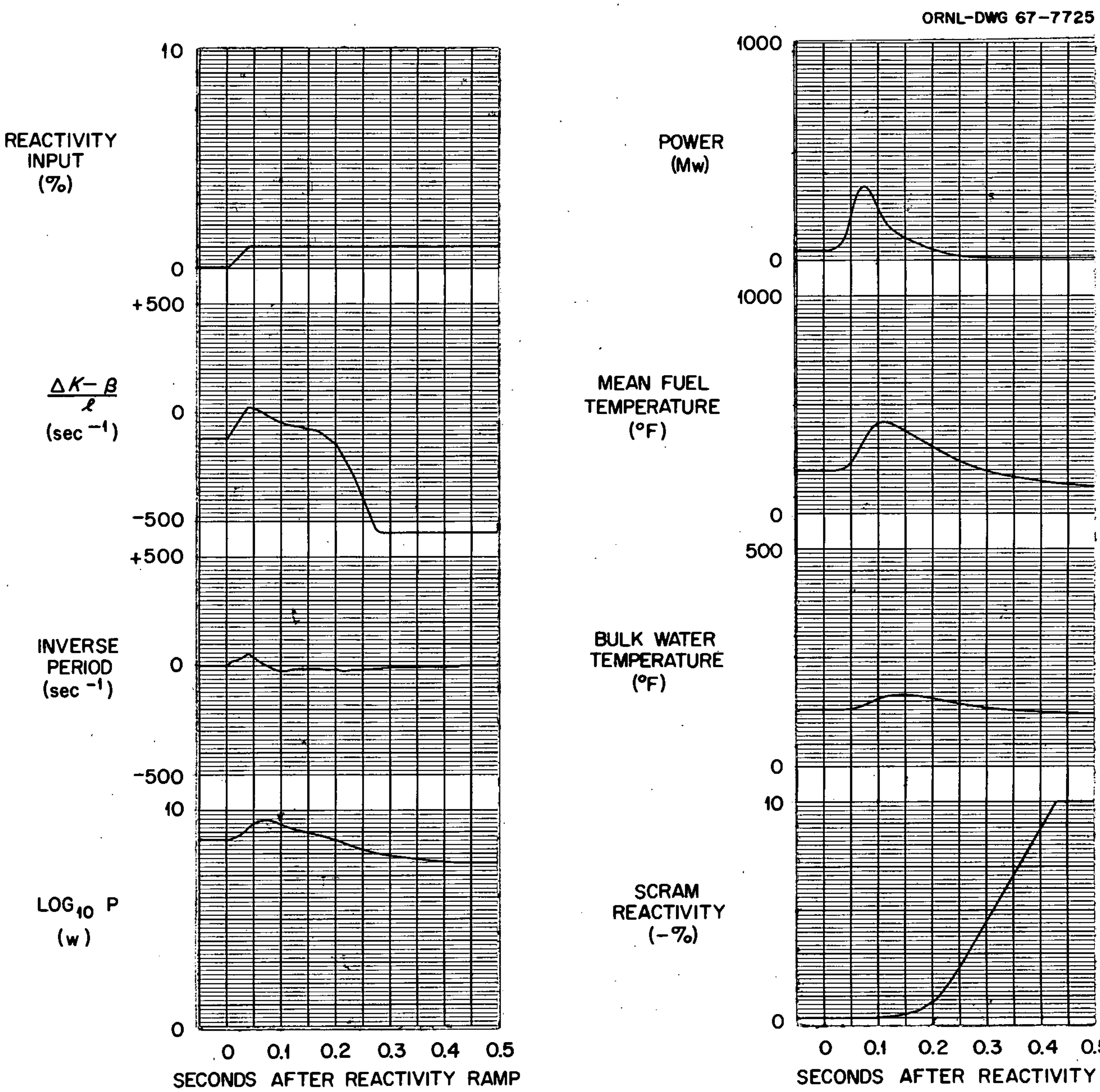

Fig. C.16. 2.5%/sec Ramp Insertion of Reactivity. Reactor initially at 45 Mw with full coolant flow and 115°F inlet temperature. Rods go in from their least effective position (almost fully withdrawn), with 25-msec delay between trip point (55 Mw) and start of drop. Run 42.

