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THE LOW-INTENSITY TESTING REACTOR - SAFETY ANALYSIS

F. T. Binford

C. C. Webster

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THE LOW-INTENSITY TESTING REACTOR - SAFETY ANALYSIS

F. T. Binford and C. C. Webster

1. INTRODUCTION

The Low-Intensity Testing Reactor (LITR) originally served as a mock-up of the Materials Testing Reactor (MTR). After completion of the MTR hydraulic tests, it was decided to utilize the facility as a critical assembly for the purpose of investigating the nuclear characteristics of MTR cores. Following the critical tests and some modifications, the reactor was operated at a power level of 1500 kw and was employed as a training facility for future MTR operators. In 1953 the power level was increased to 3000 kw and the LITR has been utilized as a general research facility since that time.

Safety analysis and approvals to operate were included in a series of letters and documents which are summarized in Appendix A. The present analysis of the consequences of potential accidents to the LITR is the second of two documents which may be considered to be a current safety analysis of the reactor. The first of these, The Low-Intensity Testing Reactor - A Functional Description, by C. D. Cagle and W. R. Casto, will be published as ORNL TM-1737. It presents a reasonably detailed description of the LITR and its ancillary facilities. The purpose of the present document is to provide information concerning the safety aspects of the LITR as it now exists. Throughout the report emphasis is placed primarily on protection against environmental contamination.

The LITR is a light-water-moderated and -cooled, beryllium-reflected research reactor designed for use as a general purpose research tool. The reactor, which is described in detail elsewhere,¹ employs highly enriched uranium-aluminum alloy, plate-type fuel of conventional design.² Since 1953, it has been operated at its present power level of 3 Mw.

The design of the reactor was constrained by the original MTR design; however, the changes and modifications required to convert the original hydraulic mock-up into an operating reactor were, to a very large extent,

dictated 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 if the reactor is 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 systems³ and by the operating procedures⁴ and is guaranteed by the reactor containment features.⁵

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.⁶ Although this system is actually vented through the seal tank⁷ to the off-gas system, any effluent must pass through an appropriate arrangement of filters before being discharged from the 3039 stack.⁸

The second type of containment, the building containment, is provided by the LITR dynamic containment or "confinement" system.⁵ Air is constantly removed from all areas adjacent to the reactor vessel and exhausted through the cell-ventilation system. This system, designed to remove ~1600 cfm of air, directs its effluent through appropriate filters to the 250-ft ORNL stack, with a stack discharge elevation of 1064 ft. It provides inleakage of air to those areas susceptible to contamination, thus preventing outleakage of air or vapor-borne fission products 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. 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 containment, it is necessary that certain air-operated dampers be opened in order to increase the flow through the affected areas. These are activated automatically upon receipt of a signal from one, or both, of the continuous air monitors located in the east and west rooms, respectively, or activation may be accomplished manually.

A second type of environmental contamination involves the accidental introduction of contaminated water into White Oak Creek as a result of spillage and run-off or because of the failure or malfunction of the LITR aqueous waste system.⁹ Such events are effectively prevented by the design of the system¹⁰ and by the operating procedures 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 introduction of contaminated water into the watershed at the LITR site and the appearance of activity at the mouth of the creek is in excess of 12 hours.¹² This gives ample time to take controlling action and to spread the warning downstream.

2. METHOD OF ANALYSIS

The method which has been selected to pursue the investigation of the environmental consequences of a reactor accident is quite similar to that used in the cases of the High Flux Isotope Reactor (HFIR)¹³ and the Oak Ridge Research Reactor (ORR).¹⁴ Moreover, because of the close proximity of the ORR and LITR, because they utilize a common stack, and because of their great similarity (except for power level), the results for the LITR have been obtained by means of extrapolation of those found for the ORR.

The method itself consists of the examination of the consequences of a unit release of activity by means of the "Gaussian Plume" model¹⁵ followed by the application of the results so obtained to releases which could result from various credible accidents following or during operation.

The only credible cause for a massive release of fission products from the LITR core 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 heat-transfer capability, 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 upon whether the fission products escape directly from the vessel into the building containment 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 vent to the seal tank before passing through the filters to the stack. There are only four 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 by either the safety system or by one or more of the 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 complete and rapid loss of all coolant during power operation.
4. A local flow blockage during full-power operation.

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

5. An accident to an experiment which contains radioactive material.
6. An accident involving radioactive material other than that contained in the fuel or an experiment; i.e., contaminated waste, ion exchange resin, etc.
7. 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 which may result in significant environmental contamination will be considered in detail.

3. THE CONSEQUENCES OF THE EMISSION OF FISSION PRODUCTS FROM THE ORNL 3039 STACK FOLLOWING AN ACCIDENTAL RELEASE

Calculation of the downwind radiation doses to be expected following the emission of fission products from the ORNL 3039 stack has been discussed in detail in ORNL 4086 and the application of the results to the case of the ORR is described in ORNL TM-1824. While a direct extrapolation of the doses due to activity contained in the stack plume to the case of the LITR is valid and has been utilized, it is necessary to modify the results of the radiation dose delivered by fission products contained in the building because of the difference in the volumes of the LITR and ORR buildings. This has been done. Because of its smaller size, release of a unit quantity of fission products into the LITR building will produce a higher dose rate near the building than will a similar quantity released into the ORR building. At distances greater than a few building diameters, this effect is no longer noticeable.

The internal doses to be expected due to inhalation of iodine following a 1-kw release have been plotted in Figs. 3.1 and 3.2. The curves give the doses both as a function of distance downwind and of exposure time and are computed for both most representative and inversion conditions.* Similar curves are given in Figs. 3.3 and 3.4 for the whole body gamma dose from the noble gases.

The direct gamma ray dose delivered by the radioactive material in the building is displayed in Figs. 3.5 and 3.6. In addition, this dose has been computed as a function of time for several nearby locations. The results are given in Figs. 3.7 and 3.8. 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 shown are gross upper bounds because of the very liberal

* Most representative conditions are characterized by C-type stability and a wind speed of 100 meters/min. Inversion conditions employ F-type stability, a wind speed of 50 meters/min, and an inversion "lid" at the physical stack elevation.

Fig. 3.1. Internal Iodine Dose as a Function of Distance Downwind and Exposure Time (Most Representative Conditions)

Fig. 3.2. Internal Iodine Dose as a Function of Distance Downwind and Exposure Time (Inversion Conditions)

Fig. 3.3. Whole Body Noble Gas Dose as a Function of Downwind Distance and Exposure Time (Most Representative Conditions)

Fig. 3.4. Whole Body Noble Gas Dose as a Function of Downwind Distance and Exposure Time. (Inversion Conditions)

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Fig. 3.5. Whole Body γ Dose Due to Direct Radiation from the Building vs. Distance from the Building (LITR); Noble Gases - Noble Gas Daughters - Noble Gas Daughters of Iodines

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Fig. 3.6. Whole Body γ Dose Due to Direct Radiation from the Building vs. Distance from the Building (LITR); Iodines

Fig. 3.7. Direct Radiation Dose Delivered at Various Points Near the Building, Noble Gases (LITR)

Fig. 3.8. Direct Radiation Dose Delivered at Various Points Near the Building, Iodines (LITR)

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 walls, structures within the building or other structural material which may intervene between the building and the receptor. (These assumptions also hold for the values given in Figs. 3.5 and 3.6.) Thus, it is quite likely that the doses computed are conservative by a factor of two or more.

All of the data cited above are given on a "per kilowatt" basis; i.e., for 3-Mw operation under the assumption that 1/3000 of the fission products are released. Application to any given accident situation can easily be made by use of the appropriate multipliers. For example, if at 3 Mw an accident releases 10% of the noble gases and 5% of the iodines, then the noble gas doses given in Figs. 3.3, 3.4, 3.6, and 3.8 should be multiplied by 300 while the iodine doses given in Figs. 3.1, 3.2, 3.5, and 3.7 should be multiplied by 150. Appropriate factors must also be included to account for iodine retention in the pool water and, in the case of Figs. 3.1 and 3.2, decontamination by the charcoal filters.

4. 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 serious damage to the LITR core. Overheating could 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.

4.1. Reactivity Accidents

By a reactivity accident is meant a situation 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 the 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 LITR, this would amount to an average rate of 0.045% reactivity per second over the entire length of the rods, with the maximum rate being 0.095% per second 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, which with respect to its nuclear characteristics is almost identical to the LITR, have been investigated by analogue techniques,¹⁶ and it has been determined that the consequences of such accidents are trivial and would not cause damage to, or even a significant temperature rise in, the ORR core. Because the rate of reactivity addition in the ORR case is nearly 50% greater than that for the LITR startup accident, it can be concluded that the startup accident cannot cause damage to the LITR core.

Because the LITR core may initially contain as much as 7% 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.

An examination of the analogue calculations mentioned above and comparison with the Spert experiments¹⁷ lead to the conclusion that the LITR can withstand a step input of at least 2% in reactivity without melting. This conclusion is conservative because it is made under the assumption that no period safety protection is available. The inclusion of the period scram circuit would permit the reactor to withstand a reactivity step of ~3% in reactivity.¹⁸ 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 reasonable to assert that it would require the rapid addition of at least 2% in reactivity to the LITR in order to initiate a transient of sufficient magnitude to cause significant damage to the LITR 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 halfway withdrawn.* Thus, when shut down, it would require that one of the three control rods be fully withdrawn and another halfway withdrawn in order to even reach criticality. This cannot be done rapidly. The withdrawal of all the control rods has been discussed above.

