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REACTOR DIVISION

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HIGH FLUX BEAM REACTOR

TECHNICAL SPECIFICATIONS

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CONTENTS

	<u>Page</u>
Introduction	3
Section 1 Definitions	1- 1
Section 2 Safety Limits and Limiting Safety System Settings	
2.1 Safety Limits	2- 2
2.2 Limiting Safety System Settings	2- 7
Section 3 Limiting Conditions for Operation	
3.1 Containment System	3- 1
3.2 Reactivity Control	3- 6
3.3 Shutdown and Emergency Cooling	3- 9
3.4 Primary Coolant and Cover Gas Chemistry	3-11
3.5 D ₂ O Heat Exchangers	3-15
3.6 Poison Solution System	3-17
3.7 Shield Cooling Systems	3-18
3.8 Radiation Monitoring	3-21
3.9 Reactor Shutdown Safety System	3-24
3.10 Standby Electrical Power	3-27
3.11 Spent Fuel Storage	3-28
3.12 Main Pump Start Interlock	3-29
3.13 Cold Startup Instrumentation	3-30
Section 4 Surveillance Requirements	
4.1 Containment System	4- 1
4.2 Reactivity Control	4- 3
4.3 Shutdown and Emergency Cooling	4- 6

	<u>Page</u>
4.4 Primary Coolant and Cover Gas Chemistry . .	4- 8
4.5 D ₂ O Heat Exchangers	4-12
4.6 Poison Solution System	4-13
4.7 Shield Cooling Systems	4-15
4.8 Radiation Monitoring	4-16
4.9 Reactor Shutdown Safety System	4-18
4.10 Standby Electrical Power	4-22
4.11 Spent Fuel Storage	4-23
4.12 Main Pump Start Interlock	4-24
Section 5 Reactor Experiments	5- 1
Section 6 Design Features	
6.1 Fuel Elements	6- 1
6.2 Reactor Site	6- 3
6.3 Design Criteria for Primary Coolant Boundary	6- 4
6.4 Control Rods	6- 5
Section 7 Administrative Controls	7- 1
References	

INTRODUCTION

These technical specifications which supersede the document "Brookhaven High Flux Beam Research Reactor AEC Operating Limits", are established in compliance with AECM 0540. The specifications define the boundaries for safe operation of the reactor and should not be interpreted as a guide for routine operation. Rules, procedures and limits provided for the daily operation of the reactor are generally more restrictive than the technical specifications and are covered in the HFBR Operations Procedures Manual.

The AEC "Guide to the Content of Technical Specifications for Nuclear Reactors", November 1968 and 10 CFR 50 were used as guides in the preparation of this document. Most of the bases for these technical specifications are given in the HFBR Safety Analyses Report and related documents which are appropriately referenced. However, in some instances - especially in Section 4, Surveillance Requirements - the bases appear here for the first time.

1. DEFINITIONS

These definitions apply to frequently used terms in the technical specifications which follow. They are not necessarily consistent with similar terms found in other HFBR documents.

1.1 Annually

Occurrences at an average rate of once every year with intervals between occurrences not to exceed 15 months.

1.2 Check

A qualitative determination of acceptable operability by observation.

1.3 Control rods

1.3.1 Main rod - One of a set of eight neutron-absorbing rods which can be moved either individually or as a group (gang) at normal drive speed or rapidly inserted as a group (see scram). These rods are withdrawn to positions above the core to increase reactivity.

1.3.2 Auxiliary rod - One of a set of eight neutron-absorbing rods which can be moved either individually or as a group at normal drive speed. These rods are withdrawn to positions below the core to increase reactivity.

1.4 Daily

Occurrences at an average rate of once a day with intervals between occurrences not to exceed 32 hours.

1.5 Fuel Cycle

Period of operation between fuel changes.

1.6 Monthly

Occurrences at an average rate of 12 times a year with intervals between occurrences not to exceed 40 days.

1.7 Movable experiment

An experiment that can be moved by the action of personnel during reactor operation.

1.8 Once a shift

Occurrences at an average rate of once every eight hours with intervals between occurrences not to exceed 12 hours.

1.9 Operable

The system or component is capable of performing its required function in its normal manner.

1.10 Quarterly

Occurrences at an average rate of four times a year with intervals between occurrences not to exceed 20 weeks.

1.11 Reactor Operating

When more than three main control rods are raised above the full in position.

1.12 Scram

Rapid, spring powered insertion of main control rods.

1.13 Semi-annually

Occurrences at an average rate of twice a year with intervals between occurrences not to exceed 8 months.

1.14 Setback

Electrically driven insertion of both main and auxiliary rods at their normal operating speed.

1.15 Test

Verification of proper response including, where applicable, alarm and/or trip initiating action.

1.16 Weekly

Occurrences at an average rate of once a week with intervals between occurrences not to exceed 10 days.

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Safety Limits are limits on important process variables which have a direct effect on the integrity of the principal fission product barriers, the fuel cladding and the reactor coolant pressure boundary. Each Safety Limit is intended to define the upper (or lower) bound for a variable when that variable is driven outside its normal operating range because of equipment malfunction, operator error or other cause. The general criterion used to establish the Safety Limit is that if the variable were at this limit and all other variables were to be at the unfavorable extreme of their operating range, and if all uncertainties in technical knowledge of the process were resolved unfavorably, no fission product release would occur.

The level of the process variable at which automatic protective action is initiated should be so chosen that the Safety Limit will not be exceeded for the most severe anticipated transient. This level is called the Limiting Safety System Setting (LSSS). Since the automatic protective system is designed for relatively rapid transients, the period of time during an accident in which the variable in question may be above the LSSS is less than a minute and very probably no more than a few seconds. Therefore, in determining the specific value of the Safety Limit, factors such as corrosion and fuel burnup which influence the integrity of the fission product barriers only in the long term, are not considered relevant. Technical specifications relating to these secondary factors will be treated in Section 3, Limiting Conditions for Operation and Section 6, Design Features.

2.1 Safety Limits

Applicability: Applies to reactor power, primary coolant flow rate, cover gas pressure, reactor liquid level and core inlet temperature.

Objective: To maintain the integrity of the fuel cladding and reactor coolant pressure boundary in order to prevent the release of significant amounts of fission products.

Specifications: The following Safety Limits shall not be violated:

Two pump operation at 40 MW

<u>Variable</u>	<u>Nominal Value</u>	<u>Safety Limit</u>
Reactor power, MW	40	79 (maximum)
Primary coolant flow rate, gpm	16,600	7700* (minimum)
Cover gas pressure, psig	200	300 (maximum) 44* (minimum)
Liquid level, inches	237	192* (minimum) 70** (minimum)
Core inlet temperature, °C	49 (120°F)	77* (170°F) (maximum)

One pump operation at 30 MW

<u>Variable</u>	<u>Design Value</u>	<u>Safety Limit</u>
Reactor power, MW	30	58 (maximum)
Primary coolant flow rate, gpm	11,800	5600* (minimum)
Cover gas pressure	200	300 (maximum) 36* (minimum)
Liquid level, inches	237	192* (minimum) 70** (minimum)
Core inlet temperature, °C	50.5 (123°F)	77* (170°F) (maximum)

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* Applies only while reactor is operating.

** Applies to accident conditions requiring low-low logic actuation.

Bases:Reactor Power

Accidents leading to an increase in reactor power would cause damage to the fuel cladding if burnout occurred. In addition, a power increase would raise the thermal stress level in the reactor vessel due to radiation heating and thereby threaten the integrity of part of the primary coolant barrier. Calculations indicate, however, that burnout will occur at a much lower level of reactor power than that required to cause serious thermal stress levels. Although the reactor vessel stresses cited in the Final Safety Analysis Report^(2.1) are close to the allowable, measurements made during the reactor startup period^(2.2) indicate that the nominal 40 MW power level would have to be elevated by a factor of 3-4 to reach the design stress values. Vessel stress considerations can therefore be safely ignored, and the computation of the Safety Limit will be related to fuel plate burnout. The Safety Limit is the power level at which the burnout ratio at the core hot spot is 1.0. At burnout ratios below 1.0 there is a finite possibility of burnout whereas a ratio above 1.0 implies no burnout. The calculation is essentially the same as that performed in the FSAR^(2.3, 2.4) with the following additional assumption: all process variables that can influence burnout, other than the variable whose limit is being calculated, (reactor power in this case) are at their LSSS.

Primary Coolant Flow Rate

Accidents resulting in loss of primary flow will cause fuel plate damage by burnout if the reactor is not shut down. Temperature levels in the reactor vessel will also rise on loss of flow, but for reasons noted in the previous paragraph, this factor is not limiting as far as the Safety Limit is concerned.

The Safety Limit is the coolant flow rate at which the burnout ratio is 1.0 and all other variables are at their LSSS.

Cover Gas Pressure

Loss of pressure will lead to fuel plate burnout if the reactor is not shut down. The lower Safety Limit on pressure was calculated in a manner analogous to the reactor power and flow rate computations using the fuel plate burnout criterion.

In addition to the possibility of fuel plate damage, loss of pressure can also cause pump cavitation if the cover gas pressure falls below 34 psig at rated flow^(2.5). However, this aspect of the loss of pressure accident will not be considered in determining a Safety Limit since pump cavitation does not jeopardize the primary coolant boundary.

Accidents which lead to increases in the cover gas pressure may ultimately damage the primary coolant boundary if protective action is not taken. The reactor vessel and other primary system components are conservatively rated for a cover gas pressure of 275 psig^(2.6) and the system was originally hydrostatically tested at a pressure in excess of 400 psig. The upper Safety Limit on cover gas pressure is set at 300 psig.

