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The attached addendum to the draft copy of Nuclear Safety, Vol. 5, No. 1, consists of the Feature Article and an additional article for Section VI. If you have any comments or criticisms concerning the contents of this addendum, please send copies to Mr. Thomas W. Hazard of DTI, Mr. H. F. Carroll of Declassification, and to me. Inasmuch as the final draft of this issue is scheduled to go to press early in September, your comments will have to be received by August 31 in order to be considered for inclusion in the final publication. Mr. Hazard's and Mr. Carroll's addresses are as follows:

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Yours truly,

Wm B. Cottrell by AWS

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FEATURE ARTICLE

SOME BASIC SAFETY CONSIDERATIONS IN THE DEVELOPMENT OF POWER-PRODUCING REACTORS

F. R. Farmer*

Safety has always been a consideration in the design, construction, and operation of nuclear facilities. In the early stages of reactor development, there were many uncertainties in the interpretation of the nuclear data then available, and understanding of the behavior of materials under prolonged irradiation was limited. The need to guarantee that a significant release of fission products would not take place led to caution in providing for safety and to stringent limitations on operating temperatures and power ratings. In the following years, safety measures of ever-increasing complexity have developed in piece-meal fashion.

Uncertainties still exist, but their extent has been greatly reduced by theoretical studies and operational experience. It is now timely to redefine the fundamental objective of reactor safety and to reappraise the methods by which this objective is to be attained. This paper discusses the range of the more serious accidents possible in large power-producing reactors and the effect the limitations imposed by safety considerations have had on the development of various reactor systems.

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*Mr. Farmer was born in Gloucester, England, in 1914. He studied mathematics and physics at St. John's College, Cambridge, from 1933–1936. After extensive experience in the chemical engineering industry, he joined the Atomic Energy Authority in 1947, where at first he worked on the design of chemical plants at Springfields and Windscale for, particularly, uranium and plutonium processing and the treatment of radioactive effluents. He was closely associated with the early work on special reactor problems, particularly reactor shielding, and for two years he was employed at UKAEA Production Headquarters on the technical planning of production units. Since 1956 he has had responsibility for safety, particularly directed toward reactors and chemical plants, and at present he is Head of the Safeguards Division of the UKAEA Health and Safety Branch, which is located at Risley.

Reactor safety provisions are then reviewed with the objective of seeking simplification of the present complex systems.

General Accident Considerations

In the normal operation of a reactor, the heat generated by the fuel is balanced by the heat removed by the coolant. This situation might be changed as a result of

1. The addition of reactivity through movement of fuel or absorber or a change in neutron moderation,

2. A reduction or loss of coolant flow either throughout the core or locally to some region,

3. A failure of the primary pressure circuit of any reactor using a pressurized coolant, whether this is gas, water, or steam.

In a large power reactor, there should be no need for an excessive rate of reactivity change to result from the movement of an absorber, whether used for control purposes or for flux flattening. The reactivity changes resulting from the addition of fuel can also be made quite small by appropriate design of the fuel-charging mechanism, suitable choice of the size of fuel components, and precise regulation of the charging procedure. In some reactors the effective density of the moderator can be altered suddenly, and this can lead to a large and rapid change in reactivity. This type of accident cannot occur in a gas-cooled reactor employing a large bulk graphite moderator, but in many reactors in which water is used both as coolant and moderator such changes can occur either through the introduction of cold water or through the collapse of voids. Reactivity changes that result in reactor periods in the range of a few

seconds to a few milliseconds have been investigated extensively in BORAX and SPERT experiments in the United States, and many theoretical studies have been made of the effects of temperature coefficients of reactivity in order to develop reactors in which any harmful effects of such changes can be minimized and to provide time for the operation of protective systems.

The early concern with the reactivity behavior of water-cooled and -moderated reactors has been long sustained, and even further study may be required for the development of new water reactor systems or for the design of large experimental and test reactors of the water-cooled and -moderated type. There are important safety problems involved, and the reactivity behavior remains a technical problem in achieving effective and economic control. In the author's opinion, however, there is a tendency to overemphasize reactivity accidents in well-proven systems for which the reactivity behavior has been thoroughly investigated and for which control systems of adequate reliability and diversity have been provided. This is true for the presently operating, large, gas-cooled reactors, as well as for the water-cooled and -moderated reactors.

The loss of cooling because of a loss of power to circulators or pumps is a possible, or even likely, event at some time during the life of a reactor and hence demands serious consideration in design. Complete loss of cooling owing to local blockage may be impossible, but partial failure has already occurred in a number of reactors. Such failures can occur because of swelling or other damage to the fuel, the fuel can, or the moderator, whether the result of irradiation, chemical effects, or mishandling. The more severe loss of cooling that results from

depressurization is much less likely and, indeed, should not occur. The consequences can be so severe, however, that it is not prudent to forego the serious consideration this type of accident receives.

By considering various types of reactor, this paper develops the theme that "loss-of-cooling" accidents give rise to the major safety problems in large power reactors. It is in this area that further safety studies should be concentrated. In parallel with this theme, it is also argued that for well-established reactor systems the basic safety measures required are few and simple if separated from other considerations of convenience or plant availability.

Gas-Cooled Reactors

Gas-cooled reactors are discussed here in terms of those systems used or planned for use in the United Kingdom power program. These reactors are of two types: (1) the so-called "Magnox" reactors, which use natural uranium metal fuel in Magnox cans, and (2) the Advanced Gas Cooled Reactor (AGR), which uses low-enrichment UO_2 fuel in stainless steel cans.

Most reactor systems have temperature coefficients, by design, that provide some degree of self-stabilization in most accident conditions. In graphite-moderated reactors using low-enrichment fuel, a significant prompt negative effect is due to the Doppler coefficient associated with the fuel temperature. The moderator temperature coefficient is also initially negative but could become slightly positive at maximum fuel irradiation. The effect of the moderator temperature coefficient is always less, however, than that of the fast-acting fuel temperature coefficient. Most designers use the inherent temperature effects on reactivity to make

the system initially self-correcting for any change in reactivity or coolant flow. There are some accidents, however, for which the self-correction is inadequate or not sustained, and, for these, sensory equipment is provided to ensure the insertion of neutron absorbers to control and, if necessary, shut down the reactor.

The evolution of control and safety systems has resulted, for example, in the later Magnox reactors, in a highly complex protective system that requires particular attention to detail at all stages of conception, fabrication, safety evaluation, commissioning, and maintenance. Whether or not such complexity is really necessary is questioned in the following paragraphs.

Reactivity Accidents. The carbon dioxide coolant controls a negligible amount of reactivity, so any significant change in reactivity is related to unrestricted withdrawal of control rods or gross error in fuel or absorber loading. After initial commissioning, a power reactor is normally expected to be operating at either low or full power. Exceptions to this could occur during routine maintenance or infrequent and prolonged periods when power is not required. Complete shutdown (and, hence, restart) should be infrequent. In these circumstances a slow startup is economically acceptable, and the rate at which the control system can add reactivity can be suitably limited. Thus in both Magnox reactors and the AGR, unrestricted rod withdrawal at power causes an average rate of rise of the maximum fuel can temperature of only a few degrees per second. This allows plenty of time for the operator to take action to terminate an incident.

In the case of unrestricted rod withdrawal at startup, the larger gas-cooled reactors present some uncertainties which should soon be resolved. These uncertainties are associated with the neutron flux distribution in the core as affected by the position of the control rods, by variations in the flow of coolant, and by radial or axial asymmetry caused by temperature and xenon effects. It is expected that further experience will lead to operating conditions in which the reactor could, by sensing the fuel element temperature, be safely tripped in the event of unrestricted rod withdrawal during startup.

An accidental divergence that occurred when the reactor was nominally shut down would, of necessity, be limited in magnitude because of the small amount of reactivity associated with fuel elements and absorbers. In general, the operator would have ample time to insert additional shutdown rods during fuel handling or maintenance operations. Fuel temperature could also actuate additional shutdown rods to prevent fuel damage. The present United Kingdom practice for gas-cooled reactors is to withdraw some control rods after the reactor is shut down and hold these "cocked" and ready for reinsertion. This can be readily achieved and obviates the difficulty of substantiating the shutdown condition of the reactor by measurement or administrative procedures.

In the preceding paragraph and later in this paper some credit is attached to the presence of an operator who may, and frequently will, shut down the reactor before a dangerous state is reached. No formalized rules have been laid down governing the time interval or reliability but, insofar as instrument systems gain in reliability through diversity, similarly there is a gain through the diverse function of an operator

that is related to the simplicity and clarity of information supplied to him, the course of action to be followed, and the time interval available.