Of the core hardware, the fuel elements, beryllium-reflector pieces, and control rods possess significant amounts of reactivity. Since the 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 retained by positive stops at the shock absorbers,¹⁹ they cannot fall out of the core. Removal and reinsertion of a fuel element by the motion of a control rod or experiment apparatus is effectively prevented by the upper and lower grids.²⁰

There remains only the possibility that failure or malfunction of an experiment could result in a reactivity increase. This is guarded against by a careful review of all experiments²¹ to make sure that no experiment is installed which, by credible failure, could rapidly introduce more reactivity than can be handled by the servo control system (~0.5% in reactivity).

4.2. Loss-of-Cooling Accidents

Loss of adequate cooling capacity due to a reactor "walkaway", i.e., an undetected gradual increase in power, or due to a gradual loss in cooling flow are effectively prevented by the level-safety scram and by the low-flow scram.²² In addition, the reactor ΔT and outlet temperature²³

* See Appendix B.

would operate to terminate such a situation before dangerous levels are reached. All four of these are elements of the reactor safety system and are characterized by redundancy and maximum reliability. Moreover, as shown in Appendix C, even with only natural convection flow the margin between the burnout heat flux and the maximum operating heat flux at 3 Mw is 1.3. Thus, loss of flow alone will not result in fuel plate melting. Clearly, no forced convection afterheat protection is required.

Probably the most likely cause of overheating is as a result of a local flow blockage in one or more fuel elements. Such blockages have occurred at several reactors,^{24,25} including the ORR,²⁶ and in the cases cited have caused minor amounts of melting. In the case of the LITR, operating at 3 Mw, it is extremely doubtful that a local flow blockage could cause any melting at all. In the case of the ORR fuel plate melting incident, an inspection of the charts indicates that detectable fluctuations began at a power level of 9 Mw and that burnout occurred at approximately 12 Mw. The fuel element involved was almost completely blocked at the top. It is reasonable to suppose, therefore, that a similar blockage in a LITR element with the reactor operating at only 3 Mw would be unlikely to cause detectable oscillations, much less melting. On the other hand, if the fuel plates became coated with some substance, such as melted or charred plastic, which drastically reduced the transfer of heat to the water some melting might occur.

Flow blockages 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.

The 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 insure that these procedures are carried out. This has been done at the LITR.

Despite the fact that most types of flow blockages cannot cause overheating in the LITR and that strictly enforced administrative procedures to prevent such occurrences are enforced, it appears that a local 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; and, in the cases experienced so far, there has been only a very minor release of fission products from the primary coolant system.

5. THE MAXIMUM HYPOTHETICAL ACCIDENT

Based upon the discussion in Section 4, it appears highly unlikely that any malfunction of the LITR could result in a major release of fission products. In fact, it is difficult to postulate realistically a meltdown which involves more than a very small fraction of the fuel. About the only possible event which could result in a major meltdown would be a catastrophic failure of the reactor vessel which caused rapid loss of all of the cooling water. Such an occurrence would quite probably result in some melting, although a recent analysis of data obtained during a series of water-loss tests^{*} indicates that the maximum fuel plate surface temperature would not, following rapid loss of coolant during operation at 3 Mw, exceed the melting temperature of aluminum. Nevertheless, for the purpose of examining the worst hypothetical accident, it is assumed that the reactor operating at 3 Mw suffers complete loss of coolant followed by a complete core meltdown in which 100% of the noble gases and 50% of the iodines are released. It is also assumed that 100% of the noble gas daughters of the iodines are released. Because the postulated release does not occur underwater, no credit is taken for removal of iodine by dissolution. The building air, however, is directed through filters before being passed up the 3039 stack. These filters are designed to have a minimum decontamination factor for iodine of thirty. They are normally operated with a decontamination factor of 100 or more. The filter factor for iodine is, therefore, taken to be 100. Although the noble gases are retained to some extent in the filters, it will be assumed that they reach the stack plume undiminished except by decay.

^{*} See Appendix C.

The radiation doses to be expected downwind following the postulated maximum hypothetical accident have been computed using the curves presented in Section 3 and the assumptions discussed above. The results of these computations for the doses due to the stack plume are given in Figs. 5.1 and 5.2. Both "most representative" and "inversion" meteorological conditions are shown. It is at once obvious that the stack release presents no problem. Even for an infinite exposure, the maximum iodine dose is nearly two orders of magnitude less than the 300 rem suggested in the guidelines of 10 CFR 100. (From this it may be deduced that a filter factor of about 1.6 is all that is required to keep the peak below this value.) The infinite exposure noble gas dose has a maximum which is a factor of 50 below the 25 rem set forth in 10 CFR 100. Since the doses delivered in finite time will all be less than those for infinite exposure, the latter have not been shown in the figures.

On the other hand, the direct radiation from fission products in the building following the maximum hypothetical accident presents a much more serious radiation hazard to persons remaining near the LITR for any length of time following the release. The whole body gamma dose delivered in this way is plotted in Figs. 5.3 and 5.4 as a function of distance from the building and exposure time. It can be seen from these figures that persons closer than about 65 meters to the building must evacuate quickly in order to avoid receiving whole body doses in excess of 25 rem. It should be recalled, however, that the doses presented in these curves have been calculated under the extremely conservative assumptions: (1) that all the released fission products are released 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 represent a gross upper bound.

The initial dose rate one meter from the building is 160 rem per minute and this falls to less than 10 rem/min at 35 meters. Thus, an individual standing just outside the reactor building could probably escape very serious consequences if he took swift action to leave the scene. The initial external whole body dose rate within the building would be about 350 rem/min; however, the initial internal dose rate from ^{131}I alone

Fig. 5.1. Internal Iodine Dose (LITR-MHA) Infinite Exposure

Fig. 5.2. Whole Body Gamma Ray Dose (LITR-MHA) Infinite Exposure (Stack Plume Only)

Fig. 5.3. Whole Body γ Dose from Direct Radiation (LITR-MHA)

Fig. 5.4. Direct Radiation Dose Near the LITR Building Following The Maximum Hypothetical Accident

would exceed 7300 rem/min. Thus, persons in the reactor building would likely become casualties unless they escaped before the fission gases became mixed with the atmosphere in the building.

In summary, it is 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 15-20 minutes) would insure that only personnel initially in the immediate vicinity of the LITR would be exposed to the risk of receiving radiation doses in excess of those set forth in the guidelines of 10 CFR 100. Those in most danger would be the staff personnel whose place of work is located near the reactor. This involves approximately 185 people located within a radius of 65 meters of the LITR building. The location affected is shown in Fig. 5.5. 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 maximum hypothetical accident. It should be noted that the figure of 185 people is the normal daytime complement. On weekends and night shifts (~76% of the time) this number is reduced to a maximum of about 12. Finally, it is emphasized that the accident discussed above represents a gross upper bound and does not represent a credible accident.

6. POTENTIAL ENERGY RELEASE

None of the credible accidents discussed in Section 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.²⁷

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,

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Fig. 5.5. Buildings Located Within a 65-Meter Radius of the LITR

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.²⁸

It is 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.

7. 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 insure that these criteria are satisfied, all experiments must undergo an internal safety review²⁹ 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, the experiment may be approved for insertion in the reactor by the Technical Assistance Department. If any significant hazard did exist, 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 which are 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

frequently 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 them 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; and 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 LITR would be as a result of some previous major accident which required the use of large quantities of water for cleanup, in which case special emergency procedures would be put into effect.

The fission products of most significance with respect to contamination of watershed are long-lived isotopes ^{90}Sr , ^{106}Ru , ^{137}Cs , and ^{144}Ce . These, together with their characteristics, are listed in Table 7.1. Clearly, the controlling factor here is ^{90}Sr .

In a study of the ORNL waste disposal pits undertaken in 1960,⁸⁰ it was found that, with the 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 few percent of the inventory listed in Table 7.1 would escape from the fuel, makes it appear unlikely that any sizable quantity of activity would reach the watershed.

Table 7.1. Characteristics of the Long-Lived Products

Isotope	Half-life	MPC*	Critical Organ	LITR** Inventory
^{90}Sr - ^{90}Y	28 y	1×10^{-7}	Bone	1.7×10^3
^{106}Ru - ^{106}Rh	1 y	1×10^{-5}	Kidney	2.9×10^3
^{137}Cs	26.6 y	2×10^{-5}	Muscle	1.7×10^3
^{144}Ce - ^{144}Pr	290 d	1×10^{-5}	Bone	0.6×10^5

* Nonoccupational MPC for water from Report of Committee II on Permissible Dose for Inland Radiation, Pergamon Press, New York, 1959.

** Based on average burnup of 16% in 4.4 kg core.

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, located ~1.4 miles downstream, is equivalent to that received from approximately one week 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.

8. HISTORY OF OPERATING DIFFICULTIES

Operation of the LITR has proceeded in an orderly fashion during the 16 years since routine operation was begun. Since 1958, the reactor has been in operation approximately 88% of the available time. The percent of on-line time has been summarized both annually and by quarters and the results are presented graphically in Fig. 8.1. Factors that contributed to the lower-than-average percent operating times were as follows:

Fig. 8.1. LITR Operating Time as a Function of Time

1. In 1958 there were a large number of shutdowns due to failures (mechanical and electrical) of certain experiments.

2. In 1961 the reactor downtime was extended to permit major repair of ionization chambers.

3. In 1963 a scheduled shutdown was extended to permit the following changes: (a) installation of side access tubes in the "A" tank of the reactor vessel and (b) installation of the new reactor instrumentation. An 8-week cycle of operation was started during the last quarter to consolidate the major experiment and reactor activities.

4. In 1966 the reactor was shut down due to the rupture of several tubes in the heat exchanger and a failure in the normal off-gas system due to faulty installation.

In general, the trend has been toward a higher percent operating time at power; however, as indicated above, several major items have contributed to excessive downtime.

