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SPECIAL REPORT

HAZARDS SUMMARY REPORT
FOR
2-MEGAWATT OPERATION
OF THE
BATTELLE RESEARCH REACTOR

to

THE U. S. ATOMIC ENERGY COMMISSION

August 15, 1958

by

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STATEMENT OF REQUESTED ACTION

This report is submitted by Battelle Memorial Institute to the Atomic Energy Commission in support of a request for approval to operate the Battelle Research Reactor at a power level of 2-megawatts.

Specifically, Battelle is requesting permission from the Atomic Energy Commission to:

- (1) Operate the Battelle reactor at a power level of 2 megawatts continuously on a routine basis.
- (2) Increase the maximum amount of excess reactivity which may be loaded into the reactor from 4.6 per cent to 6.0 per cent above cold, clean critical. This increase is necessary in order to operate the reactor routinely at 2 megawatts for research purposes.
- (3) Operate the reactor intermittently at power levels greater than 2 megawatts but not to exceed 3 megawatts for a maximum of 40 megawatt-days per year to evaluate the use of pool reactor systems at this higher power level.

SUMMARY

The Battelle Research Reactor is a modified pool reactor located in a sparsely settled 500-acre tract owned by Battelle 13 miles west of Columbus, Ohio. The reactor has been operating since October 29, 1956, at power levels up to 1 megawatt. More than 300 megawatt-days have been logged on the reactor with most of this operation at a power level of 1 megawatt. No serious incidents have occurred during the operation of the reactor.

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With only a few modifications the reactor will operate continuously at a power level of 2 megawatts. First, interlock will be added to the safety instrumentation to scram the reactor in the event of significant reduction in the flow rate of the primary coolant. Second, to provide additional margin of safety against the possibility of local boiling occurring in the core, the primary flow rate will be increased to greater than 1500 gpm. Third, a 700-gallon holdup tank will be added to the primary coolant system to allow for decay of the nitrogen-16 formed in the coolant as it passes through the core. Holdup will reduce the dose rate at the surface of the pool to 4 mr per hr or less during routine 2-megawatt operation. Dose rates in vicinities of normal personnel activities will be less than the surface dose rate by at least an order of magnitude.

Analyses of reactor safety and the hazards after an accident indicate that the change in hazards associated with the increase in power level is much less than a factor of two. It is concluded that the change in hazards does not significantly affect conclusions of the hazards analysis for 1-megawatt operation.

INTRODUCTION

Since the beginning of routine operation, the Battelle Research Reactor has proven to be a safe, reliable, and extremely valuable tool for research. Many experiments have been successfully concluded. These experiments have produced results important to nearly all fields of scientific endeavor. Irradiations have been performed on fuel specimens, hydrocarbons, semiconductors, and dosimeters, to name a few. Special facilities such as a neutron spectrometer, modified pile oscillator, and shielding tank have been operated to obtain data on nuclear properties of materials. Radioisotopes have been produced for medical and tracer applications.

During the operation of the reactor it has become apparent that significant research gains can be provided to the programs currently using the reactor and those envisioned for the future by increasing the power level. After several months of study, and on the basis of information obtained during 1-megawatt operation of the reactor, it is concluded that with only a few modifications the present system can be adapted to safe operation at a power level of several megawatts. Specifically, it was found that with these few modifications the system could be operated routinely at 2 megawatts and intermittently at power levels up to 3 megawatts. Although the system appears safe for continuous operation at power levels greater than 2 megawatts, the principal limitations are the capacities of the present cooling system and, ultimately, the control-rod system. With the reactivity allotments necessary for routine operation at 2 megawatts, intermittent

operation at 3 megawatts is possible and safe. However, for continuous operation at power levels very much above 2 megawatts the increased poison accumulation in the core dictates alteration of the present control scheme to provide more control-rod worth.

Therefore, it was concluded that the most practical and expedient course of action would be to make the few modifications required to operate the reactor routinely at 2 megawatts and to explore the behavior of the system at higher powers by operating intermittently at power levels up to 3 megawatts. A total operating period of 40 megawatt-days per year at powers greater than 2 megawatts but not in excess of 3 megawatts appeared adequate to assess the behavior of the system.