Reactor Liquid Level

The liquid level in the primary system is defined by the distance between the inside bottom of the reactor vessel and the helium-heavy water interface in the neck of the reactor vessel. The liquid level will affect the removal of reactor heat from the core in two ways: first as the liquid level falls, the pressure in the core will fall and therefore the margin to

burnout will be reduced. Secondly, if the liquid level falls below the vessel inlet nozzle (192 inches) it is likely that gas will be entrained in the water entering the vessel. With sufficient gas entrainment, blanketing and overheating of the fuel could occur.

If the cause of the loss of liquid level is a break in the primary system that cannot be quickly isolated, additional steps are taken to prevent the core from being uncovered and to provide for removal of the afterheat by natural circulation cooling. These additional steps are initiated at a different liquid level by the low-low liquid level logic system^(2.7). The Safety Limit for this condition, in which the reactor is shut down and generating fission product afterheat, is the bottom of the liquid level bubbler tube, 70 inches above the bottom of the reactor vessel and 2 inches above the top of the fuel elements.

Core Inlet Temperature

An excessive rise in the core inlet temperature can affect the integrity of the fuel cladding, the reactor vessel and other elements of the primary coolant boundary. These effects are discussed separately below.

The burnout ratio will be 1.0 at the core hot spot when the core inlet temperature rises to about 270°F. This value applies to both single loop and two loop operation and was calculated on the same basis as the previous burnout computations.

With regard to the reactor vessel, reactor startup measurements^(2.2) indicate that the core inlet temperature would have to rise above 170°F before the beam tube design temperatures

were reached. In addition, the thermal stress levels at these elevated temperatures would still be below the design values because of the relatively small temperature variations in the vessel wall and beam tubes at the normal 40 MW power level.

The third potentially serious effect of an increase in core inlet temperature is the higher stress levels in the primary piping due to thermal expansion. Design calculations^(2.8) based on a conservative estimate of primary loop temperatures indicated that the stresses are quite low, about 2700 psi at the worst point. These stresses are in direct proportion to the temperature difference between the "hot" conditions and the ambient temperature condition. Since the 2700 psi figure corresponds to a temperature increase of about 70°F, it is evident that a doubling of the temperature range between the "hot" and ambient temperature will not lead to serious stresses.

As a result of the considerations discussed above, the Safety Limit on core inlet temperature is established at 170°F for both single-loop and two-loop operation. This is 50°F above the normal operating level.

2.2 Limiting Safety System Settings (LSSS)

Applicability: Applies to the trip point settings that provide automatic protection against violating Safety Limits.

Objective: To assure protective action in time to avoid violating Safety Limits.

Specifications: The Limiting Safety System Settings and related information are given in table 1. During reactor operation, settings that are found to be less conservative than the values given in the table are not in violation if the setting is promptly corrected or the channel is placed in the tripped state.

Table 1 - Limiting Safety System Settings

<u>Service</u>	<u>Variable or Condition Monitored</u>	<u>Number of Channels</u>	<u>Trip Action</u>	<u>LSSS (2 loop)</u>	<u>LSSS (1 loop)</u>
Reactor power	Neutron flux	3	Scram on 2/3	≤ 50 MW	≤ 37,5 MW
Positive period	Neutron flux	3	Scram on 2/3	≥ 5 seconds	≥ 5 seconds
Primary coolant flow	Coolant flow and current in each of the 2 main pump motors	3	Scram on 2/3	≥ 15,000 gpm, loss of power to motors	≥ 10,500 gpm, loss of power to motors
Cover gas pressure	Pressure (low) " (high)	3 -	Scram on 2/3 Pressure relief valves open	≥ 180 psig ≤ 275 psig	≥ 180 psig ≤ 275 psig
Liquid level	Vessel liquid level	3	Scram on 2/3 Low-low logic activation on 2/3	≥ 209 inches ≥ 197 inches	≥ 209 inches ≥ 197 inches
Core inlet temperature	Temperature at vessel inlet	3	Setback on 2/3	≤ 55°C (131°F)	≤ 56.5°C (134°F)

Bases:Reactor Power and Period

The accidents that were considered in the design of the reactor safety system and in establishing the Limiting Safety System Settings were power transients initiated by a variety of postulated events leading to unexpected increases in reactivity (2.9, 2.10). In each of these accidents, scram is assumed to occur as a result of an overpower trip at 50 MW. While some of the analyses show the power exceeding the Safety Limit for very short time periods, it was concluded, because of the many conservative features of the calculation that little or no fuel damage would occur in any actual accident. Note that no credit was taken in the analyses for the fact that the actual reactor instrumentation gives a scram trip on a combination of power and period rather than power alone. The limit on period is included as a LSSS to provide increased protection. The 5 second value for a period trip at zero power permits a reasonable startup rate.

Trip action is initiated when a signal is received from any 2 out of 3 ionization chambers located in the thermal shield. These instruments are periodically calibrated in terms of power by means of heat balance instrumentation (flow and core ΔT).

In the single-loop, 30 MW operating mode, the LSSS on power was set at a value such that the ratio of Limiting Safety System Settings for the two operating modes is equal to the ratio of design power levels, i.e.

$$\text{LSSS (1-loop)} = \frac{30}{40} \times \text{LSSS (2-loop)}$$

Primary Coolant Flow Rate

The loss of flow accidents analyzed in the FSAR^(2.11) involved loss of pumping power and instrument system malfunctions. In cases where shutdown pumping was still available to remove the fission product afterheat, the only consequence was a slight rise in the fuel plate surface temperature before scram occurred. If the accident involved the loss of all forced flow cooling, the flow pattern in the core goes into the flow reversal and natural circulation cooling mode (see section 3.3).

The flow rate is sensed directly by a venturi element in the main coolant line and by devices which measure the current in each of the main pump motors. The use of the pump motor current to provide a trip signal is an improvement over the original system which relied on the opening of the pump breaker for the trip signal. The new system has both a high and low trip setting and provides protection against events which would not have caused trip action previously. These possible events include mechanical failure of the pump motor shaft, bearing failure and short and open circuits in the motor windings. It should be noted that these events would not have caused core damage with the original trip instrumentation unless both pumps were affected simultaneously. One pump has the capacity to maintain the flow rate well above the Safety Limit for 40 MW operation. However, the new system provides advance indication of certain types of failures and increases the margin of safety.

For purposes of defining the LSSS on pump motor current the specification does not go beyond the requirement that a channel trip occurs on loss of motor power. This wording retains the

essential performance capability of the original system while providing operating flexibility in adjustment of the trip setting values.

The loss of flow accidents referred to previously were analyzed in detail only for the two-loop 40 MW operating mode. The LSSS on flow for the single loop, 30 MW mode was established by employing the ratio method used for reactor power, i.e.

$$\text{LSSS (1-loop)} = \frac{11,800}{16,600} \times \text{LSSS (2-loop)}$$

Cover Gas Pressure

A number of loss of pressure accidents were analyzed in the FSAR^(2.12) and the most serious of these is a pipe break in one of the larger lines close to the reactor vessel or the accidental opening of a valve in one of the larger exhaust lines. Local boiling at the core hot spot may occur before reactor shutdown but no damage to the fuel results. Shutdown by scram action is initiated by signals from any two of the three cover gas pressure sensors.

With regard to the overpressure case, a small margin has been established between the Safety Limit and Limiting Safety System Setting. All the overpressure accidents considered^(2.13) involve a gradual rate of pressure rise which can easily be accommodated by the pressure relief valves with little overpressure.

Reactor Liquid Level

The loss of liquid level accidents analyzed^(2.14) are due to breaches in the primary system. The location and size of the rupture determines what protective actions are necessary. Small leaks that can be isolated and valved off would require only reactor shutdown. Sizeable breaks in the reactor vessel or

primary piping would involve both reactor shutdown and several additional steps to facilitate core afterheat removal. These additional steps which are initiated at a different level by the low-low level logic system are as follows:

- a. The syphon break valve and the 4 pump breakers open as a result of the low-low level trip.
- b. The depressurization valve is opened as a result of the opening of the pump breakers.
- c. The flow reversal valves open after the pumps are turned off and coast down.

Although all of the above events are triggered in the appropriate sequence when the lower LSSS is reached, the opening of the syphon break valve serves a protective function only if the break location was in the primary piping below the level of the transition plate in the vessel and the leak cannot be isolated. Its purpose is to minimize loss of coolant from the vessel by syphon action. Since the value of the pumps is uncertain in a pipe break situation, they are stopped and the flow reversal and natural circulation cooling mode begins. Both shutdown and low-low logic actuation are initiated by signals from liquid level sensors in 2 out of 3 coincidence. There is no damage to the core or to the primary coolant boundary other than the initiating event.

Core Inlet Temperature

Accidents that could cause an increase in the temperature level of the primary coolant are reduction or loss of secondary coolant flow to the main heat exchangers and fouling of these exchangers. Accidents of this type were not analyzed in detail in the FSAR because the resulting temperature transients were slow enough

to rely on operator action to shut the reactor down before the LSSS was reached. However, in connection with a recent study of HFBR instrumentation^(2.15), an analysis was done to determine the response of the system in the absence of operator action. It showed that in the event of a complete loss of secondary cooling the maximum temperature of the primary coolant is 159°F for two-loop 40 MW operation. The maximum temperature for the single loop case is lower because of the slower rate of temperature rise. In both cases, automatic reactor shutdown by setback action is initiated when signals are received from 2 out of 3 temperature sensors located upstream of the reactor vessel inlet.

3. LIMITING CONDITIONS FOR OPERATION

The technical specifications in this section cover the minimum functional capability or performance level of equipment required for safe operation.

3.1 Containment System

Applicability: Applies to the operating status of equipment associated with containment.

Objective: To ensure containment integrity when required.