Operating difficulties have been relatively minor and in most cases have been corrected by design improvements and replacement of original equipment with more modern components as these components have been developed. In addition, increasing familiarity with the characteristics of the reactor has made it possible for the operators to recognize developing troubles and take steps to correct them before they become problems.

Probably of most interest in this connection are the difficulties encountered with the shim-safety rods and their associated drive mechanisms and electronics. Presented below is a tabulation of the significant history of the shim-safety rod behavior, together with a listing of the corrective measures which have been taken since 1958.

1. Fail-to-drop incidents.--On March 10, 1964, the No. 1 shim rod did not respond to the "Jordan" button test (a test which simulates a power level $>150\%$ nominal full power) during the shutdown. Corrective action was taken immediately and the No. 2 level-safety sigma amplifier was replaced. The unit performed satisfactorily.

2. Unscheduled rod drops.--A bar graph, Fig. 8.2, indicates the number of rod drops by year with brief remarks pertinent to the occasions.

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Fig. 8.2. Unscheduled Rod Drops - LITR

The causes of the rod drops are due to the following: internal breakdown of the magnet, component failure in the drive assembly exclusive of the magnet, and electronic noise in the period channel.

In analyzing the information, one notes a reduction in drops, which is attributable to a preventive maintenance program.

3. Component replacement.--From 1958 through 1966 several important improvements were made. Some were of necessity to replace obsolete or faulty components while other changes were made to improve the operability of the facility. They are listed as follows:

a. In 1962 a new shim-rod drive was installed in the No. 3 rod position. This was an effort to standardize the three drive systems, thereby minimizing the maintenance requirements, and to provide a more reliable system of drive units.

b. In 1963 a total of seven access ports were made in the "A" tank of the reactor vessel to permit the accommodation of experiment access tubes which alleviates the cumbersomeness of access tubes passing through the top plug unit. Also in 1963, eleven electromagnets were replaced. These replacements were necessitated by faults which developed in the magnet housing or within the magnet.

c. In 1964 the nuclear and process instrumentation was replaced with a system of current design. Also, instrumentation for remote operation from the ORR control room was completed.

d. In 1965 permanent electronic equipment was installed which permits a more efficient manner of checking shim rod time-of-flight and magnet-release time measurements.

Also, a new upper grid assembly was designed and installed to permit a more versatile use of the core region for accepting experiments.

Since 1958, the safety circuits of the LITR have been tripped only twice because a reactor operating parameter exceeded its limit (i.e., for "real cause"), once in 1965 and once in 1966. On the other hand, over the same period of time "real" signals originating in the experiment safety circuits initiated 37 safety circuit trips. These trips (due to "real cause") are described below.

1. 1966 - In the majority of instances concerning experiments, the "high temperature setpoint" of an experiment was tripped due to the temperature lag of reactor power. This does not constitute a "significant situation" with respect to safety; however, it has been an operating nuisance. Two of the fifteen trips resulted in a reactor scram while the remaining thirteen produced reactor setbacks.

The "high temperature setpoint" of the reactor outlet water system was tripped on one occasion in 1966.

2. 1965 - In each instance concerning experiments, the "high temperature setpoint" of an experiment was tripped due to a "temperature overshoot". In each case, a setback of reactor power resulted.

The one occasion of a reactor safety circuit trip was due to the negative ΔT .

3. 1963 - Each tripping of the safety circuit resulted from an experiment and was due to "temperature overshoot" which resulted in actuating the high temperature setback or scram point.

4. 1962 - The one trip of the safety circuit resulted from an experiment and was due to "temperature overshoot" which actuated the high temperature setback.

5. 1960 - Each tripping of the safety circuit resulted from an experiment and was due to "temperature overshoot" which resulted in actuating the high temperature trip point, causing a setback of reactor power.

6. 1959 - The one trip resulted from an experiment and was due to "temperature overshoot" which actuated the high temperature setback.

7. 1958 - Each tripping of the safety circuit resulted from an experiment when the "high temperature setpoint" designed to protect the experiment was actuated and resulted in a setback of reactor power. Fifty percent of the incidents were due to the same experiment.

From an examination of the foregoing information, together with a more detailed analysis of each experiment, it may be concluded that in no case was the reactor in real danger and that it is quite likely that the experiment trip points are set too close to the operating level of the experiments. Nevertheless, the operating inconveniences resulting from this are tolerable.

A complete listing of all occasions when the safety circuits have been tripped is given in Table 8.1.

A review of abnormal events involving radiation surface contamination due to the operation of the reactor and the operation of the experiments at the LITR indicates a distribution as tabulated below:

	<u>Reactor</u>	<u>Experiment</u>	<u>Total</u>
High radiation		15	15
Surface contamination	3	5	8
High air activity	<u>3</u>	<u>13</u>	<u>16</u>
Total	6	33	39
Building evacuations	1	4	5

In analyzing the above data, it is worth noting that the number of high-radiation level alarms produced by experiments is high, but not unexpected since the nature of the major effort in experiments at the LITR is the study of fuel-element claddings and, generally speaking, the experiments are expected to emit some level of fission products.

The occasions of high air activity were encountered during experiment failure and/or removal and were generally caused by mechanical failure of components and/or improper manipulation of equipment.

The building evacuations, which averaged about one a year, were not lengthy, with a maximum duration of about 2 hr. It is also worthy to note that the evacuations were local in nature; i.e., only a portion of the LITR building was evacuated. Normally, only that area where an experiment was operating or was being removed was affected.

While the events, as recorded, are large in number, it is noteworthy that no personnel received either external or internal exposure in excess of tolerance.

A distribution of the events on a yearly basis is tabulated below. While no particular pattern is evident, it can be seen from the record of the last three years that efforts to reduce the number of such incidents have borne fruit.

Table 8.1. Occasions When the LITR Safety Circuits Have Been Tripped

Cause	1966	1965	1964	1963	1962	1961	1960	1959	1958	Total
Experiments										
Instrument failure	6	15	1	8	6	5	6	13	21	81
Mechanical or component failure	4	13	1	5	4	5	9	18	36	95
Real cause	15	3		6	1		3	1	8	37
Human error	2	2		2	1		1	5	7	20
Electrical power fluctuation									1	1
Other	—	—	—	—	—	—	—	—	3	3
Subtotal	27	33	2	21	12	10	19	37	76	237
Reactor										
Instrument failure	1	5		2	5	1		4	2	20
Mechanical or component failure						1				1
Real cause	1	1								2
Human error		2						1		3
Electrical power fluctuation	6	2	2	4	6	5	6	1	2	34
Other	—	—	—	—	—	—	—	—	1	1
Subtotal	<u>8</u>	<u>10</u>	<u>2</u>	<u>6</u>	<u>11</u>	<u>7</u>	<u>6</u>	<u>6</u>	<u>5</u>	<u>61</u>
Total	35	43	4	27	23	17	25	43	81	298

	<u>1958</u>	<u>1959</u>	<u>1960</u>	<u>1961</u>	<u>1962</u>	<u>1963</u>	<u>1964</u>	<u>1965</u>	<u>1966</u>
High radiation	1	4	1	2	2	2		1	2
Surface contamination		1		1	4		1	1	
High air activity		2		6	5		3		
Building evacuation		2		1	1		1		

9. CONCLUSIONS

It may be concluded from the analysis presented above that the Low-Intensity Testing Reactor can be operated at 3 Mw in an entirely safe and orderly fashion. That this is true has been amply demonstrated by the excellent record of operation compiled over the past 15 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 LITR 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.

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APPENDIX A

STATUS OF SPECIFIC ACRS AND AEC RECOMMENDATIONS
WITH RESPECT TO LITR OPERATIONS AS OF JANUARY 1, 1966

Included below is a list of all pertinent correspondence and other documentation associated with approvals, requests for approvals, and supporting data regarding the operation and safety performance of the LITR. Also included are brief summaries of the contents of each document.

Those items identified by a single asterisk are requests for approval by ORNL or requests for a safety analysis by the USAEC. Those identified by a double asterisk are approvals by the USAEC or are comments by the USAEC regarding the supporting information which accompanied the request or approval.

1. October 27, 1949, from W. R. Gall and D. J. Mallon to MTR Design File, ORNL CF-49-10-185.

Subject: Mock-Up Design Report.

This report gives photographs, drawings, design calculations, technical specifications, and other pertinent engineering data associated with the major components used in constructing and assembling the MTR and the MTR mock-up (LITR). A list of the original drawings appears as Appendices B and C.

2. May 11, 1950, from S. E. Beall, et al, to M. M. Mann, ORNL CF-50-5-140.

Subject: Interim Report of Neutron and Gamma Experiments on the MTR Mock-Up.

This report discusses the modifications made to the MTR mock-up after completion of the hydraulic experiments performed to obtain design parameters for the MTR. It describes and gives illustrations showing the arrangement of the fuel and reflector materials as well as the shielding that was used during the initial critical test performed with actual MTR fuel. Neutron and gamma measurements were made for the purpose of evaluating the shielding and shield-cooling requirements for the MTR.

Included are several photographs taken during the modification,

as well as a detailed discussion of the control system. Operation during these tests was from the 1-watt power level to the 100-watt power level.

3. May 18, 1950, by S. E. Beall, ORNL-692.

Subject: Procedures for Erection of the Materials Testing Reactor Mock-Up.

This report gives a brief early history of the project and lists personnel required in the assembly operation and the time required. The need for cleanliness is stressed and also the need for proper alignment. A detailed assembly procedure is given, as well as a list of special items required for the job. Several photographs taken during the assembly operation are included.