APPENDIX D – CHANGES REQUIRED TO PERMIT OPERATION AT 45 Mw

Only minor changes would be required in order to permit safe and orderly operation of the ORR at 45 Mw. Although not currently under active consideration, these changes are listed below.

1. External Heat-Removal System. It is planned to add one or more shell-and-tube heat exchangers and a third cooling tower cell in order to provide sufficient capacity to remove 45 Mw of heat. The temperature of the inlet cooling water will be decreased by approximately 5°F in order to maintain the exit temperature at its present value, 130°F. This is to maintain the magnitude of the thermal cycling of the dished heads on the engineering facilities at or below the present level. Since no primary flow increase is planned, the present piping is quite adequate. It is possible that the addition of the extra heat exchangers in parallel with the presently installed exchangers will reduce the system pressure drop sufficiently to produce a small increase in flow. This additional advantage is not required and will not be considered in the safety analysis.

2. Pool Cooling System. The present pool cooling system consists of a shell-and-tube heat exchanger which has a rated capacity of 1.5 Mw. It is possible that this will prove inadequate to maintain the pool temperature at the desired value when the reactor is operating at 45 Mw. If design calculations show this to be true, additional heat-exchanger capacity will be provided.

3. At present the transport of ^{16}N to the surface of the pool by thermal convection along the reactor vessel walls is prevented by downward-directed jets of water near the exterior vertical surfaces of the vessel. If required, these jets will be modified to provide adequate protection from ^{16}N convection.

4. Although both measurements and calculations indicate that the slight increase in gamma heating which will be experienced in the various structural components is insufficient to cause any concern (see Appendix E), temperatures in the pool walls and the beam tube will initially be kept under surveillance. Should any of these increase significantly, the situation will be corrected by providing thermal shielding in the case of the pool walls or additional turbulence in the case of the beam tubes.

5. The biological shield is in general adequate for 45-Mw operation; however, in some locations, such as the pump house and the demineralizer cells, additional local shielding may be required. This will be installed as required following initial tests at 45 Mw.

6. The iodine absorbers located in the building ventilation system are to be improved in accordance with the technology developed in connection with the HFIR containment studies. This includes the installation of thicker beds containing charcoal of a type adequate to remove organic iodine from the effluent stream. In addition, certain off-gas lines, notably that from the equalizer-leg degasifier, which are not at present provided with high-efficiency iodine filters, will be modified to improve their iodine-removal capacity.

7. Appropriate changes will be made to the control and safety system instrumentation set points in order to reflect the change in operating conditions. The period trips will remain unchanged; however, the level safety trips will be set at a lower percentage of "full power" than the present value of 150%. The reactor ΔT will increase from 11 to 16.5°F, so that the scram setting will be increased from the present value of 15.5 to 20.6°F. Since the reactor outlet temperature will be the same at 45 Mw as it is now at 30 Mw, the outlet temperature scram setting will remain at 140°F. Likewise, there is no necessity to increase the low-flow scram from the present value of 14,000 gpm.

APPENDIX E – REACTOR STRUCTURE TEMPERATURES

Tables E.1 and E.2 are tabulations of actual temperatures measured in various locations in and near the ORR during a test conducted on November 27 and 28, 1959. The purpose of this test was to determine the actual extent of gamma heating in the reactor structure and in the pool walls.

The locations of the thermocouples used to measure these temperatures are given in Figs. E.1 and E.2. As can be seen from the data, no high temperatures were observed, even in positions PSF-1 and PSF-3, which were expected to be quite high.