Thus reactivity accidents are of comparatively minor importance and, even in extreme cases of this type of accident, operator action or a rod trip based on the fuel element temperature ensures the safety of the fuel. Nevertheless, despite the above arguments, a typical protective system for a Magnox reactor provides for rod trips based on the faults listed below:

Fault	Parameter Revealing Fault
Unrestricted control rod or fine control (regulating) rod withdrawal at power	High flux High fuel element can temperature High primary system pressure
Unrestricted control rod withdrawal at startup	High flux Short period High fuel element can temperature High primary system pressure
Accidental divergence when nominally shut down	High flux Short period High fuel element can temperature High primary system pressure

Loss-of-Flow Accidents. The complete loss of coolant flow in the whole or in part of any power reactor must inevitably lead to fuel melting if the reactor remains at full power. In Magnox reactors the temperature rise of the fuel in regions of maximum flux would be 10 to 15°C per second and, in the AGR, 40°C per second. Total loss of flow cannot arise through mechanical blockage, and, even in local regions, complete blockage is judged to be impossible in view of the large free-channel area necessarily provided for gas cooling. For Magnox reactors, if single channels suffered flow restrictions of approximately 50% of the free area, the danger temperature would not normally be reached, and, for the AGR, a reduction in flow area of 75% could be tolerated.

It is important to show by a sensible theoretical analysis with reasonable experimental support that isolated trouble spots in a reactor would not spread progressively throughout the whole reactor. It can be shown that channels surrounding any faulty channel can remove at least 25% additional heat and, as there is no chemical reaction which can abnormally boost nuclear heat production, no route is foreseen whereby isolated faults can propagate through the reactor. In addition the spread of contamination from faults arising in isolation should be prevented by early detection through the burst-cartridge detection gear before the can reaches its melting point of 1460°C.

For the Magnox reactors, it was necessary to consider the effects of the fuel can burning in an isolated channel. Therefore a series of experiments was carried out in which canned fuel of the Calder type was ignited in an apparatus containing a full-scale channel of graphite. These experiments showed that uranium would burn slowly over a period of hours and that the combustion of Magnox would provide the major source of heat. Thermocouple measurements made in the graphite confirmed the theoretical predictions for heat flow through the graphite and showed that the average flow of heat to neighboring channels was approximately 10% of the heat normally removed at full power. The flow of heat persisted for a limited period only, and during this period the graphite served as a useful heat sink. The experiments and theoretical studies all led to the conclusion that the additional heat passing to neighboring fuel channels from a single channel of burning Magnox could be removed by preserving a small coolant flow when the reactor was shut down.

In further experiments the possible penetration of steel plates by the molten uranium from the burning channel was studied. These experiments confirmed that with existing designs no penetration of the pressure vessel would occur.

Thus both the Magnox reactors and the AGR are designed so that isolated faults do not propagate, and therefore the consequence of the fault cannot be of major safety significance. This is due to the high thermal heat capacity available in the bulk graphite moderator and to the sensitive burst-cartridge detection system, which would give rapid indication of any single, severe local fault.

In considering flow failures that affect the reactor as a whole, it is evident that minor variations in flow or the loss of a single blower would have little safety significance and that the only cases requiring serious study are closure of the main gate valves and circulator power failure. The operation of the main gate valves on the gas circuits must be arranged so that accidental closure of all valves at the same time, or rapidly in sequence, is positively prevented. A better solution is to dispense with valves entirely, as in the integrated layout used in some designs for reactors with prestressed concrete pressure vessels. It is claimed that such designs alter fundamentally the time-scale or magnitude of the depressurization accident. This has an implication in consideration given to the depressurization accident later in this paper.

If the power to all main blowers failed, the coolant flow would decrease at a rate determined by the inertia of the blower and any residual energy in the blower drive. The initial rapid rise in the fuel temperature would lead to a reduction in the reactor power. Consequently the

subsequent temperature rise would be slow and might be reversed. A test was carried out at Calder Hall in which all four blowers were stopped when the reactor was operating at a steady power of 95 Mw. The control rods were held stationary. The fuel can temperature increased initially at 2°C per second, reached a maximum after 1 1/2 min, and thereafter decreased. The particular value of this test lay in the confirmation of theoretical predictions of the stability of the reactor and of the length of time available for shutoff rods to operate. In the AGR, the use of stainless steel cans increases the temperature margin available, but this is partly offset by the use of poorly conducting oxide fuel, which stores a larger amount of heat than metal fuel. For all presently operating gas-cooled reactors, however, safe conditions can be ensured if reactor shutdown is based on signals taken from fuel temperature. Fuel can temperatures are measured by thermocouples at many points reasonably well distributed across the reactor. These measurements are related statistically to flux levels and anticipated flux changes, to known variations in coolant flow, to local core loading, to control rod positions, and to other variables. These measurements are used to operate the safety circuits through a temperature margin trip coupled in variants of the standard two-out-of-three method. In addition, the operator has an adequate margin of time to shut down the reactor manually. A typical protective system, in contrast, provides for rod trips based on the following fault indications:

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Circulator power failure	Opening of circuit breakers High fuel element can temperature High primary system pressure

Failure of a single circulator	High differential flow High fuel element can temperature High primary system pressure

Depressurization Accident. A more serious loss of cooling could arise through the accidental depressurization of the primary coolant circuit. Various rates of depressurization have been examined, and it has been shown that under some conditions the coolant flow through the core could temporarily reverse, and, under other conditions, after an initial flow oscillation, the flow through the core could be virtually zero, that is, temporary stagnation could occur. The effect of this accident has constituted a major safety study for the gas-cooled reactor, and many experimental and theoretical investigations have been carried out. The investigations have fallen naturally into two distinct parts; first, the gas dynamics of depressurization and their consequent physical effects and, second, the transient thermal behavior of the core during and following the event. The first study was a check of the transient gas pressures within the pressure vessel and within the biological shield. This check was made both theoretically and experimentally using a 1/50-scale model of a Calder Hall reactor. The transient pressure within the particular vented biological shield of that reactor was found to be well below the steady pressure at which the shield would fail. Consideration was then given to the transient forces on the core components to ensure that they would not dislodge or damage the control rods, the fuel elements, and the graphite structure. As part of this investigation the pressure changes throughout the reactor and its coolant circuits were calculated using a large computer program of some complexity and then checked against the evidence deduced from a 1/18-scale model that was a somewhat simplified

representation of the reactor system. The confirmation of theory by tests of models of one particular design enables similar analyses to be applied with confidence to other designs. The behavior of gas flow through the core during depressurization could be predicted and the design arranged to ensure that geometric stability was maintained.

These studies showed, as might be surmised from simple considerations, that following a failure of the lower or inlet duct within a narrowly defined range of size and location, cessation of flow could persist for a period of about 30 sec. This is roughly the period during which the gas discharges at sonic velocity. Since an initial period of flow stagnation has a serious and adverse effect on the fuel-can temperature transient, the postulated failure of the inlet duct has been taken as the standard case for the study of the depressurization accident. This adds further pessimism to an already unlikely event in that it presupposes that the type, size, and location of the failure combine to give the worst conditions. Understandably the study of fuel temperature behavior in this accident is extremely complex and is carried out using various types of refined computer programs. Considerable effort has been devoted to this work.

A reasonable understanding of the course of this accident can be obtained, however, without recourse to a computer, by the use of the following simple, hypothetical model. The model describes the situation in which fuel producing heat at the normal power level suddenly loses convective cooling. Initially the fuel can temperature is about 500°C, and, since the melting point of Magnox is 645°C, the effective margin, making some allowance for errors and uncertainties, is in the neighborhood

of 130°C. During normal operation, there is a parabolic distribution of temperature in the fuel, ranging from 500°C at the outer surface to a central temperature of about 600°C. On loss of convective heat transfer from the fuel can, the heat stored in the fuel is shared with the can, and the temperature of the hottest part of the can may rise 50°C. If there is a delay of 1 sec in shutting down the reactor, all temperatures will increase between 10 and 15°C, say 12.5°C. This leaves a margin of 67.5°C. This may be regarded as a margin for the fuel to accept additional heat and is equivalent to running the reactor at full power for 5 sec (i.e., 5 full-power seconds). The reactor will be shut down in about 1 sec, and heat production will decrease rapidly according to the amount of negative reactivity added. The values of shutdown heating current used in the United Kingdom for the times corresponding to these accident studies are about 40% higher than those quoted by Stehn and Clancy.[1] For a reactor shut down by 2% k, the available fuel heat capacity of 5 full-power seconds is equivalent to the first 35 sec of shutdown heating; this includes the heat from fission products and from decay neutrons. This time is considerably extended by the additional heat capacity of the graphite moderator, which can absorb heat radiated by the fuel. Owing to the extended surface of the fuel can, about 5% of full power can be radiated, so the balance soon after 100 sec is as follows:

Heat capacity of fuel	5 full-power seconds
Heat radiated from fuel	~5 full-power seconds
Margin available	10 full-power seconds

For a reactor shut down by 2% δk the integrated shutdown heat for 100 sec is just less than 10 full-power seconds. This figure is reduced to 7.6 full-power seconds if the shutdown reactivity is 4% δk.