4. August 15, 1950, from C. E. Larson (UCC) to J. H. Roberson (ORO), ORNL CF-50-8-77 (internal correspondence).

Subject: Conversion of the Mock-Up into a Temporary Facility for Polonium Production.

This letter gives the estimated cost of converting the mock-up to a polonium production facility and operating at a 5-Mw power level and states certain advantages to be gained by converting at the time. (This idea was not implemented.)

5. September 11, 1950, from C. E. Larson (UCC) to J. H. Roberson (ORO), ORNL CF-50-9-29. *

Subject: Proposal to Operate MTR Mock-Up Continuously at 500 kw.

This letter requests approval to operate the MTR mock-up at the 500 kw power level. It estimates an average of 5×10^{12} neutrons $\text{cm}^{-2} \text{sec}^{-1}$ and lists several experimental programs and isotope production that require fluxes this high for further evaluation. It also suggests using the reactor at the power to train MTR operation personnel and to familiarize experimentalists in using an MTR-type machine. It includes the required changes and an estimate of the cost. Mention is made of having run the reactor at 100 kw for a period of one hour.

6. September 13, 1950, from J. H. Roberson (ORO) to C. E. Larson (UCC), ORNL CF-50-9-67 (internal correspondence). **

Subject: Approval to Operate the MTR Mock-Up at 100 kw.

This letter confirms verbal approval to operate the reactor at a power level of 100 kw for one hour to measure existing shielding and induced activity in the cooling water in preparation for operation at 300 to 500 kw power level.

7. October 12, 1950, from L. E. Johnston (IDO) to R. W. Cook (ORO). *
Subject: Training of Contractor Personnel on MTR Mock-Up.

This letter contains suggestions that the MTR mock-up be utilized to train MTR operator personnel and, if convenient, irradiation experiments be performed for ORNL to give operation experience with in-reactor experiments.

8. October 26, 1950, from L. R. Hafstad (USAEC) to R. W. Cook (ORO). **
Subject: Operation of the MTR Mock-Up.

This letter gives approval for operation of the reactor at a low power level (unspecified) for training purposes--the exact level to be established by the MTR Steering Committee. The question of performing irradiation studies in the MTR mock-up will require further consideration as to need and policy.

9. November 6, 1950, from J. H. Roberson (ORO) to C. E. Larson (UCC). **

Subject: Operation of the MTR Mock-Up.

This letter included the letter from Hafstad to Cook as an enclosure and, thereby, implies approval to operate the reactor as a training facility at a power level (as yet unspecified) to be established at a coming meeting of the MTR Steering Committee--training program and costs to be agreed upon at that time.

10. November 16, 1950, from C. E. Larson (UCC) to J. H. Roberson (ORO), ORNL CF-50-11-90.

Subject: Transmittal of Report on Safety Aspects of Operating MTR Mock-Up as LITR.

This letter accompanied copies of ORNL CF-50-11-91, Safety Aspects of Operating the MTR Mock-Up as a Low-Intensity Training Reactor at 500 kw. This appears to be the first official designation of the MTR mock-up as the LITR.

11. November 10, 1950, from S. E. Beall to File, ORNL CF-50-11-91. *
- Subject: Safety Aspect of Operating the MTR Mock-Up as a Low-Intensity Training Reactor.

This report gives a comparison of the LITR with the MTR and describes the several modifications that have been made externally to the reactor tank. Several safety problems are discussed, including excess reactivity with the expected temperature resulting from a sudden addition of same, radiation beams from experiment facilities, release of fission-product activity, and the dispersion of any radioactive waste products which could result from an accident. A planned water-loss test is discussed and results of a run at 175 kw for 5 hours is described. Four photographs of the facility are included.

12. December 18, 1950, from C. E. Larson (UCC) to J. H. Roberson (ORO), ORNL CF-50-12-66 (internal correspondence). *

Subject: Operation of MTR Mock-Up as LITR.

This letter requests approval to operate the LITR up to 500 kw as a low-intensity training reactor for training MTR operating personnel contingent upon favorable review by the Reactor Safeguard Committee.

13. April 2, 1951, from C. E. Larson (UCC) to J. H. Roberson (ORO), ORNL CF-51-4-10. *

Subject: Program for Utilization of LITR.

This letter outlines the major experimental programs that would utilize the LITR as an irradiation facility and requests authorization to operate at a power level not to exceed 1.5 Mw (a value considered safe by the Reactor Safeguard Committee at its March 12 meeting.)

14. May 10, 1951, from N. H. Woodruff (ORO) to C. E. Larson (UCC), ORNL CF-51-5-84 (internal correspondence). **

Subject: Power Levels for Operation of the Low-Intensity Training Reactor (LITR).

This letter authorizes operating the LITR at power levels up to 1 Mw for training purposes, but a temperature study with a loss of coolant and a subsequent review is to be a prerequisite for approval to operate at 1.5 Mw.

15. May 7, 1951, John H. Buck, editor, Extract from ORNL-963.
Subject: Appendix 4, MTR Mock-Up; Appendix 5, Beryllium; Appendix 6, Fuel Elements.

This report gives a brief description of the MTR mock-up and describes some of the mechanical and hydraulic tests performed. It discusses briefly the criticality measurements and the nuclear measurements performed when the critical tests were performed for the MTR design. Gamma-heating and gamma-ray attenuation studies performed with the critical study loading are described. A reference section is given at the end of Appendix 4. Appendix 6 describes the development of the MTR fuel elements, construction of the plates, and fabrication of the elements, as well as including a list of references and a list of reference drawings.

16. August 14, 1951, from C. E. Larson (UCC) to N. H. Woodruff (ORO), ORNL CF-51-8-129 (internal correspondence). *

Subject: Justification for the Continued Operation of the LITR. This letter gives justification for continued operation of the LITR by detailing the experimental program and radioisotope production program that could be best performed at this time in the LITR.

17. September 25, 1951, by S. E. Beall, ORNL-1075.

Subject: An Experimental Determination of Fission-Product Heating After Shutdown of the Low-Intensity Training Reactor. This report gives the experimental results of 12 separate tests over a power level range of 0.5 kw to 300 kw when the water was suddenly drained from the reactor by rapidly opening a valve in the bottom of the reactor tank. The results of the experiments are compared with a theoretical development, and it is concluded that the reactor could operate continuously at 2000 kw and suddenly lose the water without danger of melting a fuel plate.

18. September 27, 1951, from N. H. Woodruff (ORO) to C. E. Larson (UCC). **

Subject: Operation of the LITR as an Experimental Facility.

This letter states that the Director of Reactor Development has given approval to continue operation of the LITR until the end of calendar year 1952 for the performance of irradiation studies as outlined in earlier correspondence (see Item 16, above). Subsequent approvals will be determined by an evaluation of the need and re-assay of expected reactor life with operating experience. However, it was stated that operation of the facility will be under policy direction and coordination of the MTR Policy Board. No power level is stated. Maybe a power level is stated in letter of June 18, 1951, Woodruff to Larson. Lists several letters between September 11, 1950, and June 18, 1951.

19. October 4, 1951, from C. E. Center (UCC) to N. H. Woodruff (ORO).
Subject: Operation of LITR as Experimental Facility.

This letter requests clarification of the statement, "Operation of the facility will be under policy direction and program coordination of the MTR Policy Board." Carbide and Carbon Chemical Company is thereby taking issue with the AEC over such an arrangement.

20. February 21, 1952, from S. E. Beall to J. A. Cox, ORNL
CF-52-2-158 (internal correspondence).

Subject: Operating the LITR (Note by author--loss-of-coolant tests).

This memorandum gives data for loss-of-coolant tests performed at the LITR from power levels of 150 kw to 770 kw. Results of the test support the analysis of safe operation at the 1.5 Mw power level.

21. March 7, 1952, from C. E. Larson (UCC) to K. A. Kasschau (ORO),
ORNL CF-52-3-28 (internal correspondence). *

Subject: Operation of the Low-Intensity Test Reactor.

This letter requests approval for operation of the LITR at a power level up to 1500 kw. It refers to reports on loss-of-coolant tests performed with the reactor as supporting evidence of lack of any additional hazards over those existing at the present 1000-kw level. This is the first official reference to the designation as the Low-Intensity Test Reactor.

22. May 1, 1952, from K. A. Kasschau (ORO) to C. E. Larson (UCC),
ORO-17996. **

This letter gives approval to operate the LITR at a maximum power level of 1.5 Mw through calendar year 1952.

23. August 22, 1952, from S. R. Sapirie (ORO) to C. E. Center (CCCC).

This letter transmits a copy of a newly distributed bulletin, OR-RDV-1 (Serial No. 222), "Reactor Safety Determination." This new outline will affect the supporting data required when requesting approval of changes in the reactor.

24. September 15, 1952, from C. E. Larson (UCC) to Kenneth Kasschau (AEC), ORNL CF-52-9-144 (internal correspondence).

Subject: Justification for Research Reactor at ORNL.

This letter and the attachment (Appendix A) is included here because of pertinent information given on the conditions of the LITR. The neoprene gasket is discussed, as are the steel tanks and other steel parts in the system and the purity of the water as well as the effect of the boiling experiments on the aluminum content of the water.

25. October 4, 1952, from C. E. Larson (UCC) to Kenneth Kasschau (ORO). *

Subject: Justification for the Continued Operation of the LITR.

This letter submits additional information and references other information for evaluation by the AEC of continued operation of the LITR beyond 1952. Reference is made to the neoprene gasket between sections "C" and "D" of the reactor tank as the probable point of failure of the reactor tank.

26. January 13, 1953, from Kenneth Kasschau (ORO) to C. E. Larson (UCC). **

Subject: Operation of the LITR as an Experimental Facility.