Table E.1. Reactor Structure Temperatures

Location	Temperature (°F) at —									
	November 27					November 28				
	20.0 Mw	5:37 PM 22.5 Mw	8:00 PM 25.0 Mw	9:00 PM 27.5 Mw	11:00 PM 30.0 Mw	3:00 AM 30.0 Mw	7:00 AM 30.0 Mw	11:00 AM 30.0 Mw	2:50 PM 30.0 Mw	6:50 PM 30.0 Mw
T-1	97	95	95	96	96	99	96	95	96	97
T-2	103	103	103	103	104	105	105	103	104	104
T-3	114	113	115	117	115	116	116	120	116	116
T-4	112	112	112	112	113	112	112	113	113	113
ITS-1	125	124	125	125	125	125	125	125	125	125
PSF-1	124	124	126	127	130	128	128	129	128	130
PSF-3	122	123	126	127	127	127	128	127	127	127
OTS-1	113	113	113	114	114	114	114	114	114	114
OTS-2	115	116	116	117	117	117	117	115	116	117
OTS-3	114	115	115	115	115	116	115	115	115	115
Top BH-3	100	100	100	100	101	102	100	100	100	100
NFTS-3	100	100	100	100	101	102	100	100	101	101

Table E.2. Shield Concrete Temperatures

Location	Temperature (°C) at –									
	November 27					November 28				
	20 Mw	5:10 PM 22.5 Mw	7:50 PM 25 Mw	9:10 PM 27.5 Mw	10:45 PM 30 Mw	3:00 AM 30 Mw	7:00 AM 30 Mw	11:00 AM 30 Mw	3:00 PM 30 Mw	7:00 PM 30 Mw
A-4	42.5	43.5	44.5	44.5	44.5	46.2	45.0	46.0	46.1	47.2
A-5	43.5	43.0	44.0	44.0	44.5	46.5	43.5	47.0	46.8	46.8
A-6	42.5	42.0	44.0	44.0	44.5	45.5	43.5	46.0	46.0	46.2
A-7	41.0	41.0	42.5	43.0	43.5	44.3	43.4	44.5	44.5	44.7
A-8	40.0	40.0	41.0	41.5	41.5	42.0	42.2	43.0	42.8	42.8
A-9	42.5	43.5	44.5	44.5	45.0	46.2	45.0	46.0	45.9	47.2
A-10	41.0	41.0	42.0	42.5	42.5	43.3	43.3	43.5	43.8	43.8
A-11	41.0	41.0	43.0	43.5	43.5	43.7	43.7	44.0	44.3	44.2
A-12	40.0	40.0	41.0	41.5	41.5	42.6	42.6	43.8	41.0	42.7
A-14	41.5	42.5	43.5	44.0	44.0	44.2	44.0	44.5	45.1	45.1
A-15	41.5	42.5	43.5	44.0	44.0	44.2	44.3	45.0	44.0	44.7
A-16	42.0	43.5	44.0	44.0	44.5	45.0	45.0	45.0	45.9	45.8
A-18	43.0	42.5	44.0	44.5	44.5	46.0	45.8	46.0	46.1	46.2
B-2	42.0	42.5	44.0	44.0	44.0	44.3	47.2	47.5	47.5	47.7
B-3	42.0	42.0	43.5	43.5	43.5	44.3	45.0	45.5	45.2	45.3
B-4	40.5	40.5	41.5	42.0	42.5	43.0	43.0	43.0	43.4	43.2