At about 100 sec the heat radiated from the fuel is equal to the shutdown heat production, which then continues to decrease. This shows in a simple analysis that safe conditions can be maintained, even if heat is not removed from the fuel by convective cooling. After the initial period of flow stagnation, it is likely that a number of main blowers will be in operation and will re-establish considerable flow of the carbon dioxide coolant. Even if all blower motors are tripped, however, some degree of coolant flow is set up by the residual inertia in the blowers, and this is followed by a small amount of thermal circulation. Heat removed by this means, together with that by direct radiation from the fuel, is adequate in the early stages of this accident to limit the temperature rise of the fuel. This conclusion is verified for each reactor by more refined calculations.

One conclusion which emerges is that the reactor must be shut down within 1 to 2 sec after the accident. This shutdown may be initiated by an instrument sensing the rate of change in pressure that can easily and reliably be made to operate within the time available. Such instruments already fitted to gas-cooled reactors will transmit a signal within 1 sec after any severe pressure reduction.

For the AGR, a simple analysis of the initial stages of a depressurization accident follows the same pattern as that for a Magnox reactor. For fuel pins of the diameter and rating used in the Windscale AGR and with helium-filled stainless steel cans, the maximum can temperature is typically 700°C, the maximum fuel temperature is 1200°C, and the equalization temperature is 950°C. The available temperature margin depends on the failure temperature of the cans. A limiting condition certainly exists

if the cans approach their melting point of 1460°C. It is likely that failures will occur at some lower temperature, however, because of the pressure built up inside the can by the accumulation of volatile and gaseous fission products. The limiting temperature will be dependent upon the precise details of fuel element design and irradiation history. Laboratory tests with typical cans have shown that the limiting temperature is unlikely to be higher than 1200°C and it could be lower. Based on a value of 1200°C, there is a margin of 6 full-power seconds. The subsequent analysis follows almost exactly the same pattern as that for a Magnox reactor, except that the fuel elements could radiate about 7% of full power heat. It follows that in a time of less than 100 sec, the heat radiated from the fuel is equal to the shutdown heat production, and thereafter temperatures would decrease.

Any gas-cooled system using thicker fuel pins or higher ratings would give higher equalization temperatures and reduced margins. In these cases the simple arguments presented above would need to be refined, especially those concerning the assumptions made about stagnation. In some designs the period of stagnation will be limited, if it occurs at all, and in addition a significant amount of heat might be removed from the fuel before the onset of stagnation. These arguments would be subject to critical and independent evaluation, and their acceptance would still impose such limitations as might be necessary to ensure that a pressure trip would act in sufficient time to prevent extensive failure of the fuel.

The discussion so far has been concerned with the heat balance during the first minute or so after the accident; however, an exothermic reaction can take place between graphite and air, with the rate of reaction being

a function of the temperature and the chemical reactivity of the graphite. This reaction only takes place if air enters the reactor, as, obviously, if air does not enter the reactor no exothermic reaction can take place. If air does enter, possible solutions lie either in the removal of the air or in the removal of heat at an adequate rate to prevent a "runaway" air and graphite reaction. In a Magnox system, this would occur only if the main gas valves failed to isolate the faulty circuit or if the fault occurred inboard of the valves. In the AGR the smaller core lends itself to a design in which the reactor and heat exchangers are assembled compactly inside an integral biological shield. A failure in the primary circuit would fill the enclosed volume with carbon dioxide and ingress of air would be severly restricted.

In either system it might be argued that even if air does enter the reactor, the provision of sufficient coolant flow at all times could ensure a continued reduction of fuel and graphite temperatures. To this end, some reactors have power supplied from two or three grid connections and auxiliary pony motors connected to batteries of 20-min capacity backed by three diesel generators. In the event of this major accident, it is difficult to guarantee that even these complicated arrangements will provide power at all times. Furthermore, dependence on coolant flow alone presupposes that a runaway graphite and air reaction cannot develop in any part of the core at any time in the life of the reactor. This requires greater confidence than exists at present in the application and extrapolation to the whole core of the relatively brief experience with graphite reactivity changes.

It would be more reasonable to assume a limited period of power disconnection followed by reinstatement of the main or auxiliary energy sources. The maintenance of safe conditions would then require the injection of carbon dioxide into the primary circuit to exclude or rapidly expel air from the core. This CO_2 could be stored on site as a liquid or solid, but this would require a guaranteed heat source for its evaporation. It would be simpler to obtain the CO_2 from other reactors on the same site by direct interconnection. This would dispense with the need for a heat source and would carry a guarantee that a large reserve of gas would be continuously available.

The limits currently accepted on the fuel heat rating and the fuel and graphite temperatures ensure that for most accident conditions a sufficiently long time exists for remedial measures to be taken and that these measures can be simple and reliable. The further extension of these limitations will depend on significant changes in design, such as the use of a concrete pressure vessel. As mentioned earlier in this paper, the development of concrete pressure vessels should eliminate the possibility of a rapid depressurization and hence remove the restrictions imposed on the reactor by the current convention of the hypothetical "burst-duct" accident.

Fast Reactor

With so little experience available, it would be presumptive to present in this paper an analysis purporting to cover all fast reactor systems. Some general points have emerged, however, from the studies already made in connection with the United Kingdom program.

Reactivity Accidents. For reactivity changes that are limited to a fraction of a dollar, the dynamic behavior of the reactor is governed by the characteristics of the delayed neutron fraction, just as in thermal reactors. There is a significant difference in behavior, however, when the excess reactivity approaches one dollar in that the reactor period falls sharply. The "safe" reactivity step is therefore determined by the prompt negative Doppler effect from the fuel but is unlikely to exceed significantly a value of one dollar.

Major reactivity accidents could occur as a result of undesirable movement of fuel or absorber or the addition of moderator. In the United Kingdom, movement of absorber is the presently preferred method of starting up and controlling a fast reactor. The particular method chosen will be a suitable compromise between the safety requirements and the need to operate the reactor in an economic way. There is no doubt that a suitable protective system can be designed to prevent damage from credible malfunctions of the control and safety systems with due regard for the range of temperature coefficients likely to be accepted by reactor designers. The design will ensure that some malfunctions are impossible, and a reactor trip based on coolant temperature will deal with other faults.

Spent fuel is discharged directly or passed through intermediate positions in a shuffling program. It is possible that the engineering difficulties of on-load charging may lead to a decision to change fuel with the reactor shut down. In the United Kingdom, it is not currently proposed to have shutdown reactivity poised during fueling operations. Safety will be ensured if no movable element controls more than the safe reactivity step. Where this is impossible, a suitable guarantee is

required that the reactor is shut down by an adequate margin. This guarantee is provided by suitable measurements of the shutdown reactivity, coupled with an accurate record of reactivity changes in the system.

The hazards produced by the addition of moderator to a fast reactor are well recognized. There is little doubt that measures can be provided to eliminate the possibility of the addition of hazardous amounts of moderator. Past operational experience serves to confirm this view.

Loss-of-Flow Accidents. An attempt has been made in some designs to prevent the complete loss of coolant flow by the use of multiple circuits and by the diversification of power supplies. For example, in the Dounreay fast reactor there are 24 primary coolant circuits and 24 primary coolant pumps powered in pairs by 12 diesel generators. It is doubtful whether complications of this type will find general favor. It is commonly accepted, however, that a loss of power to the pumps is possible.

In the sodium-cooled system the important effect of a loss of flow is that the total amount of heat removed from the reactor decreases and the bulk sodium temperature increases. If bulk boiling occurs, both coolant flow and rate of heat transfer decrease drastically and give conditions locally that are almost equivalent to a complete loss of cooling. The design must ensure that bulk boiling will not occur at any time during a loss-of-flow accident. With some values of reactivity coefficients, designing the circuit for a high degree of natural circulation and designing the pumps to have suitable rundown characteristics will ensure that a reactor scram is not required and that the reactor power will decrease safely to a value appropriate to the residual flow. These inherent

safety features may not be adequately proven or economically provided in early fast reactor designs. Safety will then depend on a reactor trip based on coolant temperature or the provision of auxiliary means for maintaining adequate flow to remove heat in step with decreasing neutron power. Both of these features may be included in some designs.