By this letter, the Division of Reactor Development gives their approval to continue operation of the LITR as a test reactor in support of the research and reactor programs at ORNL through calendar year 1953. The letter specifies that the operation of the facility will continue under the policy direction and

program coordination of the MTR Policy Board. This letter recommends close surveillance for signs of any deterioration of the reactor tank or components.

27. February 13, 1953, by T. E. Cole, ORNL CF-53-2-115. **

Subject: Proposal for Operation of LITR at Higher Power Levels.

This report discusses the modification that will be required before increasing the power level of the LITR to 3 Mw. The spray nozzles and a supply tank for the spray system, which will be needed to remove the afterheat in the event of a loss-of-coolant accident, are discussed.

Nuclear considerations, cooling requirements, and shielding requirements are discussed for continuous operation at both the 3000-kw and 6000-kw power levels.

28. February 13, 1953, from C. E. Larson (UCC) to Kenneth Kasschau (ORO), ORNL CF-53-2-119 (internal correspondence). *

Subject: Operation of the LITR at Higher Power Levels.

This letter proposes operation of the reactor at the 3000-kw power level and specifies an experimental basis of 3000 kw for one month, suggests "water drop" experiments to verify that loss of the coolant would be of negligible consequences, and emphasizes certain changes to be made as specified in ORNL CF-53-2-115, which is enclosed.

29. August 7, 1953, from Kenneth Kasschau (ORO) to C. E. Larson (UCC), ORO-35046. **

Subject: Operation of the LITR at Higher Power Levels.

This letter gives authorization to operate the LITR through calendar year 1953 at a power level not to exceed 3 Mw with the provision that careful surveillance be kept on the condition of the reactor tank and that Commission approval be obtained prior to making any extensive repairs or replacements. A review with the ACRS will take place on or about November 15, 1953, on the condition of the reactor with particular emphasis on information obtained at the higher power level. This approval is not contingent upon the completion of the water-drop tests. No mention is made in this authorization of the MTR Policy Board.

30. October 1, 1953, from C. E. Larson (UCC) to Kenneth Kasschau (ORO). *

Subject: Justification for the Continued Operation of the LITR. This letter gives information on the corrosion rates of the materials in the reactor vessel and discusses the radiation damage to the neoprene gasket as supporting the plan for continued operation of the Low-Intensity Test Reactor beyond 1953. A list, "Usage of LITR Experimental Facilities," giving many of the major experiment programs utilizing the LITR, is included as justification for planning to use the reactor during 1954.

31. 1953, by W. M. Breazeale, T. E. Cole, and J. A. Cox, ORNL CF-53-5-69 (internal correspondence) or TID-5065.

Subject: Preliminary Boiling Experiment in the LITR. This report describes briefly the initial boiling experiment performed in the LITR. Included is a copy of the strip chart record of the power level as a function of time.

32. November 9, 1953, from C. E. Larson (UCC) to Kenneth Kasschau (ORO), ORNL CF-53-11-58. *

Subject: Operation of the LITR at Higher Power Levels.

This letter, with the attached appendices, gives information requested in the letter of August 7, 1953, from Kenneth Kasschau giving authorization to increase the operating power of the reactor from 1500 kw to 3000 kw. Reference is made to the neoprene gasket between tank sections "D" and "C" (expansion bellows) as probably the weakest link in the reactor containment vessel. A section is devoted to the comparison of the LITR and MTR and another section lists desirable design changes for future reactors of this type. Appendix I discusses design changes made for 3-Mw operation. Appendix II discusses the "LITR Water-Drop Test--2300 kw--August 31, 1951" and is probably the best available report on that specific test. However, the radiation level at the top of the reactor was more thoroughly analyzed in a later report, Water-Loss Test at the Low-Intensity Testing Reactor, ORNL TM-632.

33. February 15, 1954, from Kenneth Kasschau (ORO) to C. E. Larson (UCC).

Subject: Fourth Meeting of the Advisory Committee on Reactor Safeguards.

This is a request that the Laboratory have a representative at the meeting to be held at the Radiation Laboratory at Berkeley, California, on February 22 to 24, 1954, to review ORNL CF-53-11-58, dated November 9, 1953.

34. January 18, 1954, from Kenneth Kasschau (ORO) to C. E. Larson (UCC)

**

Subject: Continued Operation of the LITR.

By this letter, the Division of Reactor Development has given approval to continue operation of the LITR at power levels up to 3 Mw through calendar year 1954 after satisfactory review of ORNL CF-53-11-58 by the ACRS. The AEC requests continued close surveillance and maintenance of records on corrosion and radiation damage to the reactor and components of the primary cooling system.

35. April 23, 1954, from Rogers McCullough (ACRS) to L. R. Hafstad (AEC, ORO-43594).

**

Subject: Operation of LITR at 3000 kw.

This letter summarizes the results of a review of ORNL CF-53-11-58 and other pertinent material by the ACRS. The statement is made, "There is no appreciable public hazard involved in operating the LITR, even at substantially higher power levels than 3000 kw. Because of the long experience and competence of the Oak Ridge National Laboratory staff and management, the Advisory Committee on Reactor Safeguards feels that considerable latitude can be given in the operation of the LITR."

36. July 2, 1954, from H. M. Roth (ORO) to C. E. Larson (UCC), ORO-43932.

**

Subject: Operation of the LITR.

This letter transmits the letter from Rogers McCullough to L. R. Hafstad dated April 23, 1954, regarding the review of the

LITR by the ACRS. This letter requests that an application for continued operation during calendar year 1955 be submitted by November 1, 1954. Authorization to operate the LITR at 3000 kw is implied but is not specifically stated.

37. July 23, 1954, from C. E. Larson (UCC) to H. M. Roth (ORO).

Subject: Operation of LITR.

This letter expresses the appreciation of confidence that the ACRS has in the operation of the LITR as expressed in their comments on LITR operation.

38. September 16, 1954, from C. E. Larson (UCC) to H. M. Roth (ORO). **

Subject: Justification for Continued Operation of the LITR.

This letter requests approval to operate the LITR beyond December 31, 1954. A list of usages of LITR experimental facilities is attached.

39. December 15, 1954, from H. M. Roth (ORO) to C. E. Larson (UCC). **

Subject: Operation of LITR in Calendar Year 1955.

This letter gives approval to operate the LITR at 3000 kw through calendar year 1955.

40. March 11, 1955, by W. M. Breazeale, T. E. Cole, and J. A. Cox.

Subject: Further Boiling Experiments in the LITR.

This report describes the experiment wherein the primary coolant flow pump of the LITR was stopped and the reactor power was raised until boiling took place. This work was performed between June and September, 1952, and reported at a classified meeting on September 10, 1952.

41. July 1, 1955, by M. E. Ramsey and G. D. Cagle, ORNL CF-55-7-158.

Subject: Research Program and Operating Experience on ORNL Reactors (Section on LITR only).

This is a description of the LITR showing photographs taken during the construction as an MTR mock-up for mechanical and hydraulic tests and others taken during the various steps in modifying the facility into a reactor and to higher power operation. A brief description of the experiment facilities and of the contamination control measures is included.

42. October 31, 1955, from J. A. Swartout (UCC) to H. M. Roth (ORO). *
Subject: Justification for Operation of the LITR.

This letter requests approval for continued operation of the LITR. The first mention is made here of remote operation of the reactor from the Graphite Reactor control room. Again, a list of the usage of experiment facilities is attached.

43. December 5, 1955, from J. A. Swartout (UCC) to H. M. Roth (ORO),
ORNL CF-55-12-17 (internal correspondence).

Subject: Recommendations of the Advisory Committee on Reactor Safeguards.

This letter states that most of the limitations on reactor operation are self-originated, either through an ORNL evaluation of safe operation or in anticipation of what the AEC Safety Committee might desire. This letter points out that all ORNL reactor operations are reviewed annually by an internal Reactor Operations Review Committee.

As to the specific recommendation of the ACRS regarding the LITR, the only one made was that beam-hole locks be incorporated into the safety system.

44. January 10, 1956, from H. M. Roth (ORO) to J. A. Swartout (UCC). **
Subject: Operation of LITR.

This letter gives approval for continued operation of the LITR at 3 Mw for an indefinite time, provided such operation meets with the safety standards of ORNL. Any changes to the reactor or its controls shall be documented in accordance with OR Bulletin 8401.

45. July 30, 1956, from J. A. Swartout (UCC) to H. M. Roth (ORO). *
Subject: Remote Control Operation of the LITR.

This letter accompanied copies of ORNL-2121, Remote Control of The LITR, and points out that a remote control test has been in progress for 10 months wherein there has been an operator at the LITR control desk and another at the remote control desk in the Graphite Reactor control room with all controls except startup performed from the remote control desk with the operator at the

LITR control desk to assume control, if necessary. Approval is sought to remove the operator from the LITR control room during routine operation.

46. August 1, 1956, by J. A. Cox, L. C. Oakes, and W. H. Tabor, ORNL-2121. *

Subject: Remote Control of the LITR.

This report discusses the proposed operational procedure and the safety aspects of remote control and gives a history of the remote operation test which has been in operation since October 12, 1955. Included as appendices are: block diagrams of remote instruments and LITR safety system, startup and shutdown procedures, and check lists.

47. August 31, 1956, from H. L. Price (AEC) to E. J. Block (AEC). *

Subject: Review of ORNL-2121, "Remote Control of the LITR."

This memorandum lists questions in the minds of the reviewing personnel which were not answered by the report, ORNL-2121. It is suggested that the information that is given in the report be supplemented by a discussion which will answer the listed questions.

48. October 2, 1956, from H. M. Roth (ORO) to J. A. Swartout (UCC). *

Subject: Remote Control Operation of LITR.