Specifications: If containment integrity is required the following conditions must be satisfied.

- a. The building has passed its most recent leak rate test.
The test requires that the leak rate not exceed 6.2%/day at 0.34 psi overpressure.
- b. The Bypass Filter Facility is valved on line and the filters show the following minimum decontamination efficiencies on the most recent test.
  1. Particulate filters   90%; 0.2 micron particle
  2. Charcoal filters   90%; elemental iodine
- c. At least one truck lock and one personnel lock operable; at least one door sealed in each air lock.
- d. All penetrations sealed except building inlet and outlet ventilation lines.
- e. The Nuclear Incident Alarm System (NIAS) is operable.
In addition, alarm functions external to the reactor building may be taken out of service for maintenance or repair without violating the requirements of this specification.

Containment integrity shall be maintained at all times unless conditions specified by item (f) or (g) below exist.

- f. There is no fuel in the reactor vessel.
- g. All of following:

- (1) All control rods are fully inserted.
- (2) The primary system has been depressurized and cooled to ambient shutdown conditions and is fully sealed.
- (3) The reactor has been shut down for at least 12 hours.
- (4) The reactor shield CO₂ seals are sealed as required for full power operation.

Bases:

Containment integrity refers to the capability of the containment building and systems to serve as a final physical barrier against the excessive release of radioactivity. The conditions which define containment integrity are required specifically to limit on site and off site exposures following a maximum credible accident (3.1,3.2) and to minimize radioactivity releases from other accidents of significant magnitude.

The basis for each specification follows:

- Item a. The building leak rate test conditions are established to insure that the building leak rate will be at least as good as that assumed in the MCA analysis.
- Item b.1 No credit was taken for removal of particulate material from the air discharged to the stack during the MCA but the requirement is considered worthwhile.
- Item b.2 The overall iodine removal efficiency of the activated carbon filter is assumed to be 95% in the MCA analysis. However, the consequences of the accident are not, in fact, very sensitive to the assumed efficiency of the filters. This is due to the fact that leakage of radioiodine from the containment building at ground level, rather than discharge through the carbon filters

to the stack, is the controlling factor with regard to the 10 CFR 100 guidelines. For the most unfavorable meteorological case considered, a relatively rare weak neutral condition with an inversion lid above the top of the stack, the 10 CFR 100 guideline figures would still be met with a overall iodine removal efficiency of 84%; for other more frequent meteorological conditions, substantially lower removal efficiencies would be adequate. The fraction of the total radioiodine liberated that is in the organic form is expected to be less than 5%. If the carbon filters remove 90% of the elemental radioiodine and no credit is taken for removal of organic iodine, then the overall efficiency will be 85.5%, an acceptable figure.

- Item c. The air lock requirements are to ensure that personnel can leave the building and emergency equipment can be brought into the building as necessary.
- Item d. This requirement is to prevent leakage in excess of the allowable. Both ventilation lines are normally open but the inlet line will close automatically during an incident. Air leaving the building through the outlet ventilation line is filtered before being discharged from the stack.
- Item e. The nuclear incident alarm system performs several functions. First, it monitors the gamma dose rate in the reactor building and alerts the reactor operator to high gamma levels. In the event of abnormally high

dose rates the system automatically initiates building isolation actions which include (1) shutting down the 20,000 cfm building exhaust blower, (2) closing the building air inlet valves and (3) closing the condensate drain valves in the inlet air handling units. The system also automatically transmits alarms to the Laboratory Police and Fire Headquarters. The temporary removal from service of the automatic alarm functions is permitted because these functions are backed up by separate manually operated alarms originating in the control room. In addition, the police and fire departments as well as the Laboratory administration can be alerted by telephone.

The radiation detection equipment consists of 3 identical channels with detectors location on the balcony level. Signals from any 2 of the 3 channels will cause trip action.

Item f. The condition of no fuel in the vessel clearly eliminates the potential for any release of fission products. The possibility of release of tritiated water from a spill still exists but this is a relatively minor hazard since there is no afterheat source available to raise the temperature of the water and cause evaporation.

Item g. This specification consists of 4 requirements. Requirement (1) ensures that no reactivity accidents can occur. The second requirement reduces the potential for release of radioactivity from the primary containment. Requirement (3) ensures that there will be insufficient

afterheat remaining in the core to cause boiling in the absence of secondary cooling. The fourth requirement guarantees the integrity of thermal shield as a coolant barrier in the event of a breach of the reactor vessel.

The exceptions to the need for full containment integrity, which have been approved by the Commission, allow modifications and maintenance to the containment itself and provide easy access to the building when major work on other parts of the plant might be required. These exceptions also permit exceptionally large experimental equipment to be brought in the building.

3.2 Reactivity Control

Applicability: Applies to reactivity control and shutdown requirements.

Objective: To ensure adequate reactivity control under both accidental and normal operating conditions.

Specifications: The reactor shall not be operated unless:

- a. The reactor can be held subcritical when all auxiliary rods and two main rods are withdrawn and the reactor is at ambient conditions.
- b. The rate of control rod withdrawal does not exceed 4.46 inches/minute.
- c. At least 6 of the 8 main control rods have a scram time, from the fully withdrawn position to 12-7/8 inches insertion, no greater than 175 msec, including the release time of the clutches.
- d. Each of the 16 control rods is capable of being moved to any position in its normal range of travel.

Bases:

Specification a. is to ensure that the "stuck rod criterion" is satisfied; that is, that subcriticality can be maintained in the event of failure of the most effective rod. In the HFBR, control rods are placed in an essentially symmetric pattern around the core and therefore will have approximately equal worths. The specification of 2 main rods withdrawn rather than only one is to cover the possibility of small variations in individual rod worths and to provide an additional safety margin.

The purpose of specification b. is to ensure that a reactivity accident due to the ganged withdrawal of either the main or auxiliary

rod bank does not cause core damage. An interlock prevents the withdrawal of both banks simultaneously. The 4.46 in./min. figure was derived from an analysis^(3.3) of a control rod bank withdrawal accident in which it was assumed that reactivity was added at a constant rate of 12.5¢/sec. to a critical reactor. The maximum differential rod worth, as determined by measurements made during the startup period with a core consisting of the original KD type elements was 1.56\$/in. for a rod bank^(3.30). Flux and reactivity calculations in which the original KD and current KL type cores were compared, indicate that the differential rod worth at the critical position of a freshly fueled core is 7.5% greater for the KL type core^(3.31). Modifying the 1.56\$/in. figure to account for this effect yields a differential rod worth for the current design of 1.68\$/inch. Combining this result with the 12.5¢/sec. limit yields the 4.46 in./min. specification. It should be noted that the accident analysis indicated that the consequences would be minor, involving a slight overshoot of the LSSS and no fuel damage.

Specification c relates to the ability of the nuclear safety system to safely terminate a fast nuclear transient. In the various reactivity accidents analyzed, the rate of negative reactivity insertion due to scram action was based on rod worth curves developed in critical facility measurements and scram test curves on prototype rod drive mechanisms. These measurements indicated that the overall control system has the capability of inserting 4.4% k in control rod reactivity effect in 181 msec. including instrument response time^(3.4). The scram motion of the drives is 12-7/8 in. from the fully withdrawn position in 175 msec., including the release time of the clutches but not including 6 msec for instrument

response time. In the analyses, it was assumed that only 4 of the main rods responded with a negative reactivity insertion of 2.2% k. However, it was noted that the calculated transients do not depend strongly on the number of rods assumed to act. Transients shut down by one rod were only slightly more severe than those shut down by all eight. The actual worth of a bank of 8 rods over the first 12-7/8 inches of travel was found, during the reactor startup period,^(3.30) to be about 75% greater than the 4.4% k figure which is based on critical facility measurements. This result, strictly speaking, applies only to the original KD type core, but calculations indicate that the KL core is not greatly different with regard to rod worth effects.^(3.31, 6.2) It is evident, therefore, that there is ample rod reactivity reserve under the specification c requirement.

The purpose of specification d is to provide the flexibility necessary to (1) balance control rod positions and thus minimize perturbation of the neutron flux in the core and experimental facilities and (2) achieve the full burnup potential of the fuel elements by allowing complete withdrawal of control rods at the end of the fuel cycle.

3.3 Shutdown and Emergency Cooling

Applicability: Applies to the availability of normal and emergency cooling during shutdown.

Objective: To ensure availability of core cooling during shutdown.

Specifications: The reactor shall not be started up unless the following conditions exist:

- a. The flow reversal system is operable, i.e., on loss of both main and shutdown pumps the depressurization valve and at least 3 out of 4 flow reversal valves will open.
- b. With main pumps off, a primary coolant flow rate through the reactor vessel of at least 800 gpm.
- c. With main pumps off, a primary flow rate through the shutdown exchanger of at least 350 gpm.
- d. Secondary flow rate through shutdown exchanger of at least 80 gpm.

Bases:

A shutdown pump and heat exchanger provide for the removal of fission product afterheat from the core when the main pumps are not operating. With loss of both main and shutdown pumps all forced flow cooling is lost and the core flow pattern will go into the flow reversal and natural circulation mode of after-heat removal. The opening of four valves in the transition plate above the core provides a return path for natural circulation flow. These valves are normally held closed by the discharge pressure developed by any one of the main or shutdown pump. The valves open automatically by spring action when the pump pressure is no longer available. Although 4 valves are available, only 3 are essential for the safe reversal of flow^(3.5).

The loss of all forced flow cooling also causes the primary system to be depressurized through a vent valve in the cover gas system. The depressurization was done because calculations indicated that the margin to burnout in flow reversal and natural circulation is somewhat greater with the resulting static head than at full system pressure. Tests which simulated the flow reversal and natural circulation regime following loss of flow, indicated that no fuel damage will occur^(3.7).