A further fault condition to be considered is the loss of cooling in a limited region of the reactor, such as a single fuel assembly. Severe damage would be expected in the region directly affected, but this would not be a major hazard if the condition did not propagate throughout the whole core. This has been recognized as a major problem in fast reactors, and an extensive investigation of this situation is under way. It is recognized that damage could spread through thermal effects, through mechanical distortion of adjacent core members owing to pressures generated by sodium boiling, or through circulation of debris by the coolant. There is reasonable assurance that melting induced by local blockage would not propagate throughout the core provided the reactor was shut down rapidly. The problem is then one of identifying the defect sufficiently quickly, as, for example, by the detection of either sodium boiling, variations in flow or outlet temperature in every subassembly, or fission products in the bulk coolant.

Depressurization Accident. A sodium-cooled reactor with oxide fuel dispersed in a stainless steel matrix and canned in stainless steel was taken as an example for discussing the depressurization accident. In normal operation the averaged temperature of a fuel element at the core center may be about 600°C, and the failure temperature for this type of fuel cannot be higher than the melting point of stainless steel, 1430°C.

It is easy to show that a reactor of this type cannot survive a complete loss of coolant. For a reactor rating at 200 Mw per ton, the complete loss of coolant would lead to a fuel temperature rise of 500°C per second at full power and a rise of over 700°C within the first 10 sec after shutdown. Sodium-cooled reactors operate at a low pressure, however, and use the coolant at temperatures well below the boiling point. Although reactor designers face some new problems in vessel designs for high temperatures and temperature gradients, the vessels are not at the same time required to retain high pressures. Designs in which the reactor and heat exchangers are housed in close-fitting compartments or are suitably double skinned should eliminate the possibility of exposing the core through loss of coolant. It may well be that the safety of fast reactors and the limitation on their rating will depend on the development of suitable sensors to give rapid and clear indications of local faults.

Water-Cooled and -Moderated Reactors

In order to make some general comparisons with the systems already discussed, a water reactor of the pressurized or boiling type that uses UO_2 fuel and stainless steel cans is taken as an example.

Reactivity Accidents. In comparison with gas-cooled reactors, control rod movement in a water reactor gives much larger rates of reactivity change and much larger amounts of reactivity are controlled by the coolant-moderator. Hence considerable attention is paid to reactivity accidents. The protective system by a combination of preventive measures (such as those used to prevent some severe cold water accidents) and fast-acting reactor shutdown from a variety of sensors is designed to ensure safety.

Reactivity accidents are not further reviewed in this paper because the author considered that the loss of cooling remains the major outstanding problem in large power-producing reactors.

Loss-of-Flow Accidents. In pressurized-water reactors the loss-of-flow accident could arise because of loss of power to the pumps. The limiting safe condition during the transient could be due to a condition of flow instability that might arise if the coolant temperature reached the boiling point or, alternatively, could be due to a local burnout condition. Even with pessimistic assumptions used in the study of this accident, it can usually be demonstrated that fuel melting will not occur even at hot spots because the reactor is shut down rapidly by its inherent characteristics (negative temperature or void coefficient) assisted by the usual shutdown devices, which could sense the power-to-flow ratio or the coolant temperature.

Depressurization Accident. Conditions that follow the severe depressurization of the primary circuit are more difficult to analyze. As in gas-cooled reactors, the immediate effect within the core is such that there is a real possibility that, during some period of time, cooling will cease completely. This is principally due to the production of voids in the core, which could lead to steam blanketing of the fuel and hence some redistribution of its stored heat. For the purpose of this discussion some typical fuel temperatures are required and these are given in Table 1. The values quoted in this table are not related to any specific reactor design. They obviously depend upon the state of the oxide during irradiation and the variation of thermal conductivity with temperature. The values quoted have been deduced from experimental evidence obtained at Chalk River.[2]

Table 1. Fuel Temperatures in Water Reactors for Various Rates of Heat Production

Heat Production Rate (kw/ft)	Fuel Surface Temperature (°C)	Fuel Central Temperature (°C)	"Equalization" Temperature of Fuel and Can (°C)
3	360	630	460
4	380	750	520
8	450	1220	750
14	550	1900	1100

The "average" fuel in the reactor would probably have a rating of between 3 and 4 kw/ft and a rate of temperature rise at full power with no cooling of between 40 and 50°C per second. There is thus a margin of at least 15 full-power seconds before a dangerous temperature is reached. This simplified analysis for average fuel is not representative of any particular channel; it simply explores the temperature behavior of fuel operating at an average rating. There is in the reactor a large amount of fuel at this condition or at even lower ratings, and it is obvious from the data of Table 1 that there is a very large margin in time and heat capacity for fuel in this condition before a dangerous situation is reached. In fact, since the reactor must necessarily shut down within the first second or so, the heat capacity in the fuel is adequate for many hundreds of seconds for heat production at the shutdown heat rate.

An examination of the temperature pattern that would exist in the more highly rated parts of the core reveals, however, a number of uncertainties that are as yet unresolved. There are large variations in local heat ratings, and it is possible to find hot spot ratings equivalent to 14 kw/ft of channel or even higher. As Table 1 shows, the equalization

temperature for a heat rating of 14 kw/ft is 1100°C; the corresponding rate of temperature rise at full power with no cooling could be about 180°C per second. Thus approximately 5 sec after the accident the can temperature would be about 1300°C and the continuing effect of shutdown heating would cause can melting after a further 5 to 10 seconds. It is thus important to consider the conditions of heat removal from the reactor in the early stages of depressurization. The USAEC has announced[3] the intention of investigating these conditions in a major research program during the next few years as additional support to an already extensive program of basic work on heat transfer. If it is shown that there would be no effective cooling, local melting would occur in regions of high heat rating unless shutdown heat-removal systems were effective a short time after the accident. Such local melting might not be of major safety significance if it did not propagate throughout the core. This would depend on the materials used for fuel and cladding and the geometrical arrangement of the core. Experience with small accidents has already shown that limited melting of cladding need not be catastrophic.

It appears therefore that a shutdown heat-removal system could be required in a short time. The main contrast with gas-cooled reactors is that it is necessary to provide a heat sink. This might be achieved by the injection of water and its subsequent evaporation; however, there are problems associated with this and with other possible methods.

At the present time it is not possible to guarantee to an acceptable standard that the fuel cans will remain intact. Thus it is considered necessary to provide large water reactors with an additional boundary to prevent the escape of fission products, that is, a containment vessel.

Future development could proceed along lines in which local areas of high heat rating are reduced or shown to have lesser importance than currently believed. The planned experiments may show that there is an effective heat transfer mechanism during some stages of blowdown. Suitable engineering solutions may be found to the problems associated with the removal of shutdown heat, especially in those designs in which subdivision of circuits and the use of large feed pumps have reduced the magnitude of the depressurization accident. Thus the need for conventional containment may eventually disappear.

Containment

In any discussion of the safety of reactor systems, it is customary to consider the role of containment. In its original sense, the word "containment" itself is mesmeric and implies attainment of the objective, which is to provide a barrier adequately strong to retain any accidental release of energy from the primary circuit and reactor and sufficiently gas tight to ensure a minimum tolerable release of fission products. The principle has much to commend it, but an examination of many forms of containment will serve to show the extent to which the stated objective is attained. The initial concept may be of a continuous all-welded steel membrane. This can be tested to a pressure that is related to an accident study for the reactor. In prototype reactors it may well happen that construction of the containment vessel is well advanced before the reactor design and its fault studies are completed. This may result in a paring down of design margins or, alternatively, in the use of less pessimistic assumptions in the accident anlaysis. An additional problem to be considered is the possibility that high-velocity missiles may be ejected

from the reactor that could penetrate the containment vessel. The designer may choose to strengthen the container with concrete or by the use of energy-absorbing materials or to prevent missile ejection by locally securing the object or by interposing shield walls. Many designs reach the stage of an uneasy truce between missile pitchers and catchers.

The design idea of using a continuous membrane soon needs to be modified. The reactor must be serviced and openings are needed. In a number of reactors, operators are not needed inside the containment vessel during operation, and here the access openings may be rewelded or made good by a reliable compressive joint. Where personnel access to the containment vessel during operation is required, several double air-lock doors must be provided. Additionally, goods access is required, and double doors are provided that may or may not be opened during power operation.