The hazards report BMI-ACRS-601 (Rev.)^{(1)*} describing the reactor facility and analyzing the hazards associated with its operation at 1 megawatt was submitted to the Commission in May, 1955. It discussed in detail the reactor site, including climatology, geology, hydrology, seismology, and the distribution of population and industry about the site. In the present report only a description of the reactor facility and a brief summary of the 1-megawatt hazards have been incorporated from BMI-ACRS-601 (Rev.).

GENERAL DESCRIPTION OF THE REACTOR

An aerial photograph of the reactor site is shown in Figure 1. The building in the foreground is the Research Reactor facility. The smaller building, in the left background, is the Hot-Cell Laboratory and the building on the right is the Critical-Assembly Laboratory.

Reactor Building

The Reactor building is a three-story structure, roughly 110 by 70 ft on the sides, with the first floor below grade. The building is constructed of a combination of cement block faced with brick and aluminum Q-metal siding.

The building contains about 7100 ft² of laboratory area, another 1200 ft² for reactor components (including control room and pump room), and an additional 4500 ft² for office space, storage room, boiler room, loading area, and miscellaneous functions.

*References at end of report.

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FIGURE 1. AERIAL PHOTOGRAPH OF REACTOR SITE

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Reactor

The reactor is a modified pool type currently operating at a power of 1 megawatt. The core is suspended from a movable bridge into a large pool of water. A second movable bridge is provided for suspending instruments and experiments into the pool near the core. The pool walls are modified at one end to form a stall area where six beam tubes and a thermal column meet the core face. Figure 2 shows a sketch of the pool and bridges. The pool is a reinforced-concrete enclosure about 60 feet long by 26 feet wide and 28 feet deep, containing approximately 125,000 gallons of demineralized water. Barytes high-density concrete was used for various portions of the pool walls where open areas at the outer wall surface presented possibilities of radiation exposure to personnel. Regular concrete was used wherever barytes was not required. The pool is divided into two interconnected parts, and the core-support bridge and instrument bridge may be moved into either section.

Above 100-kw power, forced cooling is required to prevent the pool-water temperature from gradually rising. There are provisions for forced cooling in each section of the pool. The pool water is pulled down through the core into an intake connection which leads to a U-tube-type heat exchanger in the pump room. From the shell side of the heat exchanger, ordinary tap water is pumped to a spray cooling tower outside the reactor building to dissipate the heat generated in the core. This downdraft type of forced cooling also minimizes the radiation hazard at the pool surface from the nitrogen-16 activity. In addition to the cooling system, purification and storage facilities are also provided. A diagram of the cooling, purification, and storage system is shown in Figure 3.

The horizontal cross-sectional view through the stall end of the pool (Figure 2) shows the beam-tube and thermal-column arrangement. Two 8-in. -diameter tubes meet the core face perpendicularly and four 6-in. -diameter tubes meet the core at angles. The thermal column is about 12 feet in length and is stacked with high-purity graphite blocks.

Fuel Elements

MTR-type fuel elements are used to form the reactor core. A sketch of the element cross section is shown in Figure 4. The element has ten curved plates, each plate containing a central layer of uranium-aluminum alloy sandwiched with aluminum. The total uranium-235 content of each full fuel element averages approximately 162 g. The special fuel elements adapted to receive the control rods have only seven plates, with an average uranium-235 content about two-thirds that of a standard assembly. Also, two partially loaded elements have been constructed with only half the uranium-235 content of the regular fuel elements.

FIGURE 2. SKETCH OF REACTOR POOL AND FACILITIES

FIGURE 3. COOLING, PURIFICATION, AND STORAGE SYSTEMS

FIGURE 4. STANDARD FUEL ASSEMBLY FOR BATTELLE RESEARCH REACTOR

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Control and Instrumentation

The reactor has a total of five control rods: four shim-safeties containing boron carbide (B_4C) canned in aluminum and one stainless steel shell regulating rod. The four heavy rods are held in position with electromagnets. The current to the magnets is interrupted and the rods fall by gravity into the core if certain malfunctions occur.