The specification of a shutdown flow rate (item b) ensures that flow instability does not occur in the hot channels in the core immediately following an emergency shutdown due to loss of the main pumps^(3.6). The shutdown flow required to prevent instability actually decreases as the afterheat rate decreases but no allowance is made for this effect in the specification. The specifications on flow rates in the shutdown cooler (items c and d) also cover cooling requirements following an emergency shutdown due to the loss of the main pumps. They ensure that boiling does not occur in the core^(3.6, 3.7).

3.4 Primary Coolant and Cover Gas Chemistry

Applicability: Applies to the chemical conditions in the primary system coolant and cover gas.

Objective: To ensure that proper chemical conditions are maintained in the primary system.

Specifications:

- a. The primary coolant and experimental facility coolant shall be maintained within the following limits.
  1. Isotopic purity: 95% D₂O (minimum)
  2. pD^{*}: The normal limits are 4.7 to 5.5 except that levels outside this range as far as 4.0 and 7.4 are not in violation if either (a) or (b) exist.
    - (a) The reactor is operating and the out-of-range condition does not exist longer than 24 hours during any one operating cycle or
    - (b) The reactor is shutdown
  3. Conductivity: 30 micromho/cm (maximum) corrected to 25°C.
  4. Chloride: 1 ppm maximum except that higher concentrations up to 10 ppm are not in violation if the condition does not exist longer than 24 hours in any one operating cycle.
- b. Deuterium content of helium cover gas: 4% (maximum)

Bases:

Isotopic Purity: The specification is based on reactor physics considerations. Starting with high purity D₂O (99.75%), an increase in H₂O content will cause a reactivity decrease up to H₂O concentrations of about 5%. Further increases in H₂O will cause the reactivity to increase monotonically until the maximum reactivity

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* pD of neutral D₂O is 7.4. The pD limits are not applicable to the experimental facility coolant.

is reached at 100% H_2O . At that point, with 100% H_2O in the core and reflector, the reactor is supercritical with all safety rods inserted^(3.8).

pD and conductivity: These specifications relate to corrosion of aluminum in the primary system. The system is maintained at a pD close to 5 because the corrosion rate in aluminum is a minimum at this level^(3.9). The method of control described in the FSAR^(3.10) has been modified because of difficulties experienced with the early approach. The current method is to operate with a sidestream of the primary water passing through a mixed bed deionizer and to periodically add nitric acid as necessary to replace the acid removed by the deionizer. Since the quantity of impurities (other than nitric acid) in the water is maintained at a low level by operation of the deionizer, the pD of the water can be inferred from its conductivity. On-line pD instrumentation has not proved dependable and reliance is placed on the conductivity instrumentation for continuous control and monitoring of pD. Grab samples of the primary water are taken periodically for backup measurement of pD and conductivity in the laboratory.

At the outer limit values of pD listed in the specification (4.0 and 7.4) the corrosion rates are several times higher than in the normal pD range. However, if these out of range conditions are limited to a small fraction of the total operating cycle (30 days) the overall effect on aluminum corrosion is small.

Although the pD during shutdown is normally maintained at the same level as during operation, control during shutdown conditions is less important because of the lower temperatures and consequently lower corrosion rates^(3.11).

The conductivity specification (30 micromhos/cm) corresponds to the lower pD limit of 4.0.

Chloride: Chloride concentrations are limited in order to minimize the potential for stress corrosion of stainless steel and pitting of aluminum. The chloride levels are normally below 0.1 ppm. However, in view of the relatively low temperatures in the primary system (less than 150°F) short term exposures up to 10 ppm are not considered detrimental. Chloride levels are monitored periodically by grab sample analysis.

Experimental facility coolant: The preceding comments in general also apply to the experimental facility D₂O except that this system is not acidified and therefore no pD limits apply. The water is normally polished continually by an ion exchange column so that the purity is quite high. The pD of this water is essentially equal to the pD of pure D₂O (7.4), however, in high purity water the pD is not a very good measure of the water quality because of the sensitivity to contamination.

Cover Gas Deuterium Content: The explosion limit of hydrogen-oxygen mixtures diluted in helium or in helium and water vapor is in the range 9-15% hydrogen^(3.12). Control of the deuterium and oxygen formed by corrosion and by radiolytic decomposition is normally accomplished by circulation of the cover gas through a recombiner. Oxygen is added to the gas stream before it enters the recombiner to make up for oxygen taken up by corrosion reactions. Under these conditions with helium recirculation and the recombiner functioning, the deuterium concentration is maintained below 0.08%. If either the recombiner or recirculator fails, the normal helium purge

will keep the deuterium level below 1%^(3.13). This purge is provided by the liquid level bubbler streams which supply a minimum of 0.13 scfm of helium^(3.14). Monitoring of the deuterium and oxygen concentration is provided by an on-stream chromatographic instrument. Backup, grab-sample analysis is available.

3.5 D₂O Heat Exchangers

Applicability: Applies to process condition in heat exchangers containing D₂O.

Objective: To prevent D₂O freezing and H₂O leakage in heat exchangers containing D₂O.

Specifications:

- a. The secondary water temperature entering the main, shutdown and experimental facility heat exchangers shall be maintained above 5°C.
- b. In the main and shutdown exchangers, the pressure of the D₂O side of each exchanger shall be at least 5 psi higher than the H₂O side unless the H₂O side is isolated.

Bases:

Freezing may occur in the primary side of the exchangers if the secondary water is circulated at a temperature below 3.79°C, the freezing point of D₂O. Low temperatures in the secondary system are of concern only in very cold weather with the reactor shut down. However, there are a number of operating procedures which can be employed to maintain safe temperatures^(3.15). These include bypassing the cooling towers, and running the secondary pumps.

The primary system is maintained at a higher pressure than the secondary system to prevent in-leakage of light water. The points of possible contact are the main exchanger EA-101A and EA-101B and the shutdown cooler, EA-103. During normal operation with the primary system pressurized, the pressure difference in the exchangers is about 200 psi. When the primary system is depressurized at shutdown, the margin is smaller but still above 10 psi^(3.16). Improper operation of the valving on the secondary

side of the exchanger could cause a pressure reversal when the primary system is depressurized. This possibility is covered by administrative control and by instrumentation which continuously monitors the pressure difference and gives an alarm in the control room if the difference falls below 10 psi.

The D_2O experimental facility cooling system removes heat generated in the fast and thermal irradiation thimbles. The system operates close to atmospheric pressure because of the need to insert and remove experiments during operation. Consequently, the pressure of the D_2O in the cooler is normally less than the H_2O pressure. Large leaks into the D_2O would be signalled by liquid level instrumentation and would be alarmed in the control room. Small leaks would be detected by periodic isotopic purity determinations.

3.6 Poison Solution System

Applicability: Applies to the reactor poison solution system.

Objective: To provide an adequate shutdown margin in the event of light water flooding of the reactor.

Specification: The quantity of $\text{Cd}(\text{NO}_3)_2 \cdot 4\text{H}_2\text{O}$ in solution in the poison water tank shall not be less than 2700 lbs.

Bases: A supply of cadmium nitrate solution is stored in a vessel for use as an ultimate shutdown agent. The poison solution vessel was originally charged with approximately 3300 lbs of $\text{Cd}(\text{NO}_3)_2 \cdot 4\text{H}_2\text{O}$. However, only about 1/3 of this amount is needed to maintain subcriticality in the event of light water flooding^(3.17). The chemical content and liquid level in the storage tank is checked periodically.

3.7 Shield Cooling Systems

Applicability: Applies to the thermal and biological shield cooling systems.

Objective: To ensure adequate cooling of the thermal shield, shutters and beam plugs.

Specifications: The reactor will not be operated unless the following conditions exist:

- a. The thermal shield flow rates listed below are minimum values except that a 30 second outage is allowed to switch pumps:
  1. lower thermal shield - 350 gpm
  2. upper thermal shield - 50 gpm
  3. cooling coils - 15 gpm
- b. Cooling water outlet temperature from thermal shield: 65°C (maximum)
- c. The biological shield flow rates listed below are minimum values except that a 30 second outage is allowed to switch pumps:
  1. Shutter flow - 50 gpm
  2. Beam plug flow - 40 gpm
  3. Biological shield cooling coils - 15 gpm
- d. Cooling water outlet temperature from biological shield 65°C (maximum).

Bases:

The normal flow rate in the thermal shield is 830 gpm. Of this, 700 gpm is supplied to the lower thermal shield, 100 gpm to the upper thermal shield and 30 gpm to the cooling coils in the shield. The design analysis^(3.18) based on these flow rates,

indicates temperature distributions and stress levels well within acceptable values at 40 MW. Furthermore, temperature surveys conducted at the beginning of reactor operations and daily heat balances on the thermal shield cooling system show that the radiation heating rates assumed in the analysis were very conservative, being high by a factor of 2-3. It is evident from these considerations that the individual flow rates could be safely reduced by 50% to the specification values.

The exception to the specification in which a 30 second outage is allowed will not cause an unsafe condition because the reactor goes into setback on loss of flow. Setback will continue until flow is reestablished.

The temperature specification is to ensure that enough secondary cooling is provided to prevent boiling in the shield. The bulk temperature rise of the water due to 0.5 MW of heat generation in the thermal shield at 40 MW and a total flow of 415 gpm is about 8°F.

A liquid level requirement in the thermal shield cooling system is not essential since a loss of coolant would become apparent on the flow and temperature instruments.

Cooling water flow from the biological cooling system is required while the reactor is operating to prevent overheating of the shutters and beam plugs^(3.19). Flow is also supplied to a cooling coil array at the inner edge of the biological shield although the stress levels in the biological shield are satisfactory without cooling^(3.20).