B-5	39.0	39.0	40.5	40.5	41.0	41.5	41.5	42.0	41.9	41.7
B-6	38.5	38.5	39.5	39.5	40.0	40.5	40.3	40.5	40.9	40.6
B-7	43.5	44.5	46.0	46.0	46.0	46.5	46.5	46.5	47.4	47.3
B-8	43.5	44.5	46.0	46.0	46.0	46.5	46.5	46.5	46.3	47.4
B-9	41.0	42.0	43.0	43.5	43.5	43.7	43.5	44.5	44.1	44.2
B-10	41.0	41.0	42.0	42.5	43.0	44.0	43.7	44.0	44.2	44.2
B-11	38.0	38.0	39.0	39.5	40.0	40.7	40.4	40.5	41.2	40.8
B-13	35.0	35.0	36.0	36.5	37.0	37.5	37.3	37.5	38.0	37.7
B-14	31.0	30.5	32.0	32.0	32.5	32.0	32.5	33.0	33.0	32.7
B-15	23.5	27.5	29.0	29.0	29.5	30.0	30.0	30.0	29.8	29.7
B-16	30.0	30.0	31.0	31.0	31.5	31.0	31.4	32.0	32.1	31.8
C-5	37.0	37.5	38.0	38.0	38.0	38.3	38.0	38.0	38.9	38.8
C-6	37.5	37.0	38.5	38.5	39.0	39.5	39.0	40.0	39.7	39.5
C-7	41.0	42.0	42.5	43.0	43.5	43.5	43.5	44.0	44.6	44.6
C-8	43.0	44.0	45.0	45.0	45.0	45.7	45.8	46.5	46.9	46.8
C-9	43.5	49.5	53.5	53.5	54.0	54.3	52.8	53.0	53.5	52.7
C-10	40.0	40.0	41.0	41.5	42.0	42.5	42.5	43.0	43.2	43.0
C-11	37.5	37.0	38.0	38.5	39.0	39.9	39.6	40.0	40.0	39.8
C-12	43.0	44.5	45.0	47.5	47.5	49.0	49.0	46.5	47.2	48.4
C-16	28.0	27.5	29.0	29.0	29.5	30.0	29.6	29.5	29.7	29.6
D-1	41.0	42.0	43.0	43.5	43.5	43.8	43.5	44.0	44.2	44.3
D-2	42.0	43.0	43.5	43.5	44.0	44.2	44.2	44.5	45.0	45.0
D-3	42.5	43.5	45.5	46.0	46.5	47.2	46.8	47.5	45.1	47.6
D-4	42.5	42.5	43.5	44.0	44.5	45.3	45.3	45.5	43.6	46.2
D-5	42.5	42.5	43.5	44.0	44.5	45.3	45.2	45.5	44.0	45.7
D-6	41.0	41.0	42.5	43.0	43.5	44.0	44.0	44.5	42.0	44.6
D-7	39.0	39.0	40.0	40.5	40.5	41.5	41.3	41.5	40.5	41.8
D-8	39.5	40.5	41.5	42.0	42.0	42.0	42.0	42.0	42.5	42.5
D-10	39.0	40.0	41.0	41.0	41.0	42.0	41.2	42.0	42.6	42.5
D-11	39.0	38.5	40.0	40.5	40.5	41.4	40.9	41.5	42.1	42.0
D-12	39.5	41.0	42.0	42.0	42.5	42.5	42.4	43.0	43.0	43.3
D-13	40.0	41.0	42.0	42.5	42.5	43.0	42.6	43.0	43.7	43.6
D-14	41.0	42.5	43.5	43.5	44.0	44.0	44.2	44.5	45.1	45.1
D-15	41.0	42.5	43.5	43.5	44.0	44.3	44.3	44.5	44.6	45.0
D-16	41.0	42.0	43.5	43.5	43.5	43.7	43.8	44.5	44.2	44.9
D-18	42.5	42.5	44.0	44.0	44.0	44.2	42.7	43.5	46.0	46.2