The reactor is controlled from an instrument console located in the control room on the third floor of the building. Four principal channels are used for operation: (1) a startup channel using a fission chamber to detect neutron multiplication, (2) a power channel employing a compensated ionization chamber (CIC) to indicate reactor power level, (3) a control channel using a CIC in combination with a servo drive system for automatic operation at a preset power level, and (4) a safety channel with two parallel-circular-plate (PCP) ion chambers feeding a power-level signal directly to the safety amplifier, which controls the current to the control-rod electromagnets. All four channels have safety interlocks connecting to the safety amplifier. In addition to the operating channels, the console contains instruments for control-rod drive, temperature indication, and radiation indication. A block diagram of the principal instrumentation is shown in Figure 5.

Characteristics of the Reactor

Prior to routine operation at 1 megawatt, various experiments were performed with the reactor to investigate the general characteristics of the system. Several copies of a report describing these experiments have been submitted to the Commission.⁽²⁾ A brief summary of the results of the more important experiments is given in Table 1.

OPERATING EXPERIENCE

The Battelle Research Reactor has been operated for approximately 2 years. Since initial criticality (October 29, 1956) until August 11, 1958, the reactor has operated 304 megawatt-days without a serious incident. During this operating period 729 startups have been performed. The reactor has been operated routinely at 1-megawatt power since April, 1957. As most of the operations prior to this date were at low reactor power, most of the 304 megawatt-days have been logged during routine operation.

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FIGURE 8. BLOCK DIAGRAM REACTOR CONTROL AND INSTRUMENTATION
FIGURE 5: BLOCK DIAGRAM OF REACTOR CONTROL AND INSTRUMENTATION

TABLE 1. CHARACTERISTICS OF THE BATTELLE REACTOR

Minimum Critical Mass	2.8 kg uranium-235
Average Void Coefficient (Approximate) ^(a)	$-2 \times 10^{-6} \frac{\Delta k}{k}$ per cm ³ of void
Isothermal Temperature Coefficient (95 to 105 F)	$-2.7 \times 10^{-5} \frac{\Delta k}{k}$ per C
Total Control-Rod Reactivity Worth (Loading 8-AJ) ^(b)	8.07 per cent
Control-Rod Drop Time	0.5 sec
Total Reactivity Worth of Beam Tubes ^(c)	0.5 per cent

(a) Information supplied to the Commission in a letter dated January 10, 1958 (Communication N-8).

(b) Loading 8-AJ is approximately a 5 by 6 array of fuel elements and has been used during most of the routine operation of the reactor.

(c) Changing from air filled to water filled.

As attested by the operating experience, the reactor has demonstrated satisfactory performance both from the safety and operational standpoint. The number and duration of interruptions during routine operation have diminished with operating time. From January 1, 1958, to August 11, 1958, the actual operating time was 95 per cent of the scheduled time.

Since initial criticality, the staff has gained much insight into reactor operations. More than 30 persons have been trained to operate the BRR; 15 persons are permanent members of the operating staff. Although many of the experiments on the reactor were performed prior to routine operation, a backlog of valuable information has been gained on the day-to-day operation of the reactor for research purposes.

REQUIREMENTS FOR HIGHER POWER OPERATION

Based on the operational behavior of the reactor at 1-megawatt power, the requirements for additional instrumentation and equipment and reactivity allowances for operation at higher power levels can be estimated. The principal requirements are discussed below.

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Instrumentation and Equipment

The instrumentation for reactor operation illustrated in Figure 5 is adequate for routine operation at 2 megawatts and intermittent operation at 3 megawatts for experimental purposes. However, an additional interlock on the cooling system appears desirable. At present a safety interlock requires forced convection through the core at power levels greater than 100 kw before current can be supplied to the control-rod lifting magnets. This feature was provided to insure that the reactor could never reach a boiling condition due to lack of coolant flow. This interlock will remain in the system, and, in addition, the flow rate in the primary system will be monitored and instrumented to scram the reactor if there is a significant decrease in the flow rate.

The capacity of the primary pump in the cooling system, shown in Figure 3, will be increased to greater than 1500 gpm and a nitrogen-16 holdup tank will be added. Increasing the primary flow rate will further insure that no boiling occurs in the core and will also permit intermittent operation at power levels up to 3 megawatts. Addition of a 700-gallon holdup tank will reduce the nitrogen-16 gamma radiation at the pool surface to less than 4 mr/hr under the most pessimistic conditions for continuous operation at 2 megawatts. Intermittent operation at 3 megawatts will produce radiation intensities of less than 6 mr per hr at the surface. Of course, the dose rates in areas of normal personnel activity around the pool are less than these maximum values by at least an order of magnitude.