Without coolant flow the water in the shutters and beam

would eventually boil. At the flow rate specified, the bulk water temperature rise is less than 1°F. The comments dealing with liquid level and pump outages in the thermal shield cooling system also apply to the biological shield system.

3.8 Radiation Monitoring

Applicability: Applies to the radiation monitoring of the facility.

Objective: To ensure adequate radiation monitoring.

Specifications:

a. The reactor shall not be operated unless the following requirements are satisfied.*

1. The fuel cladding failure safety system is operable (RRa-101, RRa-102, RRa-103).
2. At least one of the following instruments is operable:
  - (a) Reactor cover gas activity monitor (RRa-105).
  - (b) Exit air gas activity monitor (RRa-306).
  - (c) Primary water activity monitor (RRa-100).
3. All of the following secondary water gamma activity monitors are operable.
  - (a) Primary cooler EA-101A effluent (RRa-301A)
  - (b) Primary cooler EA-101B effluent (RRa-301B)
  - (c) Shutdown cooler EA-103 effluent (RRa-302)
  - (d) Experimental facilities cooler EA-102 effluent (RRa-303).

In the event any of the above instruments are out of service the tritium activity monitoring system can be used to monitor the effluent from the affected exchanger.

b. Whenever secondary water is being discharged to the ground the tritium activity of the secondary water shall be monitored at least once a shift.

Bases:

The fuel cladding failure system^(3.21) provides a means for the rapid detection of large fission product releases in the

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*A functional equivalent to each instrument listed may be substituted for that instrument without violation of this specification.

primary coolant. The system was not part of the original instrumentation; it was incorporated into the radiation monitoring system to provide a more rapid detection and response than could be achieved by the existing fission product monitors.* The sensing elements are 3 ionization chambers located around the reactor vessel outlet pipe. The system achieves its rapid response characteristics at the cost of sensitivity; fission product levels in the water much smaller than the normal ^{16}N background activity cannot be detected. Reactor scram occurs automatically on signals from any 2 of the 3 ionization chambers.

The instruments listed under specification a.2 are more sensitive to fission product activity than the fuel cladding failure system, but will respond more slowly. The response times of these monitors are of the order of minutes as compared to seconds for the fuel cladding failure system^(3.22). The monitors send signals to the control room alarm when a preset value is exceeded.

Specifications a.3 and b. provide a means of detecting primary to secondary leaks in exchangers containing D_2O . The gamma monitors^(3.23) will see ^{16}N and ^{24}Na activity although the ^{16}N predominates during operation. The gamma monitors are useful only for a short period after shutdown because the ^{16}N disappears at shutdown and the ^{24}Na activity is greatly reduced by decay and by removal in the purification system.

The on-line tritium monitors^(3.24) are more sensitive to primary-to-secondary exchanger leaks than the gamma monitors and also provide coverage during shutdown when the gamma monitors

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* See items under specification 3.8.a.2

are not useful. However, the response time of the tritium monitors is about 10 minutes as compared to 10-30 seconds for the gamma monitors. As a result, a greater amount of tritium leakage to the secondary water would occur, following a tube break in a heat exchanger, before operator action to isolate the exchanger could be initiated. Although a holdup tank located between the cooling tower basin and discharge point has sufficient capacity, so that further steps could be taken in time to prevent discharge of the tritiated secondary water to the ground, it would be desirable to isolate the exchanger as quickly as possible. In view of these considerations, the gamma monitors were specified as a requirement during operation, with a tritium monitor as an acceptable backup in the event any of the gamma monitors have to be taken out of service for repair.

For possible heat exchanger leaks below the detection limit of the gamma monitors and for leaks during shutdown, some form of tritium monitoring is required. Tritium determinations performed at least once a shift are acceptable if no on-line tritium monitoring is available^(3.25).

3.9 Reactor Shutdown Safety System

Applicability: Applies to all systems that can initiate automatic or manual shutdown of the reactor.

Objective: To ensure shutdown of the reactor when required.

Specifications: The reactor shall not be operated unless each shutdown function is operable as shown in Table 1, Section 2.2 and Table 2, this section. During operation, settings that are found to be less conservative than the limiting values are not in violation if the setting is promptly corrected or the channel is placed in the tripped state.

Table 2 - Shutdown Functions Other Than LSSS

<u>Service</u>	<u>Variable or Condition Monitored</u>	<u>Number of Channels</u>	<u>Trip Action</u>	<u>Trip Setting</u> (both operating modes)
1. Core pressure drop	Pressure drop across reactor vessel	3	Scram on 2/3	Low Δp equivalent to LSSS on coolant flow
2. Fuel cladding failure	Gamma activity in primary coolant at reactor outlet	3	Scram on 2/3	*Activity equivalent to 2 times the normal level due to the ^{16}N background
3. Shutdown pumps	Power to each of 2 pumps	2	Setback on 2/2	Loss of pump power
4. Thermal shield cooling	Flow in lower thermal shield and power to each of 2 pumps	3	Setback on 2/3	≥ 400 gpm and loss of pump power
5. Biological shield cooling	Flow in shutters and power to each of 2 pumps	3	Setback on 2/3	≥ 60 gpm and loss of pump power
6. Manual scrams	7 manual scrams and console lockdown switch operable.			

*The ^{16}N content of the primary water at the detector location is approximately $40 \mu\text{Ci/cc}$.

Bases:

The bases for the shutdown specifications identified as LSSS in Table 1 are given in Section 2.2. Specifications 2, 3 and 4 in Table 2 are discussed in Sections 3.3, 3.7 and 3.8. The remaining items 1 and 6 are covered here.

The trip function on low core differential pressure provides additional protection against loss of coolant flow. The trip setting in each operating mode corresponds to the core pressure drop at the LSSS for flow.

Two manual scram buttons are located on each floor of the reactor building and one at the reactor console in the control room. The console lockdown switch is a 2 position, key operated, switch. In the "off" position the key may be removed, thereby effectively locking the control rods in the fully inserted position^(3.26).

3.10 Standby Electrical Power

Applicability: Applies to standby electrical power sources.

Objective: To ensure availability of standby electrical power.

Specification: The reactor shall not be operated unless the 250 VDC battery system is operable.

Bases: In the event of a loss of the commercial AC power source, the reactor is safely shut down without reliance on standby power sources^(3.27). Removal of fission product afterheat is accomplished by shutdown pumps which, in the absence of the normal AC source, can be powered by a 250 VDC battery bank or a generator driven by a propane engine. In the event all three electrical sources fail, the flow reversal and natural circulation cooling mode begins; the core is adequately cooled and no fuel melting results.

As a matter of operating convenience, it is desirable to have a standby source available so that the reactor can be restarted rapidly following restoration of commercial power. Although surveillance is performed on both the battery bank and generator, only the battery bank is called for in this specification since it is the more reliable of the two sources.

3.11 Spent Fuel Storage

Applicability: Applies to the handling and storage of fuel elements in the canal.

Objective: To ensure safe handling and storage of fuel elements.

Specifications:

- a. The water in the canal shall be maintained at a level sufficient to keep all spent fuel being stored in the canal submerged under at least 5 ft. of water.
- b. No more than five fuel elements shall be stored outside of critically safe containers.

Bases: The spent fuel elements are stored vertically in racks at the bottom of the canal which is 20 feet deep. Since the water level is normally maintained within a foot of the canal top, there is approximately 14-15 feet of water above the top of the uncut element (57 in. high) in the storage rack. This margin provides sufficient flexibility for fuel handling operations without significant radiation exposures to personnel. However, for the purpose of assuring the integrity of the fuel element cladding and avoiding a fission product release, it is only necessary to adequately cool the elements. Maintaining the fueled section of the fuel elements submerged in water provides this assurance. A 5 ft. margin is specified to allow for level changes due to evaporation and possible leakage. This is a very substantial margin since operating experience indicates that the rate of level change is about 2 inches per week.

Specification b. is intended to define the limiting value on the number of fuel elements that can be safely brought together in the canal. Physics calculations performed in connection with the fuel element modification^(6.2) indicated that 5 fresh elements arranged in the most reactive configuration will be subcritical.

3.12 Main Pump Start Interlock

Applicability: Applies to instrumentation related to main pump operation.

Objective: To prevent operation of main pumps under potentially hazardous conditions.

Specification: The reactor shall not be operated in the single loop mode unless the main pump start interlock is operable.

Bases: The pump start interlock requires the reactor to be shut down before either of the main pumps can be started. It is needed only during 30 MW one pump operation to prevent starting a second pump and introducing cold water into the reactor vessel while the reactor is operating. It should be noted that a postulated accident was analyzed in which this interlock failed and a cold loop was started up while the reactor was running. The reactor would have been shut down by scram action with no serious consequences (3.29).

3.13 Cold Startup Instrumentation

Applicability: Applies to special instrumentation required for cold core startup.

Objective: To ensure adequate flux monitoring during cold startups.

Specification: A counting rate system and a detectable neutron source will be used on all startups where there is no indication of flux level on the normal power level instrumentation.

Bases: During normal startups with 14 partially burned fuel elements in the core, there is sufficient photoneutron activity to cause an on-scale reading on the normal instrumentation. However, on startups following a complete refueling of the core there may not be enough activity from structures in the reactor vessel, especially if the reactor had been shut down for an extended period of time. In this circumstance, a neutron source and one of the counting rate channels will provide the necessary flux indication^(3.28).

4. SURVEILLANCE REQUIREMENTS

4.1 Containment System

Applicability: Applies to the containment building and containment systems.

Objective: To assure that the containment building and containment systems function as specified in Section 3.1.