Table E.2. Continued

Location	Temperature (°C) at — November 27, 20 Mw	November 27, 5:10 PM 22.5 Mw	November 27, 7:50 PM 25 Mw	November 27, 9:10 PM 27.5 Mw	November 27, 10:45 PM 30 Mw	November 28, 3:00 AM 30 Mw	November 28, 7:00 AM 30 Mw	November 28, 11:00 AM 30 Mw	November 28, 3:00 PM 30 Mw	November 28, 7:00 PM 30 Mw
E-1	34.5	34.5	35.0	35.5	36.0	37.0	36.9	37.0	36.8	36.9
E-2	34.0	34.5	35.5	36.0	36.5	37.0	37.0	37.0	35.6	36.9
E-3	36.0	35.5	36.5	37.0	37.0	37.5	37.0	37.5	38.1	37.7
E-4	36.0	35.5	36.5	37.0	37.0	37.0	37.2	37.5	37.9	37.7
E-5	36.0	35.0	36.5	36.5	31.0	37.2	36.8	37.0	37.5	37.2
E-6	34.0	34.5	35.5	36.0	36.5	37.0	36.8	37.0	36.3	36.9
E-7	36.0	35.0	36.5	36.5	37.0	37.0	36.9	37.0	38.5	37.2
E-8	33.5	33.0	34.0	35.0	35.5	36.0	35.5	36.0	36.1	36.0
E-9	34.0	34.0	35.0	35.5	36.0	36.5	36.0	36.5	36.3	36.5
E-10	34.0	34.5	33.5	35.5	36.0	36.5	36.0	36.5	38.7	36.5
E-12	36.0	34.5	37.0	35.5	35.5	36.0	36.0	37.5	37.9	37.7
E-13	35.0	34.5	36.0	35.5	35.5	36.0	36.0	37.0	36.9	36.8
E-14	34.0	34.0	35.0	35.5	36.0	36.3	36.0	35.0	36.0	36.2
F-1	34.0	34.0	35.0	35.5	36.0	36.3	36.0	37.0	36.8	36.2
F-2	34.0	34.0	35.0	35.5	36.0	36.3	36.0	37.0	37.8	36.2
F-3	34.0	34.0	35.5	35.5	36.0	36.3	36.0	37.0	38.0	35.6
F-4	34.5	34.5	36.0	35.5	35.5	36.0	36.0	36.5	37.2	36.8
F-5	34.5	34.5	36.0	35.5	35.5	36.0	36.0	36.5	37.0	36.8
F-6	34.0	34.0	35.0	35.5	36.0	36.2	33.5	36.0	36.8	35.7
F-7	34.0	34.0	35.5	35.5	35.5	36.0	36.0	36.5	36.5	36.5
F-8	30.5	30.5	31.5	31.5	32.0	32.3	32.2	32.0	32.4	32.3
F-9	28.5	28.0	29.5	29.5	30.0	30.5	30.2	30.5	30.6	30.4
F-11	34.0	33.5	35.0	35.5	35.5	36.0	36.0	36.0	36.1	36.2
G-5	34.5	34.0	35.5	35.5	35.5	35.8	36.0	36.0	35.9	36.6
G-6	32.0	32.5	35.0	33.5	33.5	36.0	36.0	36.0	33.5	34.2
G-7	31.5	31.5	32.5	33.0	33.5	33.7	33.7	34.0	33.8	33.8
G-8	32.5	32.5	33.5	34.0	34.5	34.8	34.5	34.5	34.7	34.7
G-9	32.0	32.5	34.0	34.0	34.0	34.2	34.0	34.0	36.9	34.3
G-10	34.5	35.0	36.0	36.0	36.0	36.2	35.8	36.5	39.2	36.5
G-11	36.0	37.0	37.5	37.5	37.5	37.8	37.2	37.5	38.0	38.2
G-12	36.5	37.5	38.5	38.5	38.5	38.7	38.3	38.5	38.9	38.9
G-13	32.5	32.5	33.5	33.5	34.0	34.5	34.3	34.5	34.6	34.6
G-14	33.5	33.5	35.0	35.0	35.5	36.0	35.5	36.0	36.1	36.1