Other penetrations exist because all reactors require services that must be carried through the containment membrane. The number of these penetrations can be large, as shown by the following examples:[4]

Enrico Fermi	500 penetrations
Piqua	78 penetrations for conduits 66 penetrations for pipes
Dragon	300 penetrations for pipes 1000 penetrations for single-wire cables 300 penetrations for multiple-wire cables
AGR	880 penetrations for cables 140 penetrations for pipes

An attempt was made on the Dounreay fast reactor, where personnel access to the containment vessel was required, to maintain acceptable working conditions by using a self-contained air cleaning system. This proved to be difficult, and the reactor is now fitted with the normal arrangement of a fresh-air ventilating system and isolating valves that will

seal the containment vessel on demand. Thus, if the containment structure is to provide the protection required after an accident, a number of valves must operate and seat perfectly on piped services and on the ventilation plant.

A review of United States practice in the initiation of valve closure to isolate a containment building has shown wide variation in standards. There are systems that use ten different parameters and others only one. In almost all cases a single electrical or pneumatic signal is sent to all valves, and failure of the signaling device or of its energy source results either in automatic closing of all valves or in inability of any of the valves to close.[5] In some situations, if a large valve fails to close, the leak rate may be very high and the containment structure virtually useless. If all valves operate as designed, significant leakage can still occur past a poor seating or a faulty door seal. The mass flow through a hole of given size will depend, of course, on the composition of the gas (air, CO_2, steam, etc.) and the pressure difference across the hole, and the percentage leak rate will depend on the size of the containment structure. For a hole 0.25 in. in diameter, the leak rate could vary between 0.5 and 5.0% per hour, which is to be compared with the leak rate specification that is usually between 0.1 and 1.0% per day.

This review has shown that the simple concept of a perfect steel enclosure is, in practice, transformed into a more complex structure. Various alternatives to full containment have been under investigation for some years. These aim at reducing the amount of fission products available for leakage or reducing the pressure causing the leakage. Some fission products may be absorbed on charcoal, copper, or other active

surfaces or combined in solutions. Water or solution sprays can be used to remove heat, to condense steam, and to trap fission products. Combinations of these methods can be used.

One type of containment is illustrated by the "confined" reactor, for example, the Oak Ridge Research Reactor (ORR) at Oak Ridge, Tennessee. The reactor building has a reasonably low leakage rate under atmospheric pressure conditions and is ventilated through a scrubbing plant. This system may be acceptable for those reactors which have a low-energy primary circuit that cannot accidentally release considerable volumes of steam or gas.

A second system, called "vapor suppression," has been under investigation in the United States for several years for use with water reactors. In the case of a fractured circuit, this system seeks to reduce the peak pressure to which the containment is subjected by rapidly condensing the steam released. In one form the escape path of the contents of the primary circuit is arranged so that the steam will pass through a water trap and the inner building will thus be subject to a lower total pressure pulse of short duration.

Another system uses the idea of "vented containment" and exploits the fact that a significant interval could elapse between the fracture of the primary circuit and the melting of fuel, with its accompanying release of fission products. Thus, the containment structure would allow the initial release of the gas or steam-water-air mixture, but the valves in the structure would be arranged to close before there was a significant release of activity. The containment structure is designed for relatively low pressures.

In the view of the author, these developments in the containment concept, representing a move away from the idea of full pressure containment, are to be welcomed as a first step. If release of fission products from the reactor can be positively prevented by simple emergency procedures of the type discussed in this paper, no containment would be necessary. Structures around reactors would serve such functions as preventing ingress of air in a gas-cooled reactor, maintaining an inert atmosphere in a sodium-cooled reactor, etc. Simple filtration systems fitted to these structures would deal with fuel-handling accidents, spillage of active coolant, etc.

Conclusions

Some of the arguments presented have been oversimplified in an effort to bring out more forcibly the main ideas of the paper. There are exceptions and qualifications that could modify, but not fundamentally change the themes which have been developed, and which may be summarized as follows.

In new reactor systems, the safety analysis proceeds more slowly than the design, hindered by lack of information (either basic to the new system or related to imcomplete design details) and, perhaps, by lack of effort. Safety measures are then superimposed, often at considerable expense, and are sometimes strained unnaturally to fit the finished model. Much of this is initially inevitable but, as development is undertaken to give improved economy or reliability, so should safety provisions be developed as an integral part of the over-all growth. Such development should reduce, rather than increase, the extent of the safety provisions.

One form of this development is the recognition of those accidents which represent the greatest hazard to the reactor. For gas-cooled

reactors the worst accident is depressurization. At the present stage of development, acceptable ratings have been achieved for these systems even with the limitation that in this worst accident no significant release of fission products should occur from the fuel; hence, at no stage is a containment structure required. For the fast reactor, the safety limitation may be the time taken to sense local blockage of coolant flow and shut down the reactor to prevent spread of damage. If fast reactor safety in this situation could be assured, it might be possible to show that no substantial release of fission products could occur and, hence, that containment would not be needed. In the present development of pressurized-water reactors, the fundamental approach to safety by necessity differs from that outlined above. Although the system has inherent characteristics that assure safety in the event of most types of accident, it is not yet possible to guarantee that fuel melting will not occur in a depressurization accident. The provision of containment is a natural consequence of this finding.

In addition, the development of understanding of the transient behavior of a reactor has an important effect on safety considerations. This paper has shown in detail for Magnox reactors, and in outline for other reactors, that for well-established systems the basic safety provisions need to be few and simple and that the safety system should be isolated from other systems. The parameters measured, the equipment used, and the diversification of the equipment should depend in each case on the transient behavior of the particular reactor and the proven reliability of the systems currently available. The reactor safety systems should comply with very high standards in design and installation, should be subject

to stringent supervision and test during commissioning and thereafter at defined periods, and should not be modified or allowed to fall below some designated level of availability between inspection periods.

Other and different equipment may be installed to give information to the operator and may be used to an extent determined by the owner to anticipate operator action by automatic reduction of power for reasons other than safety. Equipment used for this purpose should not form part of the safety system.

Present thoughts on the safe operation of existing power reactors or those shortly to be commissioned need to be projected over a period of 10 to 20 years. During this time operating staff will change and equipment will be modified. The author is convinced that some simplification of safety measures is required in order that people concerned should retain a proper understanding and continue to apply an adequate standard of maintenance and inspection over this period. This can be achieved if the objectives are sufficiently well defined and accepted by all groups who have a responsibility in this matter. This is not soley the responsibility of the designers; others are involved, notably their safety advisers, operators, and the licensing authority.

Acknowledgment

Much of the material presented in this review appeared separately in papers to the British Nuclear Energy Society[6] and the Anglo-Japanese Nuclear Power Symposium.[7] In the preparation of this paper the author acknowledges the considerable help given by members of his staff, notably E. V. Gilby and J. R. Beattie. The reader is also referred to other publications (refs. 8-33), mainly of United Kingdom origin, which, together with

the extensive references already available to the readers of this journal, provide a general background but do not purport to substantiate the theme presented, for which the author and his colleagues take sole responsibility.

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VI. CURRENT EVENTS

OPERATING EXPERIENCE OF THE OAK RIDGE RESEARCH REACTOR THROUGH 1962

W. H. Tabor

The Oak Ridge Research Reactor and its associated experimental facilities provide general-purpose tools for advancement of the research program of the Oak Ridge National Laboratory and other organizations affiliated with the Atomic Energy Commission. The Oak Ridge Research Reactor (ORR) was the first of a class of reactors that combine the features of both the pool- and tank-type reactors. The design emphasized versatility to provide for convenient access to high-flux regions and for rapid fuel changes, as described in the design report.[1] A second published report[2] reviews the overall design, modifications to the initial design, and the major operating problems encountered.

The ORR is highly enriched, light-water moderated and cooled, beryllium reflected, and contained in an all-aluminum system. The reactor tank is shown in Fig. 1. The heat load of the reactor (30 Mw) is dissipated to the atmosphere through shell-and-tube heat exchangers and a conventional cooling tower.