Specifications:

- a. A containment leak rate test shall be performed prior to initial operation, and then at a minimum frequency of semi-annually the first 2 years, annually the next 2 years and every 2 years thereafter. If the maximum allowable leak rate (6.2% per day at 0.34 psi overpressure) is exceeded in any test, the test frequency shall revert to the initial frequency.
- b. Any additions or modifications to the containment building and its components that could significantly increase the leak rate shall be tested to verify that the required building leak-tightness can be maintained.
- c. The filters in the Bypass Filter Facility shall be tested for particulate and iodine removal efficiency at least annually.
- d. The nuclear incident alarm system shall be tested between fuel cycles.

Bases: Since reactor operation began, the interval between successive containment tests has not exceeded one year. The results of these tests and the experience developed in maintaining the containment structure provide assurance that the test frequency could be safely reduced to once in 2 years provided the testing and maintenance requirements noted in Specification b are also satisfied.

The filters in the bypass filter facility have been tested annually since their installation 6 years ago. The particulate filters have shown little deterioration while the charcoal filters have begun to show a reduction in iodine removal ability due ostensibly to "weathering" by the NO_2 in the air. The rate of deterioration of the charcoal filters is such that continuation of the annual test frequency should provide adequate advance indication of when filter replacement is necessary. The annual test will also be retained for the particulate filters even though it is apparent that a lower test frequency would be satisfactory.

The entire nuclear incident alarm system cannot be conveniently tested during reactor operation because of its effect on building ventilation and therefore testing between operating cycles is specified. This test frequency together with the 2 out of 3 logic employed in the system assures more than adequate reliability.

4.2 Reactivity Control

Applicability: Applies to reactivity control and shutdown requirements.

Objective: To assure that reactivity control is available as specified in Section 3.2.

Specifications:

- a. A test to determine the minimum number of main rods required to maintain subcriticality shall be conducted at least annually and whenever a main rod is replaced.
- b. A test to determine the scram time from the fully withdrawn position of each main rod shall be done at least quarterly.
- c. Each time a main rod is replaced, the new rod will be tested for scram time and reactivity worth.
- d. A measurement of the withdrawal speed of each rod shall be made at least biennially. If any significant part of the drive train of a rod is replaced, a test of the individual rod withdrawal speed shall be made.
- e. The control rod gang withdrawal interlocks shall be tested at least quarterly.

Bases:

The critical rod position and the number of rods required to maintain subcriticality will vary with rod burnup, core loading, time in cycle and number and type of experiments. The most reactive core condition is a freshly loaded core with no experimental poisons present. However, this condition occurs very rarely because (1) the normal refueling scheme involves a half core rather than full core replacement and (2) some experiments are of a semi-permanent nature and cannot be conveniently removed. Thus, the most reactive core condition

normally encountered occurs at the start of a cycle after refueling has taken place with the core at ambient temperature. The specification a. test will be conducted under this condition. The reactivity worth of experiments, which are determined before the experiments are inserted in the reactor, will be recorded as part of the specification a. test results. With regard to the test frequency, experience has shown that the rate of change of rod reactivity worth due to burnup is sufficiently low to permit annual verification. The rate of loss, about 1-2% per year, is small compared to the margin available^(4.1).

The rod drive mechanisms are nominally constant speed devices which are affected by variations in power supply frequency and by mechanical wear. The frequency of the commercial power supply is routinely maintained to within 0.1% so that variations from this source will be negligible^(4.2). Mechanical wear in the drive mechanisms will likely appear as a reduction in rod withdrawal speed rather than an increase. Since a reduction in rod speed is conservative with respect to the control rod withdrawal accident, testing on a biennial basis is considered adequate.

Because of their greater importance, the rod scram time tests are conducted more frequently than the normal speed tests. Experience has shown that an annual test is more than adequate to provide assurance that the shutdown requirements are met. However, as a result of recommendations made by the Commission in its 1973 annual appraisal, a quarterly test is specified.

The control rod withdrawal interlock tests are done on a quarterly basis only because the tests can be conveniently fit

in with other instrument maintenance that is done on the same schedule. Considering the relatively minor consequences of of system failure and the fact that no failures have been observed to date, it is apparent that these tests could be performed less frequently.

4.3 Shutdown and Emergency Cooling

Applicability: Applies to the surveillance of systems required for normal and emergency cooling during shutdown.

Objective: To assure reliability of systems required for normal and emergency cooling during shutdown as specified in Section 3.3.

Specifications:

- a. The entire low-low liquid level logic system with the exception of the flow reversal valves shall be tested at least quarterly.
- b. Operability of the flow reversal valves will be checked between fuel cycles.
- c. The flow rates on the primary and secondary side of the shutdown loops shall be checked between fuel cycles to assure that these variables are within specified limits before startup.
- d. Calibration of the flow instrumentation associated with the shutdown loop shall be done at least annually.

Bases:

A test of the low-low liquid level logic system also constitutes a test of the flow reversal system since the latter is a subsystem of the former. During this test, however, it is not possible to observe the functioning of the flow reversal valves since the vessel is closed and there is no external indication of valve position. Operability of these valves is checked when the vessel has been opened for refueling.

The frequency of testing specified for the low-low logic system is consistent with surveillance specified for similar equipment in the Reactor Shutdown Safety System.

A number of the original flow reversal valves were not completely satisfactory because the valves had occasionally become stuck in the closed position when tested. Although these valves have been replaced and no difficulty has been experienced with the new valves, it was deemed prudent to retain the original test frequency of once per shutdown.

The flow rates on the primary side of the shutdown loop are verified only during shutdown because there is no significant flow in these lines when the reactor is operating and main pumps running. The flow on the secondary side of the shutdown exchanger could be checked on a more frequent basis than specified but the low flow alarm on this system provides adequate protection between tests.

The annual frequency specified for the calibration of the flow instrumentation is consistent with manufacturers' recommendations and operating experience.

4.4 Primary Coolant & Cover Gas Chemistry

Applicability: Applies to the chemical conditions in the primary coolant and cover gas.

Objective: To ensure that chemical conditions are maintained as specified in Section 3.4.

Specifications:

- a. Measurements of the isotopic purity of the primary and experimental D₂O systems shall be performed at least monthly.
- b. Measurements of conductivity or pD of the primary system shall be performed at least daily while the reactor is operating and during shutdown, if pD changes are being made through acid addition or resin bed operation. At other times during shutdown the measurement shall be performed at least weekly. A pD or conductivity measurement shall be made prior to reactor startup.
- c. A measurement of primary system pD independent of the conductivity measurement specified in b shall be made at least monthly.
- d. Measurements of chloride content of the primary and experimental facility systems shall be made at least monthly.
- e. Measurements of the deuterium and oxygen content of the primary system cover gas shall be made at least weekly while the reactor is operating.
- f. Measurements of the conductivity of the experimental D₂O system shall be performed at least daily while the reactor is operating and at least weekly during shutdown.

Bases:

The principal cause of light water contamination of the primary system is exposure to the atmosphere during shutdown when the system is opened for fuel changing and occasional

maintenance. During operation, light water contamination is not credible since the system is sealed and at a positive pressure with respect to sources of light water contamination. Procedures for the control of contamination when the primary system is open have been refined to the point where the rate of change of D_2O content is now less than 0.02% per year as compared to the 0.05% per year experienced at the beginning of operations. With this in mind, a monthly analysis of isotopic purity should be more than adequate. This frequency roughly coincides with the frequency of fuel changing operations. Although there are no reactor hazards associated with light water contamination of the experimental facilities system, periodic analysis is specified for programmatic reasons.

While the reactor is operating, the conductivity of the primary water is normally monitored continuously by an on-line instrument. However, the response of the primary system to acid addition and deionizer operation is sufficiently predictable to permit daily conductivity measurement as an alternative to on-line monitoring when conditions require it.

During shutdown, operation of the resin beds and acid addition is generally suspended. Thus with the reactor shut down and radiation induced reactions no longer occurring, there is no mechanism for major pD changes to occur and the need for monitoring is accordingly reduced.

Although conductivity has proved to be a very reliable measure of pD in the primary system, an independent measurement performed on a monthly basis is also specified (Specification c). This measurement can be made on a laboratory pH meter or by chemical analysis for nitrate and nitrite.

Monitoring of the experimental facilities cooling system is also normally done on a continuous basis. However the requirements are less restrictive than for the primary system because no pD control is practiced. A sidestream deionizer continually polishes the water which is at a neutral pD. On the other hand there exists the possibility of ionic contamination of the experimental system by secondary-to-primary heat exchanger leakage and leakage from individual experiments. To cover this possibility, a daily measurement is specified as an alternative to on-line monitoring if conditions require it. During shutdown, the potential for contamination of the experimental system is lower and therefore the monitoring frequency has been reduced for this period.

The chloride level in the primary system has been maintained below 50 ppb since reactor operations began. This is close to the lower limit that is possible with cleanup by mixed bed deionizers. Since there are no sources of chloride in the primary system, contamination can occur only during shutdown when the system is opened. The monthly analysis that has been specified coincides with the normal fuel cycle and should provide adequate coverage.

The probability of contamination of the experimental facilities system is not great but is more likely to occur than in the primary system. However, the conductivity measurement provides a direct indication of excessive chloride levels and therefore a monthly chloride analysis should be adequate.

The reactor cover gas is normally monitored by an on-line chromatograph which automatically records the concentration of deuterium, oxygen, nitrogen and carbon dioxide at 10 minute intervals. However, monitoring is not essential from a reactor

safety viewpoint, because deuterium-oxygen concentrations will be maintained well below the explosive limit even if the recombiner fails (see section 3.4). To cover the possibility of unusually high deuterium generation rates due to unforeseen events a weekly measurement is specified.