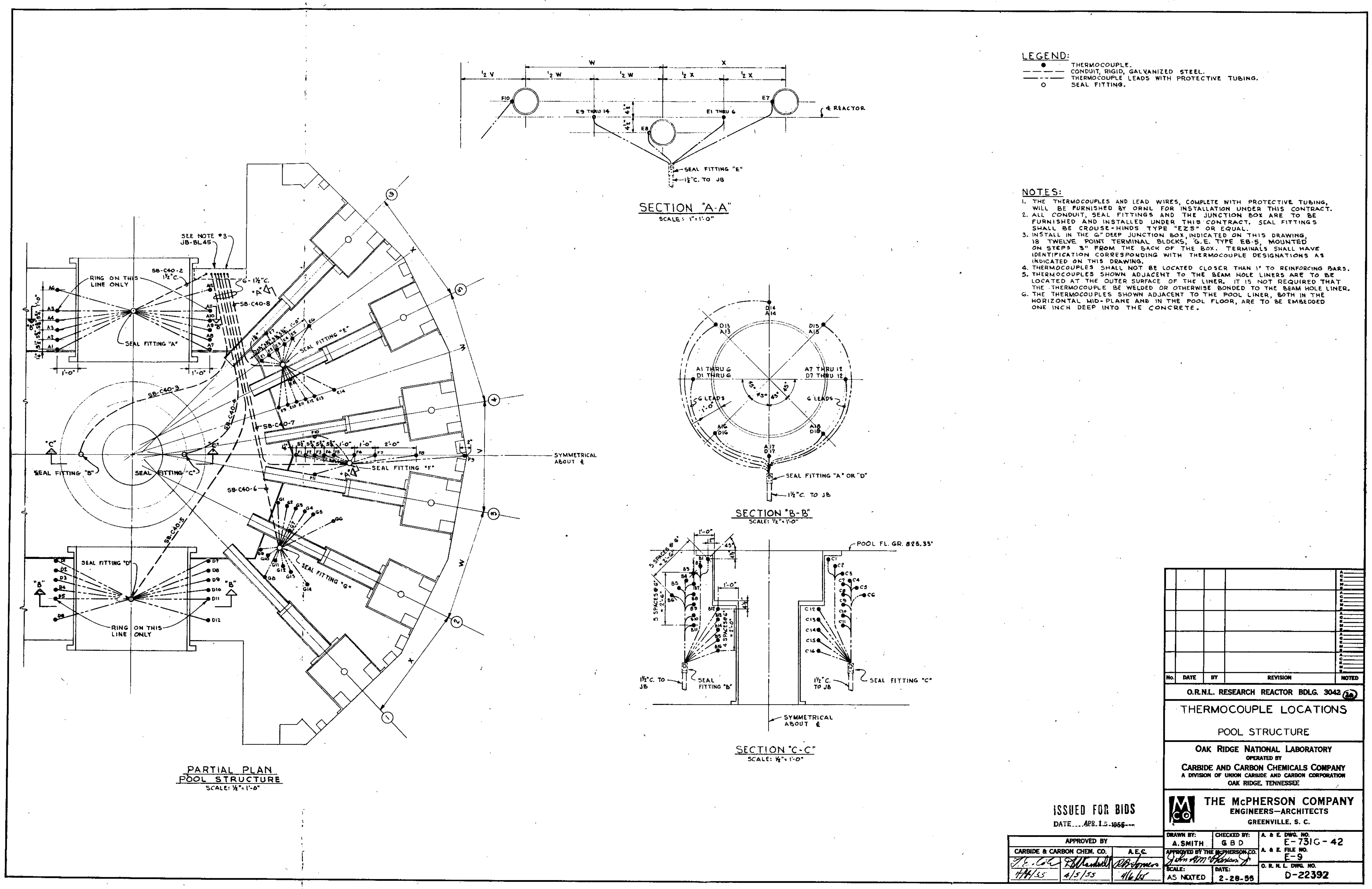

Fig. E.1. Schematic Diagram of Thermocouple Locations in the ORR.