Routine Operation

Following the completion of preoperational hydraulic testing,[3] criticality was attained on March 21, 1958. Routine power operation at 20 Mw began on July 20, 1958. Normal operation consisted of three weeks at power followed by a one-week shutdown for insertion and removal of experimental equipment and isotopes, refueling, and miscellaneous routine maintenance work. During the early months of operation, the three-week operating

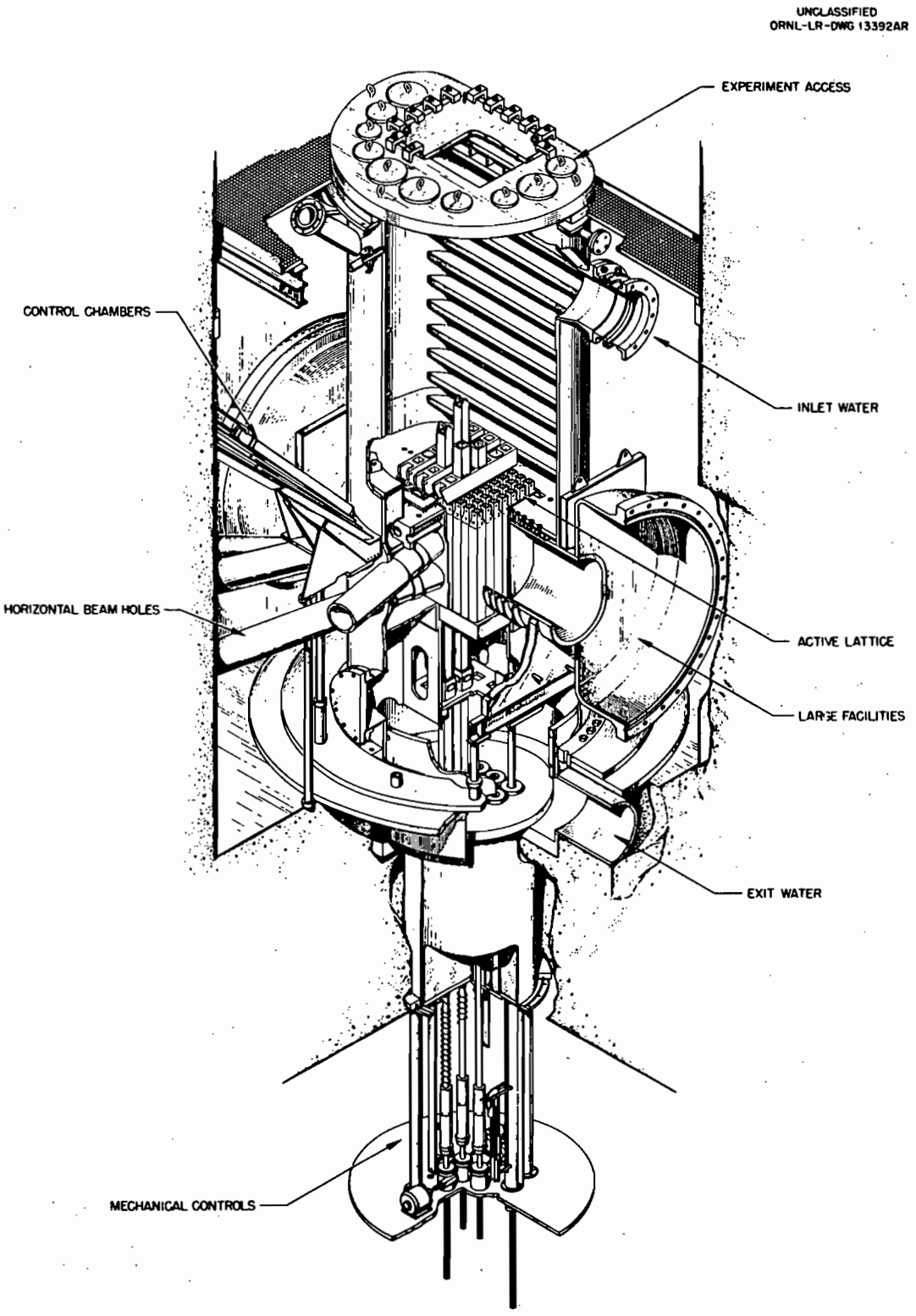

Fig. 1. Cut-Away View of Reactor Tank.

cycle was interrupted frequently because of component malfunctions. Occasionally, as a result of such interruptions, refueling was required because of xenon poisoning.

During the first two years of operation, the reactor was operated at 20 Mw during the cooler months of the year from about October to April. During the remaining months, the reactor power was limited to 16 Mw because of the inadequate heat-removal capacity of the originally installed water-to-air coolers.

Operation at 30 Mw was begun on July 29, 1960, following major modifications to the cooling system. Shell-and-tube heat exchangers and a conventional cooling tower were incorporated in a secondary cooling system to replace the inadequate water-to-air coolers. Concurrent with this higher operating power level, the operating cycle was altered to cover eight weeks - seven weeks of operation and one week for shutdown activities. Upon adopting the longer operating cycle and higher power level, it was necessary to interrupt the operating phase periodically for refueling and for isotope removal and insertion. Refueling is necessary at least every 18 days.

Proper scheduling of shutdown activities, which includes close coordination between simple experiments and the more complex in-reactor, loop-type experiments, and a good preventive maintenance program resulted in operation during greater than 80% of the year 1962, as indicated in Table 1. An analysis of reactor shutdowns is also included in this table.

Many activities associated with operating the ORR, or any nuclear reactor, depend upon administrative control. These activities are of varying degrees of complexity and affect the operating conditions of the

Table 1. ORR Operating Data

	1959	1960	1961	1962
Operating time, %	75.4	72.6	79.4	82.6
Number of shutdowns	65	78	83	74
Caught by xenon	2	1	3	2
Number of shutdowns caused by reactor component failures	24	21	29	26
Instruments	8	1	0	0
Rod drive units (mechanical and electrical)	12	14	28	22
Results of human error	4	3	1	2
Other	0	3	0	2
Number of shutdowns caused by failure in experimental equipment	6	24	12	7
Instruments	6	23	11	6
Other causes	0	1	1	1
Number of shutdowns caused by power failures and other events	5	4	5	6
Scheduled shutdowns				
For operations	25	24	33	32
For experiments	5	5	4	3

reactor in many ways. In order to standardize operating techniques, formal procedures were written covering all aspects of operation and were published in the Operating Manual for the Oak Ridge Research Reactor,[4] which is maintained current by updating sections or parts of sections as they become obsolete. New procedures or revisions of old procedures must be approved by both the ORR supervisor and the operations department superintendent. In addition, the general administrative controls applicable to all Oak Ridge National Laboratory research reactors have been described and published.[5]

Reactor Components

A number of changes have been made in instrumentation both to enhance operational safety and to provide continuity in operation. The nuclear and process instrumentation, as initially designed and installed, was considered to be inadequate for routine operation above 20 Mw. Consequently, it was necessary to make several major modifications to the system to provide the reliability and coverage required for 30-Mw operation. In addition, experience gained in operating the reactor at power indicated that several other improvements were required. Only the major modifications and additions are described here. A comprehensive review of the maintenance program for the instrumentation and reactor control systems has been published.[6]

Nuclear Instrumentation. The initial system included only one magnet amplifier for each control rod, with the result that a failure of one amplifier would cause a reactor shutdown. In order to eliminate this annoyance, dual amplifiers were installed for each control rod and circuit modifications were made to permit one of the amplifiers to assume the entire current load upon failure of the other amplifier.

During the first few cycles after these modifications were made, unexplained insertions of one or more of the control rods occurred. An investigation indicated that the hydraulic force acting on the rods exceeded that which had been expected, thereby yielding magnet-release times of about 1 to 2 msec instead of the 20 msec for which the system was designed. The response time was increased by incorporating a resistor-capacitor network in the magnet-amplifier and sigma-amplifier circuits that minimized the effect of voltage "spikes." Operation

continued, after these modifications, with very good results. New magnets were installed during the period between July and September 1959, and the resistor-capacitor network was removed.

The neutron-sensitive control chambers are, because of their location, greatly influenced by the contents of the beam holes, which are located directly beneath them. Special tests revealed that draining the shielding water from a single beam-hole liner could increase the ion-chamber currents by as much as a factor of 5. For this reason, safety precautions precluded the filling or draining of a beam-hole liner during operation. The initial design of the beam-hole plugs made the void of the plug common with the liner, but, since it was occasionally desirable to provide shielding to permit access of experimenters to their equipment, the beam-hole plug was redesigned. The new design effectively makes a "water can" of the plug, so the plug can be filled and drained independently of the beam-hole liner. Special tests showed that no significant changes in ion-chamber currents occurred when filling or draining the modified beam-hole plug. Operating history indicates that neither the research program nor reactor operation is retarded by using the modified beam-hole plug.