4.5 D₂O Heat Exchangers

Applicability: Applies to instrumentation measuring (1) the secondary water inlet temperature to each D₂O heat exchanger and (2) the differential pressure between the primary and secondary sides of each primary coolant system D₂O heat exchanger.

Objective: To maintain operating conditions relative to the main, shutdown and experimental facility D₂O heat exchangers as specified in Section 3.5.

Specifications:

- a. The following shall be checked at least daily:
  - (1) The secondary water inlet temperature to the D₂O heat exchangers.
  - (2) The pressures on the primary and secondary sides of the main and shutdown heat exchangers.
  - (3) The liquid level in the experimental facilities coolant system.
- b. Instrument systems measuring the above variables shall be calibrated at least annually.

Bases: Daily verification of the above measurements is considered adequate since a control room alarm is actuated in the event an abnormal condition exists. Alarms are set above or below the specified limits as needed to provide sufficient margin for initiation of corrective action.

Experience has shown that an annual calibration frequency is adequate. In addition, any fault that affects the accuracy of one instrument would be readily uncovered by reactor operators who frequently record and compare instrument readings.

4.6 Poison Solution System

Applicability: Applies to reactor poison solution system.

Objective: To assure that the poison solution system contains the required quantity of nuclear poison as specified in Section 3.6.

Specifications:

- a. The cadmium concentration in the poison solution tank shall be measured at least annually.
- b. The liquid level in the poison solution tank shall be checked at least annually.
- c. An operability check of the poison solution dump valves shall be performed biennially.

Bases:

No significant change has been observed in the cadmium nitrate content of the poison solution storage tank since the tank was first filled during the reactor startup period. This is not surprising since the conditions of storage are mild (70°F, inert gas cover) with respect to the potential for corrosion of the stainless steel storage vessel. In addition, the lines connecting the storage tank to the primary system contain 3 valves in series to minimize the possibility of leakage. There is also a connection from the tank to the off gas system but this line is normally valved off to prevent evaporation. In view of these considerations, it is believed that the annual frequency specified is more than adequate.

Until recently, no routine test of the operability of the dump valves has been performed because such a test could not have been done without major disassembly or discharge of the solution into the primary system. However, the system has now

been modified to make testing feasible, and an initial test of the dump valves has been successfully completed. Given the successful test results after 7 years without testing, a biennial test frequently is considered adequate.

4.7 Shield Cooling System

Applicability: Applies to the temperature and flow measuring instrumentation in the thermal and biological shields.

Objective: To assure maintenance of the shield cooling conditions as specified in Section 3.7.

Specifications:

- a. During operation of the reactor, the following temperatures and flow rates shall be checked at least daily.
  1. Lower thermal shield flow
  2. Upper thermal shield flow
  3. Thermal shield cooling coil flow
  4. H₂O temperature from thermal shield cooling circuits
  5. Shutter coolant flow
  6. Beam plug coolant flow
  7. Biological shield coil flow
  8. H₂O temperature from biological shield cooling circuits
- b. Calibration of the measuring instrumentation shall be verified annually.

Bases: Daily verification of the above variables is considered adequate in view of the fact that abnormal temperatures signal a control room alarm and low flows automatically initiate a reactor setback.

During the specified interval between calibrations, instrument measurements should normally remain well within acceptable accuracy margins. Should a fault occur which affects the accuracy, the error would readily be uncovered by reactor operators who frequently record and compare instrument readings.

4.8 Radiation Monitoring

Applicability: Applies to radiation monitoring equipment.

Objective: To assure availability of radiation monitors as specified in Section 3.8.

Specifications:

- a. Each of the following radiation monitor channels shall be operationally tested at least daily when operability is required by Section 3.8 and calibrated at least annually.
  1. Reactor cover gas activity monitor (RRa-105)
  2. Exit air gas activity monitor (RRa-306)
  3. Secondary water gamma activity monitors (RRa-301A, RRa-301B, RRa-302, RRa-303)
  4. The tritium activity monitoring system
- b. The primary water activity monitor (RRa-100) shall be checked at least once every fuel cycle and calibrated at least annually.

Bases: The operation of the radiation monitors is verified by positioning a radioactive source near the detector and observing the increase in reading on the control room indicator. At the same time, each alarm and associated annunciator circuit is tested. Since redundancy is not utilized for the radiation monitoring channels, the availability of this instrumentation is assured by more frequent testing.

The primary water activity monitor does not have a built-in test source; therefore, the daily test specified for the other radiation monitors cannot be conveniently done for this instrument. The method used to verify operability is to observe the

pattern of activity increase that is recorded from the start of each cycle and compare this pattern with previous records. Since the pattern repeats itself from cycle to cycle, any unusual indication will require investigation and verification by independent methods.

4.9 Reactor Shutdown Safety System

Applicability: Applies to instrumentation comprising the reactor shutdown (scram and setback) system.

Objective: Verify the availability, calibration and response time of reactor shutdown system instrumentation as specified in Section 3.9.

Specifications:

- a. The availability of each reactor shutdown instrument channel listed below shall be verified from tests performed quarterly. When the reactor is shut down for extended periods during which time it is impractical to maintain the specified test interval, functional tests shall be performed prior to resumption of normal reactor operation.

(1) Nuclear Safety Channels (power and period)	Neutron flux high/period short
(2) Reactor Pressure Channels	Pressure low
(3) Primary Coolant Flow Channels	Flow low, loss of pump-motor power
(4) Core Differential Pressure Channels	Differential low
(5) Primary Coolant Level Channels	Level low
(6) Fuel Cladding Failure Channels	Radioactivity high
(7) Core Inlet Temperature Channels	Temperature high
(8) Thermal Shield Flow Channels	Flow low, loss of pump power
(9) Bio-Shield Flow Channels	Flow low, loss of pump power
(10) Shutdown Coolant Flow Channels (primary system)	Loss of pump power
(11) Manual Shutdown Stations	-

- b. Channel calibrations shall be performed within the time

intervals listed below:

(1) Nuclear Safety Amplifiers	daily (only during operation)
(2) Reactor Pressure	annually
(3) Primary Coolant Flow	semi-annually
(4) Core Differential Pressure	annually
(5) Primary Coolant Level	annually
(6) Fuel Cladding Failure	annually
(7) Core Inlet Temperature	annually
(8) Thermal Shield Flow	annually
(9) Bio-Shield Flow	annually

- c. Trip settings of the nuclear safety channels shall be verified daily during operation.

If conditions exist in which testing may cause an unnecessary reactor shutdown, tests of the nuclear safety channels covered under b and c may be suspended temporarily without violation.

Bases:

- a. The adequacy of the specified test interval is based upon (1) the ability of redundant instrument channels to provide continued protection despite the occurrence of single random failures and (2) the high degree of reliability maintained for components comprising the reactor shutdown system.

Redundant instrument channels, utilized for each protective subsystem, employ 2 out of 3 coincidence logic which requires the tripping of at least two channels for protective action to be initiated. Consequently, a single unannounced channel failure can occur without loss of system availability or protection.

If a recognized channel fault occurs which does not

result in a trip or if a channel is suspected of being incapable of performing its intended function, procedures require that this channel be placed in a tripped condition immediately. Logic, as a result of this action, automatically converts to 1 out of 2. Thus, in this situation a second channel failure can take place without loss of subsystem protection.

No protective channel may be blocked out for instrument maintenance, calibration, faults or other reasons during operation of the reactor. Procedures require, prior to the performance of work, that the channel affected be placed in the tripped condition with a resulting subsystem logic of 1 out of 2. Accordingly, a fault within a subsystem during this period is not capable of preventing protective action.

Reliability of components comprising the reactor shutdown system has been maintained at a high level, especially with respect to the occurrence of unannounced faults. When components fail to perform within specifications or if degradation and failure trends become apparent, necessary steps are initiated to upgrade and maintain this instrumentation at a high performance level.

- b. Calibration intervals are based principally upon the needs of specific instruments to maintain measurement accuracies within acceptable margins. Operating experience has shown that drifts which do occur within these intervals generally are insignificant. Should excessive calibration changes occur, this would become apparent to operators within a short period of time either from the performance of other tests or from frequent logging and comparison of readings among redundant channels.

More frequent calibrations of the nuclear safety system channels are specified since this instrumentation is somewhat more susceptible to drift, and more important, must be periodically calibrated to conform to current computed reactor heat balances.

- c. Whenever a nuclear safety system channel is faulty or suspected of being faulty, procedures require that this channel be placed in the tripped condition. Since the logic now becomes 1 out of 2, testing the remaining two operable channels would result in an unnecessary reactor shutdown. Tests of the channels may, during this situation, be suspended to allow reasonable time during which appropriate repairs may be made.

The temporary suspension of tests does not reduce the reliability of the nuclear safety instrumentation since the conversion to 1 out of 2 logic compensates for the slight increase in test intervals.

4.10 Standby Electrical Power

Applicability: Applies to the standby electrical power sources.

Objective: To assure the availability of the standby electrical power as specified in Section 3.10.

Specifications:

- a. Tests of the 250 VDC battery system shall be performed as follows:
  1. Individual cell voltages of all cells and specific gravity of designated cells shall be measured at least monthly.
  2. The discharge capacity of the battery banks shall be tested at least once each five years.

Bases: The specified tests are designed to demonstrate the continued availability of the standby power system to accept and supply loads. The frequency and type tests performed are considered in agreement with recommended practices for equipment of this type.

4.11 Spent Fuel Storage

Applicability: Applies to the handling and storage of fuel elements in the canal.

Objective: To ensure safe handling and storage of spent fuel elements as specified in Section 3.11.

Specification: The canal water level instrumentation will be tested and the alarm point calibrated at least annually.