ORNL-LR-DWG 32175

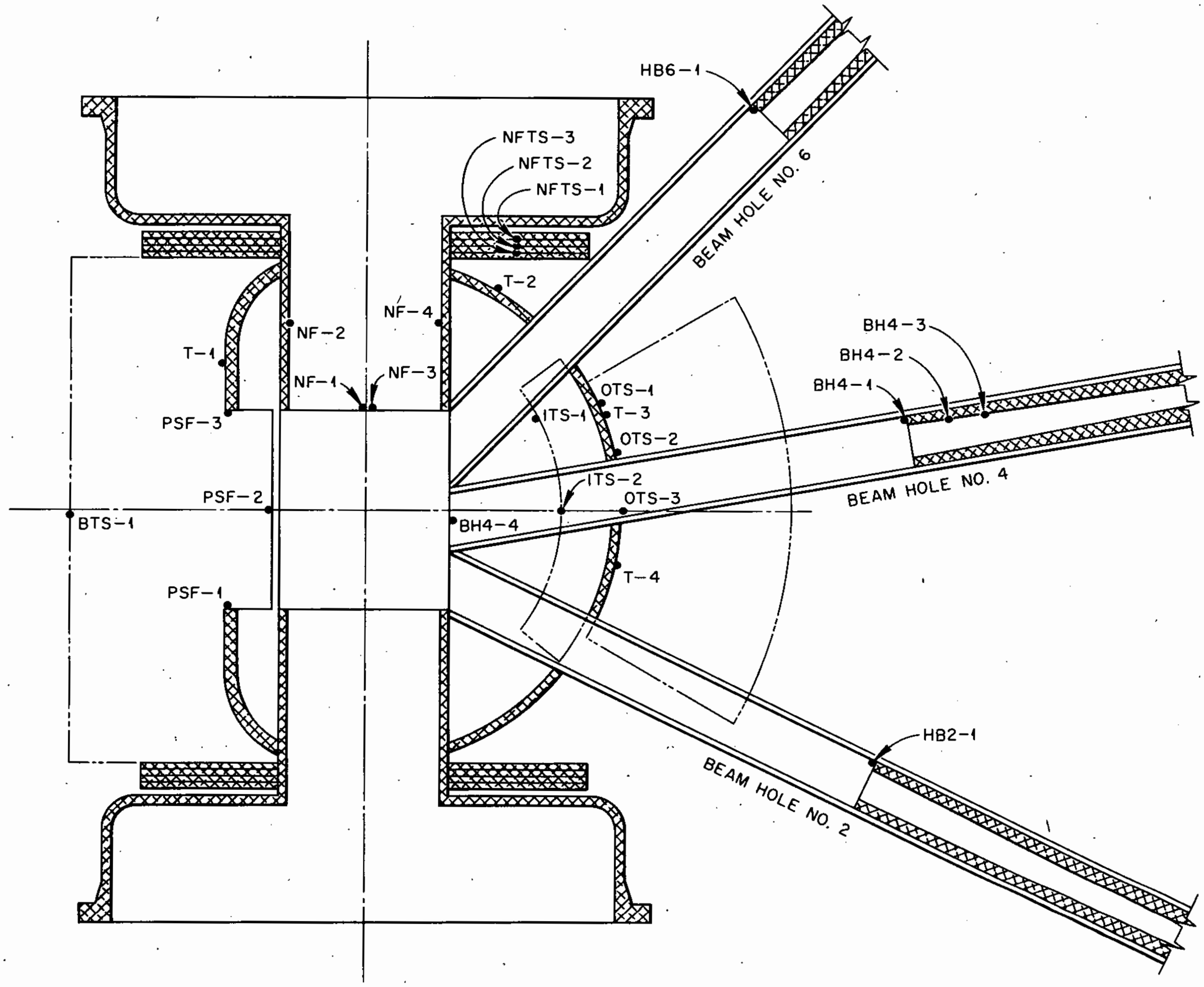

Fig. E.2. Thermocouple Locations in the ORR.

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