The permanently mounted fission-chamber channel, which is used as the startup channel, has displayed some peculiarities. The fission chamber, as initially designed, was driven from the bottom of the reactor tank by a remotely operated drive unit that gave movement of from 56 in. to within about 18 in. of the fuel. During a hot startup the fission chamber may indicate a decrease in the counting rate by a factor of 10 prior to indicating the neutron multiplication in the reactor. This

action is attributed to moving the fuel section of a control rod containing a large amount of fission products away from the fission chamber. This decreases the number of photoneutrons generated by the D_2O in the water in the vicinity of the fission chamber. The fission-chamber channel "sees" the neutron multiplication of the reactor when the control rods are about 10 in. withdrawn (criticality is attained with the rods at about 15 in.). During a cold startup the channel responds in a normal manner.

Mechanical troubles with the fission-chamber drive unit in December 1960 made it necessary to improve the reliability of the fission-chamber system. A temporary auxiliary chamber was located at the pool-side to permit continuity of operation while means for providing a reliable chamber having power response to shim-rod movement were investigated. Based on the results of a special experiment,[7] two poolside fission-chamber channels were fabricated. Operational and reliability tests indicated that the new channels were acceptable, and one channel was incorporated in the reactor safety control circuit, while the second channel is reserved for use in research and development work.

In an effort to minimize shutdowns resulting from failures in instrumentation, redundancy has been the rule in critical circuits. For example, two log-N channels, identical in design and independent of each other from the chamber to the output of the period amplifier, have been installed. Circuitry is provided to permit the use of one channel or the other, but not both simultaneously, as the operating channel. Only the channel selected is connected to the reactor control circuit, and thus the remaining channel is available for maintenance.

Three independent safety channels, which provide fast scram protection if the power level increases to 150% of the normal value, have been in service since initial operation. A minimum instrumentation criterion has been adopted which requires that at least two channels be operable. In order to provide electronic control action to assure that the reactor will not be operated with less than two reliable safety channels, circuit modifications were made which provide an automatic reduction in power if minimum requirements are not met. To provide additional safety protection while the reactor is operating at low power (less than about 450 kw) without normal water flow, the safety-channel circuitry was modified to increase its sensitivity by a factor of 100. A "fast scram" will occur when the reactor exceeds the 450 kw level under low-flow conditions.

Process Instrumentation. Some of the process instruments on this reactor are integral parts of the reactor safety system. Continuous effort has been exerted to update the circuitry and to provide reliability comparable with that of the nuclear instrumentation. With the increase in power to 30 Mw, the need for reliability was even more acute because cooling at the higher power is more critical. Improvements to enhance safety included (1) replacing thermocouples in the reactor exit water lines with resistance-bulb thermometers and providing a redundant circuit, (2) installing a duplicate differential-temperature channel in the reactor water system, (3) providing flow-monitoring circuits for the engineering test facilities to determine changes in flow distribution resulting from experimental system changes, and (4) providing redundancy in flow monitoring by installing a channel to measure the pressure differential across the core. Since these instrument circuits monitor critical

operating parameters, it was felt that redundancy or circuit duplication was necessary to establish the degree of reliability desired. In addition, duplication of channels permits continuity of operation upon failure of one channel; and immediate repair can be made.

The majority of the process instruments perform control functions that are not directly related to safety. The control function may be, for example, adjustment of the water flow through a heat exchanger. Since it is desirable from an operational viewpoint that all instrumentation function as a unit, considerable effort is exerted to assure proper preventive maintenance. As a result of this program, there have been very few interruptions resulting from component failure or malfunction. For example, during 1962 there were no process instrument failures that caused reactor downtime.

Mechanical Controls

With the increase in water flow from 16,000 to 18,000 gpm, the number of spurious rod drops increased. In addition, there were instances of control rods failing to drop within preset time limits and sluggishness in the operation of the scram-latch mechanism of the control-rod drive. These abnormal occurrences necessitated a comprehensive investigation of the complete operation of the control-rod-drive assembly. At the onset of drive trouble, however, a schedule of routine replacement of drive units was initiated, since it was suspected that several parts of the drive unit were contributing to the malfunctions. Rigid quality control and technical inspection of components of the replacement units reduced the number and severity of the malfunctions. A new drive system has been

developed, and the investigation will continue until a prototype has been proved operationally acceptable.

The frequency of spurious rod drops has been an operating nuisance. While this action does not jeopardize reactor safety, it is an operating inconvenience and could, if allowed to persist, prove costly in undesirable reactor downtime and delay of research programs. It now appears that the remaining difficulty, that of differential thermal expansion within the drive unit, can be minimized by controlling the cooling water flow to the drive seal so as to minimize temperature changes. Operation under this controlled condition is contemplated until the newly designed drive unit mentioned above is installed. The new unit will include an automatic compensator to eliminate the problem of differential thermal expansion.

Water System

The ORR water system, as initially designed and constructed, consisted of three independent systems: (1) a reactor primary system, (2) a pool primary system, and (3) a pool secondary system. Heat removal from the reactor system was accomplished by passing approximately 16,000 gpm of water through the core and using air-cooled heat exchangers to dissipate the heat. The pool primary system exchanges heat to the pool secondary system in a shell-and-tube heat exchanger, and an induced-draft cooling tower dissipates the heat. The cooling system is shown in Fig. 2.

Heat Exchangers. Shortly after the initial full-power operation (20 Mw) was attained, it became obvious that the air-cooled heat exchangers did not meet performance specifications. In addition, a serious wear problem developed in the heat exchangers because of vibration of the

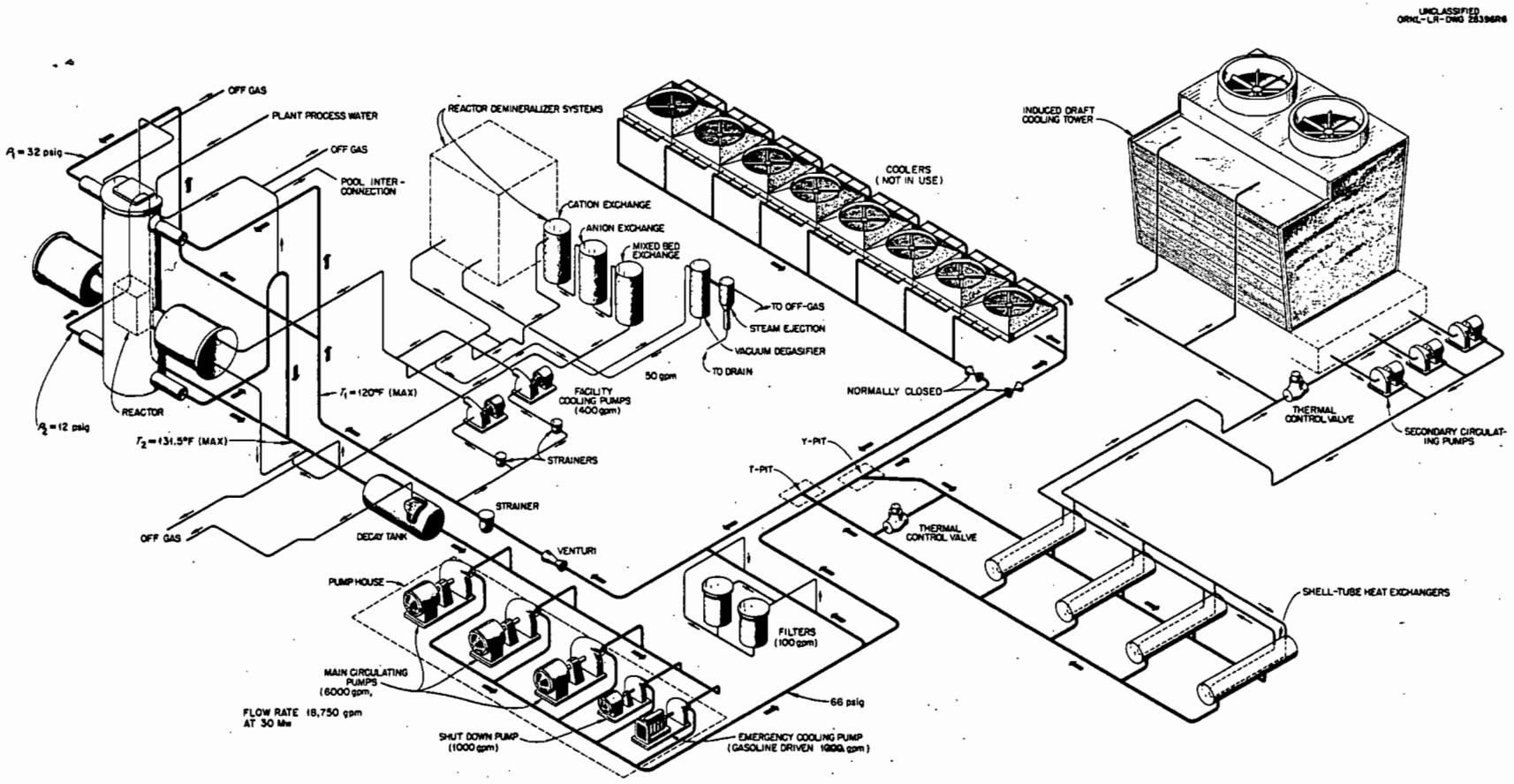

Fig. 2. Reactor Cooling System for 30 Mw Operation.

turbulators in the tubes. The turbulator is a long, spiraled, metal strip that is used in this application to increase the heat transfer area. An attempt by the manufacturer to correct the deficiency in performance proved unsuccessful; however, the wear problem was minimized by changing the method of securing the turbulators in the tubes. As a result of the cooling deficiency, the reactor was operated at a reduced power during the summer months until July 1960.