This letter requests clarification of the questions presented in a memorandum from H. L. Price to E. J. Block, dated August 31, 1956. The subject memorandum was an attachment to this letter.

49. October 22, 1956, from J. A. Swartout (UCC) to H. M. Roth (ORO). *

Subject: Remote Control Operation of LITR.

This letter answers the questions in the memorandum from H. L. Price to E. J. Block, dated August 31, 1956, which had been raised by the Reactor Hazards Evaluation Staff when they reviewed ORNL-2121.

50. January 11, 1957, from H. M. Roth (ORO) to J. A. Swartout (UCC). **

Subject: Remote Control Operation of the LITR.

This letter transmits the approval of the Reactor Hazards Evaluation Staff, Division of Civilian Application, to operate the LITR by remote control subject to the conditions:

- a. An operator will have dual responsibility only when both reactors are in the condition of steady operation.
- b. Another experienced operator and a supervisor will always be within call of the intercommunication system.
- c. The operator freed from the LITR control desk will act as a roving inspector.

51. July 20, 1961, from S. R. Sapirie to F. K. Pittman.

Subject: LITR.

In this letter, Sapirie discusses the planned maintenance and upgrading program for the LITR instrumentation in explanation of questions brought out in the 1961 LITR Safeguards Review by the ORO Review Committee. The letter also points out the fact that cancellation of the ANP program has had very little effect on the experimental load in the LITR. This letter recommends continued operation of the LITR, at least through calendar year 1962.

52. April 18, 1962, from J. A. Swartout (UCC) to H. M. Roth (ORO).

Subject: Aluminum-Cadmium Control Rod for the LITR.

This letter discusses the plans for replacing one of the fuel-cadmium control rods with an aluminum-cadmium control rod for the purpose of comparing and evaluating their effectiveness. If the new control rod is equally as effective, the plans are to replace all of the fuel-cadmium rods in the future. The tentative test program is attached.

53. May 7, 1962, from H. M. Roth (ORO) to J. A. Swartout (UCC).

**

Subject: Aluminum-Cadmium Control Rod for the LITR.

This letter states that the Research and Development Division of the Oak Ridge Operations Office of the AEC has no objections to the proposed test but expresses a desire to receive a report on the results of the test with this first rod.

54. June, 1962, by W. L. Smalley, et al, for the RSRC for ORO.

Subject: USAEC-OROO. Safety Review of the Oak Ridge National Laboratory Reactor Facilities Located at Oak Ridge, Tennessee. This report covers the review, by a special committee of primarily USAEC-ORO personnel, of several ORNL reactors, including

the LITR, from the standpoint of administrative procedures, internal reactor safety surveillance, and the individual reactors. A bibliography is included which lists several reports of current application to operational information for the various ORNL reactors.

55. July 19, 1962, from H. M. Roth (ORO) to J. A. Swartout (UCC). *

Subject: Request for Directive CR-336, "Containment, Low-Intensity Testing Reactor, Building 3005."

In this letter, it is requested that an MCA analysis for the LITR be submitted.

56. November 2, 1962, from J. A. Swartout (UCC) to H. M. Roth (ORO).

Subject: ORO Reactor Safety Review Recommendations.

In this letter, several earlier reports which discuss the safety aspects of the LITR are listed. This letter was an attempt to satisfy ORO personnel as to the documentation describing a potential MCA and its consequences.

57. December 28, 1962, from H. M. Roth (ORO) to J. A. Swartout (UCC). *

Subject: Low-Intensity Testing Reactor (LITR) MCA Analysis.

This is a request to supply ORO with a resume of LITR MCA, including expected fuel temperature, extent of meltdown, total energy release, fission product release, and resultant dose rates, as well as a description of possible initiating sequences of events.

58. May 6, 1963, from J. A. Swartout to H. M. Roth.

Subject: Low-Intensity Test Reactor (LITR) MCA Analysis.

This letter describes the LITR MCA, the calculated dose rate, energy release, and maximum fuel element temperature resulting therefrom. The results of the SL-1 accident were used to verify the validity of the method of calculation.

59. April 22, 1964, from J. A. Swartout (UCC) to H. M. Roth (ORO).

Subject: LITR Beam-Hole Lock Switches.

This is a simple proposal to remove the LITR beam-hole lock switches which have been a part of the LITR control system since 1950.

60. May 5, 1964, from H. M. Roth (ORO) to J. A. Swartout (UCC).

Subject: LITR Beam-Hole Lock Switches.

In this letter, ORO asks for supporting documentation justifying that the locks be removed from the safety system and that the modification be reviewed by the Reactor Operations Review Committee.

61. May 22, 1964, from J. A. Swartout (UCC) to H. M. Roth (ORO).

Subject: LITR Beam-Hole Lock Switches.

In this letter, the request for approval to remove the subject switches is cancelled because any approval forthcoming would be too late to permit incorporating the change into the updating of the LITR control system.

62. June 17, 1964, from S. R. Sapirie to C. E. Larson.

Subject: LITR Beam-Hole Lock Switches.

In this letter, ORO justifies their request in the letter of May 5, 1964, from H. M. Roth to J. A. Swartout asking for additional information before approval to remove the beam-hole lock switches. This transfer of correspondence points out the need for supporting documentation and evaluation by an internal review committee before ORNL requests approval of ORO for changes in reactor facilities that were listed as requirements in earlier documentation.

63. August, 1964, by J. A. Cox and C. C. Webster, ORNL TM-632.

Subject: Water-Loss Test at the Low-Intensity Testing Reactor.

This report describes and attempts to analyze the water-loss tests performed in the LITR in 1951, 1952, and 1953. An equation for determining the afterheat generation rate as a function of reactor operating time and time after shutdown is developed in the appendix.

64. September 21, 1964, by C. C. Webster and H. V. Klaus, ORNL CF-64-9-43 (internal correspondence).

Subject: Operating Safety Limits for the Oak Ridge National Laboratory Low-Intensity Testing Reactor (LITR).

This document specifies the limits within which the reactor can be operated safely.

65. February 18, 1965, from A. M. Weinberg to M. Shaw (AEC Washington).

This letter discusses various features of the LITR as well as the other ORNL reactors under the control of the Operations Division and points out the need for continued operation of the LITR after the HFIR achieves full-power operation.

66. October 14, 1966, from A. M. Weinberg to H. M. Roth.

Subject: Annual Reactor Safety Review of the ORR and LITR, Letter from H. M. Roth to A. M. Weinberg, dated September 23, 1966.

This letter states that safety analysis of the LITR and ORR will be prepared as early as possible.

APPENDIX B

OPERATING SAFETY LIMITS FOR THE OAK RIDGE NATIONAL LABORATORY
LOW-INTENSITY TESTING REACTOR (LITR)

INTRODUCTION

It is the intention of this document that a limit be established for each operating variable which has direct reactor safety significance. Each limit designates a realistic boundary to the operating range of the variable; therefore, each limit can be approached with confidence that the safety of the reactor will not be compromised. The LITR Operating Procedures will be prepared so as to provide reasonable assurance that the reactor will be operated within the stated Operating Safety Limits. AEC approval is required for changes to the Operating Safety Limits.

OPERATING LIMITS

- A. Reactor Building--The reactor is housed in an industrial-type building with no other containment features. At the present time, any radioactive gases released into the building must leave the building by the natural diffusion process. Such a process is sufficiently slow to permit evacuation of all the personnel who might be in danger. In addition to the local control and monitoring instrumentation in the building, a duplicate set of those instruments required for power operation is located in a separate building.
- B. Reactor Core
 - 1. Maximum fuel loading--The maximum total mass of fuel in the core is adjusted so that the ganged rods will have to be withdrawn at least 50% of their worth before criticality is achieved for full-power operation.
 - 2. Maximum steady-state power level--3,000 kw nominal (administrative limit).

3. Safety power level scram limit--Set at not greater than 150% of nominal flux-power level (administrative limit).
 4. Maximum heat flux in core (under normal flow)--
1,260,000 Btu hr⁻¹ ft⁻² (ref. 1).
- C. Primary Cooling System
1. Maximum differential pressure across the core--12 psi.
 2. Maximum operating pressure--75 psi.
 3. Maximum outlet water temperature--150°F.
 4. Minimum flow rate for full-power operation--500 gpm (ref. 1).
 5. Maximum flow rate--2,000 gpm.
 6. Maximum activity of coolant water--The radioactivity of the water will be maintained at a level sufficiently low that no excessive exposure to personnel will occur as specified in AEC Manual, Chapter 0524.
- D. Secondary Cooling System
1. Maximum operating pressure--100 psig.
 2. Minimum flow rate for full-power operation--zero gpm.
- E. Emergency Cooling System (for afterheat removal)
1. Minimum emergency cooling capacity and time--No flow if the water remains in the reactor tank; 3 gpm flowing through the core for 60 minutes if the water in the reactor tank is lost.
 2. Minimum emergency power system--No emergency power is required.
- F. Control and Safety Systems
1. Mechanical control system
 - a. Minimum number of control elements--3.
 - b. Minimum number of control rods which must be removed to attain criticality--The reactor will be so arranged that criticality cannot be achieved by complete withdrawal of any one of the control rods while the others are completely inserted.

¹This is below the calculated burnout heat flux using W. R. Gambill's correlation presented in Chemical Engineering Progress Symposium Series 59 (41), 71-87 (1963), but is within the boiling range.