Bases: The canal water level is monitored and high and low level alarms are transmitted to the control room. The instrument has proved to be reliable with no significant drift in the alarm setting between the annual tests. In addition, the rate at which the level might change is slow so that any sizeable change is not likely to escape the attention of operating personnel even if the level instrument were to fail. In view of the above, it is believed that the annual test specified is adequate.

The second specification in Section 3.11 has to do with surveillance to ensure that no more than 5 fuel elements are allowed out of criticality safe containers. No surveillance requirement is specified here since movement of fuel in the canal is covered by administrative controls.

4.12 Main Pump Interlock

Applicability: Applies to instrumentation related to main pump operation.

Objective: To assure functioning of pump start interlock as specified in Section 3.12.

Specification:

The operability of the main pump start interlock will be checked prior to one pump operation.

Bases: The main pump start interlock is required only during one pump operation to prevent starting a second pump while the reactor is operating. The interlock serves no protective function during 40 MW operation since both main pumps must be running and the primary flow above the trip level before reactor startup can begin. In consequence, an operating check of this interlock during the shutdown period preceding 30 MW operation is considered adequate.

5. REACTOR EXPERIMENTS

Applicability: Applies to experiments in beam tubes, thimbles or elsewhere around the reactor.

Objective: To define an envelope of safety limitations on experiments.

Specifications:

a. Thimble and Beam Tube Reactivity Limitations

1. The reactivity associated with a movable experiment placed in single beam tubes or thimble shall not exceed 50 cents, or 0.39%k.
2. The sum of the reactivity increments of all movable experiments to be placed in thimbles and beam tubes at one time shall not exceed 1.15 dollars, or 0.9%k with the sum to be obtained by adding the absolute values of the individual reactivity increments.
3. Where flooding of a beam tube or thimble could occur, the total reactivity effect due to the sum of the flooding and ejection of the experiment shall not exceed 85 cents, or 0.66%k.

b. Thimble and Beam Tube Integrity Limitations

Each experiment to be placed in a reactor vessel experimental facility must be contained in such a way that in no circumstances will the presence of the experiment give rise to damage of the thimble or beam tube. In implementing this requirement the following minimum limitations shall be observed:

1. The heat generation rate in the experiment must not be large enough to cause damage to the thimble or beam tube

wall should the coolant flow to the experiment stop with the reactor at full power, except that where the coolant loop has been suitably instrumented and connected to the reactor safety system so that a reactor shutdown results in the event of a coolant failure, the heat generation rate in the experiment just after a reactor shutdown shall not be great enough to cause damage to the thimble or beam tube walls.

- 2, Potentially explosive or reactive materials, including unstable compounds, any material which might react vigorously with the coolant in the thimble or beam tube, or any material which might decompose under radiation and produce potentially explosive or reactive products, must be contained so that in the event of the maximum credible energy generation from the sample no damage to the thimble or beam tube walls will result. To establish the safe containment of such materials, experimental results on the detonation or reaction of the sample materials in the proposed containers will be required, except that if the maximum credible energy release from the experimental material is less than 1 kilocaloric, detailed engineering design calculations on the container may be accepted in lieu of experimental results. In determining the magnitude of the maximum credible energy release, consideration will be given to design features which reduce or eliminate the potential for such releases.

c. Experiments Outside of Thimbles and Beam Tubes

1. If one or both of the in-core fast thimble (V-15, V-16) are removed and replaced by experiments, in direct contact with the primary coolant, the general provisions of specification b (above) apply. However, the specific requirements regarding experimental coolant flow are not applicable since the experiments would be cooled by the primary flow through the core.
2. Any experiments involving the use of high speed rotating machinery or potentially explosive material at or near beam ports will be evaluated to ensure that no damage to the reactor vessel will occur in the event of a maximum possible energy release from the experiment.

Bases:

The basis for the reactivity limits is the criterion that the maximum reactivity associated with any sample to be inserted or removed from the reactor must not be large enough so that the instantaneous insertion or removal of the sample could cause a nuclear transient which would damage the reactor. The other limits are designed to prevent reactor damage due to potential chemical and mechanical energy releases and inadequate cooling of experiments (3.1).

6. DESIGN FEATURES

6.1 Fuel Elements

Applicability: Applies to loading and operating exposure of fuel elements.

Objective: To ensure safe loading and operating exposure of fuel elements.

Specifications:

- a. Type of fuel elements : KL (See figure 1)
- b. Number of elements in core : 28
- c. Maximum ^{235}U core content : 8.94 kg
- d. Maximum amount of ^{235}U consumed in an element by fission burnup as percent of unirradiated ^{235}U content : 39%.

Bases:

The reactor operates with 28 elements arranged in a fixed configuration^(6.1). There are 30 positions normally available but the 2 inner positions are provided for irradiations. Under certain configurations it is possible to achieve criticality with fewer than 28 elements; however, operation under these conditions is prohibited by administrative controls.

The original fuel element design (type KD) was modified^(6.2,6.3) in 1971 to permit a longer operating cycle and thereby reduce fuel costs. The essential feature of the modification was to increase the ^{235}U content per element from the nominal value of 274 grams to 315 grams. The 8.94 kg technical specification corresponds to a loading of 28 new elements with each element having a ^{235}U content (319 g) at the upper limit of the design tolerance^(6.4).

The technical specification on burnup (39%) corresponds to the average burnup level reached in two HFBR fuel elements

6 - 2

that were exposed, without damage, to higher than normal burnup levels in a test program. The peak burnup that was measured in these elements was 66%.

6.2 Reactor Site

Applicability: Applies to the reactor site boundary.

Objective: To identify site requirements relating to reactor safety.

Specification: The reactor exclusion area, low population zone and population center distance shall all lie within the BNL site boundary.

Bases:

The ownership of the BNL site assures that in the event of a reactor accident, necessary steps regarding the movement of personnel and property can be taken as specified in the 10 CFR 100 guidelines. The inclusion of the population center distance within the site boundary covers the future possibility of a large increase in population density developing near the BNL site.

6.3 Design Criteria for Primary Coolant Boundary

Applicability: Applies to the primary coolant pressure boundary.

Objective: To identify the essential design requirements of the primary coolant boundary.

Specification: All components of the primary coolant boundary shall be designed to be at least equal to the code requirements specified in the Final Safety Analysis Report.

Bases: This specification is included to ensure adequate design standards for future modifications.

6.4 Control Rods

Applicability: Applies to the design features of the control rods that relate to reactivity control.

Objective: To identify the essential design features of the control rods.

Specifications:

- a. The poison material shall be Eu_2O_3 and Dy_2O_3 .
- b. The distribution and quantity of poison material in the control blade shall be as shown in the FSAR, fig. 4.7, pg. 4-61.

Bases: The reactivity worth of the control rods is based on the materials and distributions specified.

7. ADMINISTRATIVE CONTROLS

Applicability: Applies to organization, procedures, record keeping, review and audit, training and reporting requirements relating to reactor operation.

Objective: To provide effective implementation of the safety policy of the Laboratory with respect to reactor operation.

Specifications:

- a. The BNL Safety Policy Manual shall be the relevant guideline document with regard to organization, procedures, record keeping, review and audit, training and reporting requirements relating to reactor operation.
- b. The following are elaborations of the Safety Policy Manual guidelines regarding violations of technical specifications.
  1. If a Safety Limit is violated, the reactor shall be shut down until proper review and corrective action has been taken. The AEC shall be notified as soon as information is available. AEC approval for startup is required.
  2. If a specification other than a Safety Limit is not met, either remedial action shall be promptly taken to satisfy the required condition or the reactor shall be shut down. The AEC shall be notified within 72 hours.

Explanatory Note: Material which would normally be discussed in detail in this section of the technical specifications is covered in an existing document, the BNL Safety Policy Manual. The Manual is subject to AEC review and concurrence.

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- (2.5) J. M. Hendrie, BNL-7661, Section 5.4.1, p. 5-17.
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- (2.7) Ibid., Section 9.5.3, p. 9-50.
- (2.8) Ibid., Section 5.3.4.6, pp. 5-16.
- (2.9) Ibid., Section 14.2, pp. 14-2 to 14-8.
- (2.10) Amendment to Safety Report, HFBR, and Errata.
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- (2.13) Ibid., Section 5.5.2.1, pp. 5-24 to 5-25.
- (2.14) Ibid., Section 14.3.4, pp. 14-12 to 14-14.
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- (3.6) Ibid., Section 4.7.11, p. 4-49.
- (3.7) P. Tichler, "Primary System Temperatures Following an Emergency Shutdown", BNL Internal Memorandum (July 10, 1973).

- (3.8) J. M. Hendrie, BNL-7661, Section 4.6.12, p. 4-34.
- (3.9) Ibid., Section 5.5.6, p. 5-30.
- (3.10) Ibid., Section 7.5.3, p. 7-14.
- (3.11) Ibid., Section 4.7.6, p. 4-44.
- (3.12) Response to questions submitted by the ACRS in review of the FSAR, p. 38 (June 19, 1964).
- (3.13) P. Tichler, "HFBR Primary System Chemistry", BNL Internal Memorandum (April 29, 1968).
- (3.14) J. M. Hendrie, BNL-7661, Section 7.4, p. 7-10.
- (3.15) Ibid., Section 6.7, p. 6-7.
- (3.16) Ibid., Section 5.4.7, p. 5-21.
- (3.17) Ibid., Section 7.3, pp. 7-9 to 7-10.
- (3.18) Ibid., Section 10.2.4, pp. 10-5 to 10-9.
- (3.19) Ibid., Section 8.2.3, pp. 8-9 to 8-10.
- (3.20) Ibid., Section 10.3.3, pp. 10-14 to 10-16.
- (3.21) High Flux Beam Reactor Operating Procedures Manual, Section VI. U.
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