In order to increase the operating level to 30 Mw, it was necessary to make major modifications of the reactor primary cooling system to provide the needed heat-removal capacity. A decision to switch from water-to-air exchangers to water-to-water exchangers was based on the postulated radiation level that would result in the event of a fuel-element meltdown. Consequently the major modification consisted of the installation of a reactor secondary system, which uses stainless-steel shell-and-tube heat exchangers, located in an earthen-walled pit, and a conventional cooling tower.

Operation with the new cooling system has been successful, with the exception of one major component breakdown. During January 1962, leakage from the primary system to the secondary system in the heat exchanger was detected, and immediate repair was made by blanking off two tubes that had failed at the tube sheet because of excessive vibration. Tests conducted after the repair was made, consisting of ultrasonic measurements to determine wear on the tube walls and strain-gage measurements to determine vibration levels, showed that no additional leaks existed.

Corrosion in the System

Since the primary systems of both the pool and reactor contain aluminum components, extremely close observance of the corrosion rate has been necessary. Sample coupons of the different aluminum alloys of which the system is composed are continuously exposed and periodically inspected. The results indicate that the corrosion rate of the aluminum piping, pool liner, and core components is very low, being less than 1 mpy.[8]

In the pool secondary system, which uses chemically treated process water, the corrosion rates are also very low. Since the shell-and-tube heat exchanger of the pool system is made of aluminum, it also has been a point of concern. Corrosion rates measured on specimens exposed to the secondary water showed maximum corrosion rates of approximately 1 mpy.[8]

Corrosion rates where the aluminum plate is in contact with concrete or where aluminum piping insulated with a fibrous glass-wool material is encased in concrete are not quite so favorable. In July 1958, a section of 6-in. aluminum piping (63ST-6AL) was replaced because of a leak. Inspection revealed that the outer surface was badly pitted and had some holes that penetrated the pipe wall. The line was enclosed in 1 in. of glass-wool insulation, surrounded by a waterproofed cardboard shell, and then embedded in concrete, but the glass-wool had become water soaked from an unknown source. The pipe failure was attributed to galvanic corrosion promoted by the water-soaked glass wool.

As a result of this failure, means for providing a dry environment on the outside of the principal water lines were investigated. Core holes were drilled through the concrete to the annular spaces around the lines,

and a vacuum system was provided to evacuate these regions. Condensate collections from all points indicated that evacuation provided an acceptable environment. Observations of the environment are being continued on a routine basis by cold-trap collections.

During the shutdown of February 26, 1962, a leak in the pool liner was found. A detailed inspection revealed several small holes in the area where a 6-in. pool water line penetrated the liner. A sample of the liner was therefore removed for inspection and testing. A preliminary inspection showed severe pits on the side which was in contact with the concrete. This concrete-contact area was also water soaked. A quick repair was made by welding an aluminum plate on the poolside to cover the effected area. A line was also attached to a vacuum system to collect the water. After a few hours of pumping, the rate of water collection subsided to a level which was almost equivalent to air leakage. Pumping on this area is being continued. A complete inspection of the pool liner is being made using ultrasonic means for detecting pits and other faults.

Reactor Kinetics

Extensive tests for determining the kinetic behavior of this reactor were completed prior to power operation, and a comprehensive report[9] on the results was published. The initial core loading utilized fuel elements that weighed from 70 to 170 g, and it contained a total of about 3.9 kg of U^{235}. The temperature coefficient was negative, being -6.57×10^{-3} $(\Delta k/k)/°F$, and linear between the operating limits of 70 and 120°F. During approximately five years of operation, however, the core configuration has been changed to compensate for the changing experimental load,

with the result that the total U^{235} loading has increased to about 5.0 kg. Results of a recent rod-worth test indicate that the absolute worth has decreased slightly with the larger core now in use. Since the criterion for fuel loading in this reactor stipulates that not more than one half the worth of the four control rods shall be loaded in the core and since this requirement is determined by empirical calculations and checked experimentally for each core loading, it is not necessary to keep a current determination of the absolute rod worth. Further, the rod worth changes continuously during a fuel cycle, and thus it is impractical to determine the absolute worth.

The reactivity effects of flooding the experimental facilities were investigated, and the maximum void worth of a single facility was found to be $-0.86\% \Delta k/k$ (ref. 9). Since the experimental loading of the reactor is continuously changing, the reactivity worth of each experimental loading is determined. The total reactivity allowed for a single in-core experiment is $0.5\% \Delta k/k$ or less.

Experiments

The program for use of this reactor centers around the large number of experimental facilities, which vary from simple, relatively inexpensive equipment for the irradiation of capsules to the more complex gas-cooled loops for testing fuel elements. There are presently 35 experimental facilities, and they occupy about 88% of the space available for irradiations, the remainder being used for isotope production.

During the past two years, an average of 16 experimental facilities has been connected to the reactor-control system through the experiment

tie-in system. The range of complexity of the protective instrumentation is considerable. For example, one simple, static-irradiation facility has a single parameter capable of initiating a reactor power reduction; however, a complex facility for dynamic experimentation has 88 parameters capable of initiating a reactor-power reduction. Standardization of the method[10] by which the protective instruments develop and transmit control signals to the reactor-control system has resulted in each facility having, to an identical degree, the reliability to initiate power-reduction signals. This standardization of design has made the preventive maintenance program more efficient and has limited reactor downtime caused by instrumentation failure to less than 2% of the total reactor downtime, even though there are about 370 independent tracks of instrument control information that are capable of reactor power reduction. Many of these channels have multiinstrument systems for integration, differentiation, etc., thus increasing the actual number of control channels to well over 800.

The operating history of experimental facilities in the ORR indicates that the experiments were well planned and carefully reviewed for safety. As a result, inconveniences to the operation of the reactor have been minor. Although several incidents have occurred in which an experimental facility released small amounts of radioactive gas into the building, none of the releases resulted in personnel overexposure and none produced extensive contamination. In some instances, the building was evacuated for short periods of time; for instance, since 1960 the building has been evacuated on three occasions. Most of the releases have occurred during removal of experimental specimens and equipment while the reactor was shut down. The recurrence of similar incidents has been precluded by the improvement of removal procedures and techniques.

Personnel Radiation Exposure

A summary of personnel radiation exposure data for 1962 for the 41 men in the Reactor Operations Department, who operate three reactors, is given in Table 2. The most significant fact apparent from these data is that the highest dose of penetrating radiation for the year was only 37.4% of the maximum permitted and the average was only 18.4%. Since the data represented are for personnel associated with three operating reactors, the contribution of exposure from the ORR is estimated to be about 40% of the listed values.

Table 2. Radiation Exposure Experience of the Reactor Operations Department for a One-Year Period Ending November 1962

	Highest Exposure	Average Exposure
Skin of the whole body and thyroid, rem[a]	2.76	1.45
Skin of the whole body and thyroid, percentage of annual maximum permissible dose	9.2	4.8
Total body penetration, rem[b]	1.87	0.92
Total body penetration, percentage of annual maximum permissible dose	37.4	18.4

[a]The maximum permissible dose for radiation to the whole body at the skin is 30 rem per year.

[b]The maximum permissible dose for penetrating radiation to the whole body is 5 rem per year.

Conclusion

In the field of nuclear research the ORR continues to serve as the useful research tool for which it was designed. While minor problems have been encountered during years of operation, none has been of a

severity that required an extended shutdown. Present plans are for continued operation at the 30-Mw power level, and it is felt that this reactor can serve its useful purpose for years to come.

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