- c. Maximum release time of control elements with or without coolant flow--Two of the three magnets must have a release time of 30 milliseconds or less.²
 - d. Maximum scram time--During regular tests, two of the three control rods must have a time of flight (from fully withdrawn to bottom seat, including release time) with or without coolant flow, of less than 800 milliseconds.² If one rod has a time of flight of 1000 milliseconds or more, adjustments must be made.
 - e. Maximum rate of addition of reactivity by the shim control rods--0.1% $\Delta k/k$ per sec.
 - f. Servo control--The servo control can add positive reactivity at a maximum rate of 1% $\Delta k/k$ per sec but is limited to a maximum amount of approximately 0.5% $\Delta k/k$.
2. Control and safety instrumentation--These instrument channels are so installed as to initiate a reactor scram, a reactor setback, or a reverse action whenever certain predetermined conditions exist. A duplicate set of the instrumentation required for power operation of the reactor is located in a separate building.
- a. Minimum reactor control and safety instrumentation required for startup when the final power will exceed $3 N_L$.
 - (1) Two level safety channels.
 - (2) One log-N period channel.
 - (3) One fission-chamber channel if the neutron background, as indicated by the log-N recorder, is less than 0.001.
 - (4) One ΔT channel.
 - (5) One reactor-water exit temperature channel.
 - (6) One primary coolant flow channel.
 - (7) One radiation monitor that will detect changes in the radiation from the reactor water system.

²The release time and time of flight have been found by experience to be essentially the same with and without coolant flow for this reactor's flow conditions.

- b. Minimum reactor control and safety instrumentation required for startup when the final power will not exceed $3 N_L$ (with raise-test switch in test position).
 - (1) Two level safety channels.
 - (2) One log-N period channel.
 - (3) One fission-chamber channel if the neutron background, as indicated by the log-N recorder, is less than 0.001.
 - c. Minimum safety and control instrumentation required during power operation.
 - (1) Two level safety channels.
 - (2) One ΔT channel.
 - (3) One reactor water exit temperature channel.
 - (4) One primary coolant flow channel.
 - (5) One operating radiation-detection monitor associated with the reactor cooling system.
 - d. Safety instrument checks--An operational check, including release time and time-of-flight determinations, must be made of the safety instrumentation before the start of each operating cycle.
 - e. All startups are performed at the local control room under the surveillance of a qualified supervisor.
- G. Radiation Monitoring Systems--Radiation level monitors. A minimum of three operable radiation monitors will be located at appropriate points within the reactor building.
- H. Experiments--Each in-reactor experiment is subjected to comprehensive reviews and hazards evaluations by the Laboratory's Reactor Experiment Review Committee and/or the Operations Division. In this way an experiment is approved for operation within safety limits applicable only to that specific experiment. Appropriate limits are placed upon any materials, systems, or components that may (for any credible reason) affect the reactor reactivity in such a manner, or to such a degree, that unsafe conditions could result.

With respect to reactivity effects, experiments are considered and approved as follows:

1. An experiment is approved more or less routinely if the maximum change in reactivity that can be caused by the experiment (by any credible means) is conservatively less than the total amount of reactivity controlled by the servo system.
2. Experiments having reactivity worths greater than that in (1) are considered in more detail--in particular, if failure or malfunction of the experiments may cause changes in these values. Considerations are with respect to total worth, rates of change, and to particular situations that may be associated with these changes. No experiment is approved if, for any credible reason, it can cause changes in reactivity that cannot be safely handled by the reactor control system.

I. Administrative and Procedural Safeguards

1. Personnel qualifications--The reactor will only be operated by qualified personnel approved by the Operations Division Superintendent.
2. Minimum staff requirements for reactor operation during any shift.
 - a. One supervisor on duty.
 - b. One operator on duty.
 - c. The control room (local or remote) must be attended by at least one of the persons named previously at all times during operation.
3. Procedures--The reactor is operated in conformance with documented operating procedures. In no instance will the operating procedures designate authorization to operate the reactor in excess of any operating safety limits listed above.

APPENDIX C

THERMAL ANALYSIS OF THE LITR CORE FOR 3-MW OPERATION

C. C. Webster

1. INTRODUCTION

The following thermal analysis of 3-Mw operation of the LITR core is based partially upon calculations using conventional and well-substantiated methods and partially upon direct extrapolation of observations made in the LITR and upon the results of tests performed at the ORR. Because the core configuration and fuel distribution in the two reactors are quite similar, information and knowledge gained in one reactor is usually applicable to the other.

2. POWER DENSITY AND NEUTRON FLUX DISTRIBUTION

The LITR core is composed of a 5 x 9 lattice with 45 core component positions. The internal structure of the reactor tank is arranged so that up to 8 shim-control rods may be used. However, only 3 shim-control rods are normally utilized; the remaining 42 spaces contain either fuel (full or partial) elements, beryllium pieces, or special components fabricated for use by experiments or for other specific purposes. For the last several years, the core loading has consisted of 22 eighteen-plate full fuel elements, 1 fourteen-plate partial fuel element, and 3 shim-control rods, 3 or 4 isotope stringer units, and about 6 full beryllium in-core pieces. The remaining 9 positions are occupied by experiments or special experimental pieces. Figure 1 shows a typical core loading; and, although some changes may be made from time to time, this loading has been utilized for most of the calculations upon which this analysis is based. The fuel content of the element located in a specific core position is not the same at the beginning of each fuel cycle but may vary from one fuel loading to the next and changes during the cycle. As the

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LITR LATTICE ARRANGEMENT ON DECEMBER 2, 1958, SHOWING FUEL LOADING AND POWER OUTPUT

Fig. 1. Typical LITR Core Loading

fuel content of an element decreases, it may be repositioned to maintain control of the flux level at particular locations in the reactor or replaced by a new unirradiated fuel element, by a partially-depleted element, or by any of the special pieces that are capable of being positioned within the core lattice. The fuel content of a full 18-plate element in use in the core may vary from about 135 to 200 g ^{235}U , a 14-plate partial element from about 60 to 140 g ^{235}U , and a shim-rod follower from zero to about 135 g ^{235}U .

The fuel loading is generally rectangular in shape and, at present, is situated on the east side of the core lattice. The fuel elements, which are located one beryllium piece away from the main core, have little effect on the neutron flux profile within the main core region but serve to enhance the flux in the neighborhood of a reactor experiment--particularly if the experiment is a strong neutron absorber. The presence of these isolated elements likewise reduces the perturbation effect of a high cross-section experiment on the main reactor core. During the past several years, the fuel elements in the main part of the core have been arranged at each scheduled refueling operation in such a way that the worth of each of the three shim-control rods is about the same.¹ This results in a power-density profile that does not change significantly from cycle to cycle or throughout a given cycle. The power output and the ^{235}U content for each fuel element, at the time the data used in this analysis was obtained, is given in Figure 1.

On the basis of investigations and experience with the ORR, it is concluded that neutron flux peaking and the power peaking is greatest immediately after startup following a scheduled shutdown. This is due to the fact that with the shim-control rods only about 50% withdrawn most of the power is generated in the lower part of the core. For this reason the calculation for this analysis is based on the neutron flux profile under startup conditions. Figure 2 shows the relative flux distribution for that shim-rod-fuel follower for which calculations indicated the

¹A. L. Colomb and D. Cavin, Fuel Cycles and Loading Programming for Water-Cooled Research Reactors, ORNL TM-277 (July 31, 1962).

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Fig. 2. Relative Axial Neutron Flux Distribution

greatest power output and the two fuel elements with the highest individual power output. The fuel element in core position C-36 has a slightly higher relative flux at its upper end than does the element in position C-25 because only one shim-control rod influences C-36; whereas, two control rods influence C-25. Figure 3 is an axial plot of the macroscopic fission cross section (Σ_f) of the three fuel pieces used in the calculations. The information shown for partially depleted elements is based on earlier measurements made in the ORR.² The information in Figures 2 and 3 is used to determine the axial distribution of the power density for positions C-26, C-25, and C-36, as shown in Figure 4. Note that the power-density profile of C-36 is considerably flatter than that for C-25. This is a characteristic of partially depleted fuel elements because of the concave nature of Σ_f . This feature is significant during a fuel cycle when the ^{235}U content of the fuel element is being depleted.

3. FUEL-PLATE SURFACE TEMPERATURE

The fuel-plate surface temperature for the three core positions used in this analysis was determined from heat fluxes obtained from the power density distribution determined as shown above and from forced-convection heat-transfer coefficients derived from experimental observations made with rectangular cooling channels of similar geometry to that in LITR fuel elements.³ The results are shown in Figure 5 for normal flow conditions with coolant inlet temperature of 100°F and a flow rate through the reactor of 1200 gpm (2.3 fps). The calculated burnout wall temperature,⁴ the coolant saturation temperature, the estimated minimum wall

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

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

⁴L. Bernath, "A Theory of Local Boiling Burnout and Its Application to Existing Data", Chemical Engineering Progress Symposium Session No. 30, 56, pp 95-116 (1960).

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Fig. 3. Axial Distribution of Macroscopic Fission Cross-Section

Fig. 4. Axial Distribution of Power Density

Fig. 5. Axial Temperature Distribution (Normal-Flow)

temperature at the onset of nucleate boiling, and the bulk coolant temperature are shown. The variation of the heat-transfer coefficient, which is dependent upon axial position and wall surface temperature, is also shown.

4. LOW COOLANT FLOW CONDITION

The reactor control is so arranged that the reactor will scram if the reactor coolant flow rate drops below 500 gpm (0.96 fps). Therefore, it is of interest to compute the thermal conditions in the hottest fuel element under the assumption of 500 gpm flow. In examining this case, the fact that other safety circuits including the ΔT and the coolant outlet temperature are set so as to initiate a reactor scram has been ignored. Figure 6 shows the conditions in the fuel element in core position C-25 with a coolant flow rate of 500 gpm (0.96 fps) and a reactor power of 3 Mw. The results indicate that nucleate boiling could be expected in the hottest fuel element but that there is still a considerable safety margin before burnout conditions would be reached. In viewing Figure 6, one should keep in mind that the heat-transfer relationship for forced convection, nonboiling conditions was used in the calculations. Under boiling conditions, the heat-transfer coefficient would be much higher than for nonboiling; so there is a considerable factor of conservatism in the information shown.