The biological shielding provided by the pool water and concrete enclosure is adequate for much higher power operation. Except in vicinities of special research activities, the dose rate at the surface of the biological shield is less than 1 mr per hr at the present 1-megawatt power level.

With the modifications discussed above, the instrumentation and equipment will provide for safe operation at the higher power levels.

Reactivity Requirements

Battelle is currently licensed to load a maximum of 4.6 per cent reactivity into the reactor to permit routine operation with experiments at 1-megawatt power. To operate the reactor routinely at 2 megawatts requires an additional reactivity allowance of 1.4 per cent. This additional requirement is due entirely to increased xenon and samarium poisoning. Table 2 summarizes the minimum reactivity requirements for 2-megawatt operation.

TABLE 2. REACTIVITY ALLOWANCES FOR CONTINUOUS
OPERATION AT 2 MEGAWATTS

	Reactivity Required, per cent $\frac{\Delta k_{\text{eff}}}{k_{\text{eff}}}$
(1) Control	0.3
(2) Xenon Poisoning at Equilibrium	2.8
(3) Samarium Poisoning at Equilibrium	1.2
(4) Fractional Element	0.2
(5) Core Experiments	<u>1.5</u>
Total	6.0

It should be pointed out that the reactivity required for Items (2) and (3) is not available to the operator once equilibrium has been reached.

No allotment has been made for override of xenon poisoning after a shutdown. It is estimated that at a power level of 2 megawatts an additional reactivity of 1.4 per cent is required to override shutdown poisoning at its peak (about 7.5 hours after shutdown). Although it would be desirable to incorporate this additional reactivity into the core as a convenience in operation, it is felt that the operation will be satisfactory with the reactivity allowances shown in Table 2. Should override at shutdown be required, experiments can be removed from the core and the reactivity allotted for Item (5) used to override the peak poisoning. This procedure has been followed during 1-megawatt operation when it has frequently been desirable to operate the reactor at low powers on "shutdown" days to map the flux distribution in the core or to perform other measurements required for operational information.

Item (5) is a maximum allowance and only an amount sufficient to compensate for experiments will be added at any time. This is normally less than 1.5 per cent.

No reactivity allowance is requested for the intermittent 3-megawatt operation. This operation will be performed either at the expense of Item (5) or prior to the establishment of xenon-poisoning equilibrium. This procedure will be adequate, since it is desired to operate at the higher power only for sufficient time to obtain data to evaluate the system.

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No allowance for override of temperature effects has been included in Table 2 since these effects have been shown to be much smaller than originally estimated⁽²⁾. The small losses occurring during operation due to changes in the bulk temperature of the pool water will be absorbed in the allotment for Item (5).

ANALYSIS OF REACTOR SAFETY

A summary of the hazards involved in the operation of the BRR at 1-megawatt power has been presented in BMI-ACRS-601 (Rev.)⁽¹⁾. The results of this hazards analysis will be briefly reviewed and supplemented with more recent data. The changes in the hazards resulting from increasing the reactor power to 2 megawatts will then be discussed.

Review of 1-Megawatt Hazards

The principal public hazard in the operation of a reactor is the possibility of exposing persons to the radioactive fission products accumulated in the fuel during operation. The most probable manner by which large amounts of fission products can be released from the core is by a power excursion in which the reactor power increases at such a rapid rate that the first power burst is not controllable and is limited only by the inherent safety of the reactor design. If the power rises to a sufficiently large value, the core may melt down and possibly release some fission products to the atmosphere.

In hazards analysis, the possibility of a power excursion is considered during two conditions: reactor startup and during normal operation (reactor critical). The last condition generally leads to a more serious accident and is termed the maximum credible accident.

Startup Accident

A startup accident results when the reactor operator withdraws the control rods at their maximum rate without heeding the response of the reactor instruments. Simultaneously, all scram safety circuits are assumed inoperative. Table 3 lists the maximum possible rates of manual reactivity withdrawal for the BRR for various conditions based upon measurements of rod worth and withdrawal speeds.