5. BOILING EXPERIMENTS

Several boiling tests were performed with the LITR in 1952 wherein the water flow through the reactor vessel was stopped and the only circulation through the reactor was by natural convection.^{5,6} For circulation

⁵W. M. Breazeale, T. E. Cole, and J. A. Cox, Preliminary Boiling Experiments in the LITR, ORNL CF-53-5-69 (internal correspondence), and TID-5065 (1953).

⁶W. M. Breazeale, T. E. Cole, and J. A. Cox, Further Boiling Experiments in the LITR, AECD-3670 (March 11, 1955).

Fig. 6. Axial Temperature Distribution (Low Flow)

to occur in the LITR, it was necessary for the coolant to flow upward through the hotter elements and downward through the cooler elements and other flow passages within the core. The results of the test indicate that under the conditions of no forced circulation through the core, some boiling started at a reactor power of 1.2 Mw. The power level was increased in steps to 1.8 Mw and the instruments indicated considerable fluctuation in the signals from the power monitor chambers. On the approach to the 2-Mw power level, the power fluctuations resulting from collapse and buildup of steam bubbles caused a peak power surge of 3.6 Mw which tripped the reactor scram circuit. With the subcooled coolant, this oscillation in power caused by the collapse and reforming of steam bubbles would be expected. The tests indicate that at a given power level boiling is stable; oscillating conditions exist, but the amplitude of the oscillations does not build up without limit. Figures 7 and 8 are taken from Reference 6.

The results of the boiling experiments indicate that if boiling occurred at the 3-Mw power level the oscillations that would occur because of the subcooled water would give power surges of sufficient magnitude to trip the high-power scram circuit. However, conditions for boiling are prevented by the multitude of scram circuits which are initiated by independent features of the reactor control system.

6. LOSS OF COOLANT TESTS^{7,8}

During the early operation of the LITR (1950-1953), several loss-of-coolant tests were performed whereby the water flow was shut off and then drained rapidly from the reactor. The reactor would then shut itself down from the loss of moderator. During several of the tests (particularly around the 1-Mw power level), one of the shim-control rods inadvertently dropped, causing a premature reactor shutdown. The purpose of

⁷S. E. Beall, An Experimental Determination of Fission Product Heating After Shutdown of the Low-Intensity Testing Reactor, ORNL-1075 (Sept. 25, 1951).

⁸J. A. Cox and C. C. Webster, Water Loss Test at the Low-Intensity Testing Reactor, ORNL TM-632 (Aug. 1964).

Fig. 7. Boiling Test Record

Fig. 8. Temperature and Regulating Rod Position During Boiling Test

these tests was to determine that power level at which it would be advisable to provide an emergency cooling system for the purpose of preventing overheating of the reactor in the event of a sudden total loss of coolant.

The maximum safe operating level for a loss-of-coolant accident without emergency cooling was originally estimated to be 2 Mw (Ref. 7). This was based on data taken from a 0.300-Mw test. A more recent analysis of the data given in ORNL TM-632 (Ref. 8) indicates that if the reactor were operated at a 3-Mw power level for 6 days and then the water was removed rapidly from the reactor, the maximum fuel-plate surface temperature would reach 494°C (921°F) in 93 minutes with no emergency cooling. Since the melting point of aluminum is 600°C (1220°F), it appears that no melting would occur. Figure 9 shows the maximum fuel-plate surface temperature as a function of time after shutdown (by loss of moderator) when the reactor has lost all the coolant through a 6-in. opening at the bottom of the reactor vessel. In this analysis, the shim-control rods are held up and not allowed to drop. Dropping of the control rods would remove fuel from the core region and replace it by a mass of cooler metal that does not have fission-product heat generation. The information shown in Figure 9 is based on an extrapolation to a 3-Mw power level using the results of a 1-Mw loss-of-coolant experiment. In this extrapolation to higher temperature, no credit was taken for the increase in thermal conductivity of aluminum with increase in temperature, improved heat convection (metal to air) with higher metal temperature, or for the increase in heat transfer by radiation with increase in temperature.

As the reactor is now arranged, it is provided with a spray tank capable of supplying a spray of water over the core for more than 2 hours following loss of water from the tank. With the spray in use, the equilibrium point which occurs at 4.5 minutes in Figure 9 (the "no-spray" case) where the fuel-plate surface temperature starts increasing above the boiling point of water would occur at least 2 hours after the water loss; and by then many of the short-lived fission products would be decayed so that the heat-generation rate would be less than half what it is at 4.5 minutes. It can, therefore, be concluded that there is no danger of melting the fuel plates with a loss of coolant, even if the emergency spray system is not working.

Fig. 9. Maximum Fuel Plate Surface Temperature with Loss of Coolant

7. 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 that exists between normal operating conditions and those which could lead to melting of the fuel.

The heat flux to be expected during normal operation can be computed from the power distribution data discussed earlier. The burnout heat flux correlation used for the LITR because of the low coolant flow rate is one developed by W. R. Gambill and presented at a heat transfer symposium held in Houston, Texas, in 1962 (Ref. 9). The correlation has the form:

$$\phi_{bo} = \frac{\pi}{24} L_v \rho_v \left[\frac{\sigma_c^a \rho_l - \rho_v}{\rho_v^2} \right]^{1/4} \times \left[1 + \left(\frac{\rho_l}{\rho_v} \right)^{0.923} \left(\frac{c_p \Delta t_{sub}}{25 L_v} \right) \right] + h_{nb} (t_w - t_b)_{bo}$$

where

$$h_{nb} = 0.105 \left[(Re)_{bz}^{2/3} - 125 \right] (Pr)_{bz}^{1/3} \left[1 + \frac{1}{3} \left(\frac{\eta_e}{z} \right)^{2/3} \right] \left(\frac{\mu_b}{\mu_w} \right)_z^{0.14}$$

(Ref. 10).

ϕ_{bo} = burnout heat flux (Btu hr⁻¹ ft⁻²)

h_{nb} = nonboiling heat-transfer coefficient (Btu hr⁻¹ ft⁻² F⁻¹)

L_v = latent heat of vaporization (Btu lb⁻¹)

σ = surface tension (lb ft⁻¹)

ρ_l = density of liquid (lb ft⁻³)

ρ_v = density of vapor (lb ft⁻³)

⁹W. R. Gambill, "Generalized Prediction of Burnout Heat Flux for Flowing, Subcooled, Wetting Liquid", Chemical Engineering Progress Symposium Series 59(41), 71-87 (1963).

¹⁰W. R. Gambill and R. D. Bundy, HFIR Heat Transfer Studies of Turbulent Water Flow in Thin, Rectangular Channels, ORNL-3079 (June 5, 1961).

- g_c = conversion constant (ft sec⁻²)
 a = local acceleration (ft sec⁻²)
 Δt_{sub} = saturation temperature-bulb cooling temperature (°F)
 t_w = temperature at heated wall (°F)
 t_b = bulk coolant temperature (°F)
 $(\text{Re})_{bz}$ = Reynold's Number of bulk water at axial location
 $(\text{Pr})_{bz}$ = Prandtl Number of bulk water at axial location
 D_e = equivalent diameter (ft)
 z = axial distance (ft)
 μ_b = viscosity of bulk liquid (lb hr⁻¹ ft⁻¹)
 μ_w = viscosity of liquid at wall temperature (lb hr⁻¹ ft⁻¹)
 c_p = specific heat (Btu lb⁻¹ ft⁻¹)

The burnout heat flux was determined for the reference core position, C-25, for 3-Mw operation and normal flow of 1200 gpm (2.3 fps) and for flow conditions of 500 gpm (0.96 fps).

Hot-channel factors were obtained for both new elements and partially depleted elements by using the following factors.

<u>Factor</u>	<u>Element</u>	
	<u>New</u>	<u>Depleted</u>
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
Geometric variation	1.05	1.05
Nonuniformity of flux within element	1.40	1.40
Hot-channel factor	1.73	1.80

The power level corresponding to the burnout heat flux was determined 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 determined by dividing this value

by the hot-channel factor. The burnout ratio was determined by taking the ratio of the equivalent-burnout-power level and the normal-operating-power level. The results in the following table give the values for a new element and a depleted element in core position C-25 under normal flow and low-flow conditions and under conditions of natural convection.

It can be seen that the burnout margin is very large. Even under conditions of natural convection, it is unlikely that melting would occur.

<u>Power (Mw)</u>	<u>Flow</u>	<u>Burnout Heat Flux (Btu ft⁻² hr⁻¹)</u>	<u>Burnout Heat Flux (Btu ft⁻² hr⁻¹)</u>	<u>Burnout Power (Mw)</u>	<u>Burnout Ratio</u>
New Fuel Element					
3	Full	1.81×10^6	0.0526×10^6	59.8	19.9
3	Low	1.53×10^6	0.0526×10^6	50.6	16.9
3	Natural Convection	0.126×10^6	0.0526×10^6	4.2	1.4
Depleted Fuel Element					
3	Full	1.81×10^6	0.0526×10^6	57.4	19.1
3	Low	1.53×10^6	0.0526×10^6	48.5	16.2
3	Natural Convection	0.126×10^6	0.0526×10^6	3.99	1.3

8. CONCLUSIONS

It is clear from the analysis that operation of the LITR at 3 Mw provides a large margin of safety. Recent analysis indicates that the spray tank and two spray heads are not needed to provide emergency cooling in the event of a loss-of-coolant accident.

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