TABLE 3. MAXIMUM POSSIBLE RATES OF MANUAL REACTIVITY
WITHDRAWALS FOR VARIOUS CONDITIONS FOR
LOADING 8-AJ^(a)

Condition	Remarks	Maximum Rate of Reactivity Change,
		per cent $\frac{\Delta k_{eff}}{k_{eff}}$ per sec
Shim safety rods 2, 3, and 4 withdrawn in gang	Normal procedure during startup	0.0293
Shim safety rods 2, 3, and 4 plus regulating rod withdrawn in gang	Possible but not normal procedure	0.0462 ^(b)
All four shim-safety rods withdrawn in gang	Not possible without wiring change	0.0358
All five control rods withdrawn in gang	Not possible without wiring change	0.0527

(a) Loading 8-AJ in approximately a 5 by 6 array of fuel elements; this loading (or a similar one) is used during routine operation of the reactor.

(b) The regulating rod, although worth much less than the other rods, withdraws about 5 times as fast.

As seen from Table 3, the maximum ramp rate conceivable with the control system is about 0.05 per cent reactivity per second and the normal rate during startup (prior to nearing criticality) is about 0.03 per cent per sec. A brief analysis of two startup accidents simulated in the SPERT-1 experiments which bracket the ramp rates shown in Table 3 indicates the following behavior for this type of reactor system.⁽³⁾ For a ramp rate of 0.01 per cent reactivity per second from an initial reactor power of 5 milliwatts, the reactor attains a period of approximately 800 millisec and reaches the first power maximum (about 1 megawatt) about 55 sec after the start of the ramp. The power behavior for this startup accident is not very time-sensitive and several minutes are available for manual action. For a ramp rate of 0.10 per cent reactivity per second (several times greater than the values shown in Table 3) from an initial reactor power of 5 milliwatts the reactor attains a period of approximately 56 millisec and reaches a power maximum of 30 to 40 megawatts about 8.3 sec after the start of the ramp. The power level then decays and stabilizes near 2 megawatts. Again, the behavior is not extreme.

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While it is questionable that the details of the behavior of the SPERT-1 experiments described above can be directly applied to the BRR, the significant conclusion is that for the ramp rates possible with the BRR control-rod-drive system, the time scale of a possible excursion is such that in the unlikely event of failure of the instrument safety system, the reactor operator has ample time to appraise the reactor behavior and to take effective measures to scram the reactor. Also, the peak power levels reached during the startup accident are not likely to cause meltdown of the core. Although it is possible that some damage to the core could occur, it is extremely unlikely that any fission products will be released.

On the basis of experience gained in operating the Battelle Reactor (729 startups as of August 11, 1958), there is every reason to believe that the safety record will continue; the probability of a startup accident appears extremely small.

Accidents at Operating Conditions (Maximum Credible Accident)

A sudden addition of reactivity with the reactor in an initially critical condition (operating) can also cause a power excursion. The amount of energy released in this excursion is a function of the reactivity instantaneously inserted. In estimating the energy release in the event of the maximum credible accident as reported in BMI-ACRS-601⁽¹⁾, a total instantaneous reactivity insertion of 2.6 per cent was assumed. This amount was based on removing core experiments (limited to 1.5 per cent) and simultaneously flooding or removing one 8-in. -OD and two 6-in. -OD beam tubes whose reactivity effect had been estimated at 1.1 per cent. However, as shown from experiments^(2,4), the removal of three tubes will introduce only about 0.25 per cent reactivity rather than the previously estimated 1.1 per cent. Thus, the revised total instantaneous reactivity insertion is about 1.8 per cent for the assumed maximum credible accident.

The energy released in an excursion caused by a sudden reactivity increase of 1.8 per cent has been calculated and the results included in Battelle's letter of July 3, 1957⁽⁴⁾. This calculation includes the effects of the BRR fuel-element geometry, i. e., the use of only 10 plates per fuel element. The method of calculation has been shown in Reference (5) to predict the BORAX energy release rather well. The results of the calculation indicate an energy release of 7 megawatt-sec, compared with an energy release of 40 megawatt-sec previously estimated for the maximum credible accident and 135 megawatt-sec for the most destructive BORAX experiment.

236 018

From the information present in the BMI-ACRS-601 (Rev.) and subsequent addendums, and subject to the conditions of the license, the Commission found that "There is reasonable assurance that the facility can be operated without endangering the health and safety of the public".⁽⁶⁾

Hazards From 2-Megawatt Operation

The probability of a startup or operational accident is not changed by the change in power level. For power excursions occurring when the core is still relatively free from fission products (clean), the magnitude of the activity released is likewise unchanged. The only significant change in the hazards situation results from a greater formation of radioactive fission products at the higher power level. Although the equilibrium value of short-lived isotopes will be doubled at the higher power, the fuel elements will be exposed for a shorter time (i.e., same fuel burnup). Thus, the accumulated radioactivity at 2 megawatts is less than twice that at 1 megawatt.

In evaluating hazards for 1-megawatt operation, the reactor was assumed to have been operating for 6.2×10^7 sec, the time required to produce 15 per cent burnup in the fuel elements. Evaluating the hazards at maximum burnup results in the worst case. Increasing the routine reactor power level to 2 megawatts decreases the operating time required to attain 15 per cent burnup of the core, i.e., the number of megawatt-days of operation is identical for the two cases. Thus the time behavior of fission-product formation and decay must be considered in the evaluation of potential hazards. Burnett⁽⁷⁾ has deduced that for inhalation the hazard index, HI, is related to power level, P, and operating time, t, as

$$HI = 112.5 P^{0.37} (Pt)^{0.63} \quad (1)$$

for times in the range 30 to 1000 days. Thus, if the allowable burnup is fixed at 15 per cent for both 1-megawatt and 2-megawatt operation, the product (Pt) is identical for both cases, and the hazard index HI_2 incurred in going to 2 megawatts becomes

$$HI_2 = \left(\frac{P_2}{P_1} \right)^{0.37} HI_1 = 1.29 HI_1, \quad (2)$$

or a 29 per cent increase.

In the same article, Burnett shows that the hazard from external exposure (direct radiation) received from the cloud is quite small when compared with inhalation effects. Direct radiation is about 2 per cent of the total received by a person in the cloud for operating time planned for the Battelle reactor. The fact that the internal exposure is considerably more important than the external exposure is due to the long retention time of the inhaled isotopes in the body.

The effect of the increased power level on the hazard from washout of the cloud can be estimated by comparison of the activity in the cloud for the two cases. The activity in the cloud can be approximated by⁽⁷⁾

$$A = 8880 [t^{-0.21} - (t_0 + t)^{-0.21}] \text{ curies per kw ,} \quad (3)$$

where

A = specific activity in cloud, curies per kw

t_0 = duration of operating time of reactor, sec

t = time after reactor excursion, sec.

In general, $t_0 \gg t$ for the times of interest, so that the predominant term in the brackets is $t^{-0.21}$, which is independent of power level. Thus, to a first approximation, the activity in the cloud is directly proportional to the reactor power level. (NOTE: This statement does not contradict the previous conclusion that the inhalation hazard increases by only 29 per cent, since for inhalation it is the isotopic composition of the cloud which determines the power-dependent function and not total cloud activity.) Since the dose rate is directly proportional to the cloud activity, the total dose rate received by a person due to washout at 2 megawatts would be twice that for 1 megawatt.

The hazard from fallout or washout is considered of less importance than that from inhalation since persons can be evacuated from washout areas before they accumulate harmful dosages from this source; i. e., for a washout to be harmful, a person must remain in the contaminated area for a considerable time.

Conclusions of Hazards Analysis

It is concluded that the increase in potential biological hazard associated with the increase in reactor power to 2 megawatts is not directly proportional to the power increase, and for the most significant case - hazard from inhalation of fission products - the increase in hazard is only

about 30 per cent. This increase is considered small enough to not significantly perturb the conclusions from the hazards analysis for 1-megawatt operation.

Hazards From Intermittent Operation
at 3 Megawatts

Although the increase in hazards associated with long-time operation at 3 megawatts can be deduced from the previous discussion, the true hazards associated with intermittent operation at 3 megawatts do not differ significantly from those associated with routine operation at 2 megawatts. Permission is requested to operate a maximum of 40 megawatt-days per year at power levels greater than 2 megawatts and not to exceed 3 megawatts. This operating time is small compared with the scheduled 626 megawatt-days per year operation at 2-megawatt power.

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- (7) Burnett, Thomas J., "Reactors Hazard vs. Power Level", Nuclear Science and Engineering, 2, 382-393 (1957).