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Technical Specifications
Tower Shielding Reactor II

OAK RIDGE NATIONAL LABORATORY

OPERATED BY UNION CARBIDE CORPORATION · FOR THE DEPARTMENT OF ENERGY

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TECHNICAL SPECIFICATIONS
TOWER SHIELDING REACTOR II

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PREFACE

These technical specifications define the key limitations that must be observed for safe operation of the Tower Shielding Reactor II (TSR-II) and an envelope of operation within which there is reasonable assurance that these limits cannot be exceeded. These specifications were written to satisfy the requirements of the Department of Energy (DOE) Manual Chapter 0540, September 1, 1972, and they cannot be changed without the recommendation of the Reactor Operations Review Committee (RORC) of the Oak Ridge National Laboratory (ORNL) and the approval of the Oak Ridge Operations Office of DOE.

As discussed in the Tower Shielding Reactor II Design and Operation Report: Vol. 2 - Safety Analysis, the Maximum Credible Accident (MCA) would not result in core melting. For the MCA it is assumed that the reactor has been dropped from an elevated position and that all water is lost from the core at a time when the reactor has just been operated at 1 MW for 75 hours and the fission product inventory in the core is that from 3000 MWh of exposure. Furthermore, it was concluded that fuel damage would not result from any realistic reactivity accident or from reduction in the flow of the core coolant and that release of fission products to the atmosphere from a defective fuel plate would be restricted to noble gases and halogens and would not represent a significant hazard. The protection afforded by administrative procedures, protective devices, shielded control building, and exclusion area is adequate under normal conditions of operation and maintenance to prevent damage to the reactor and to safeguard both the general public and the operating personnel.

The various specifications given are intended to ensure that operations are conducted in a manner to assure that the above protection remains effective.

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TECHNICAL SPECIFICATIONS
FOR THE
TOWER SHIELDING REACTOR II

GLOSSARY OF TERMS

The following list of terms are defined to aid in the uniform interpretation of these specifications.

1. Abnormal Occurrence -
   a. Any actual safety system setting less conservative than specified in 2.2, Limiting Safety System Settings.
   b. Operation in violation of a Limiting Condition for Operation.
   c. Incidents or conditions which prevented or could have prevented the performance of the intended safety function of an engineered safety feature or the reactor safety system.
   d. A release of fission products of a magnitude to indicate a failure of the principal physical boundary.
   e. An uncontrolled or unanticipated change in reactivity.
   f. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy has caused the existence or development of an unsafe condition in connection with the operation of the reactor.
   g. An uncontrolled or unanticipated release of radioactivity.
2. Afterheat - Rate of heating because of fission products.
3. Background Afterheat - The afterheat due to previous operation (e.g. there are limitations not only on the afterheat generation due to current reactor operation but also on the background afterheat).
4. Certified Operator (RO or SRO) - Any individual who has successfully completed the training, examination, and certification for reactor operator (RO) or senior reactor operator (SRO) pursuant to DOE Manual Chapter 0540 and pursuant to IAD 8401-6.
5. Channel - A system of components to perform a specific function with respect to measurement or control of system parameters.

6. Channel Calibration - An adjustment of the channel such that its output responds, with acceptable range and accuracy, to known values of the parameter which the channel measures or to known input signals when access to the primary element is limited.

7. Channel Check - A qualitative verification of acceptable performance by observation of channel behavior. This verification shall include comparison of the channel with expected values or with other independent channels or methods of measuring the same variable.

8. Channel Test - The introduction of an input signal into the channel to verify that it is operable.

9. Control Element - A device integral to the reactor that has the designed purpose of changing the reactivity in a reactor by perturbing the neutron population.

10. Degradation of the Reactor Shutdown System -

Class 1. The actual failure of the reactor shutdown system to initiate the protective action when the reactor variable has exceeded the limiting safety system settings or the premature termination of the protective action.

Class 2. Failure or malfunction of components, personnel error, or procedural inadequacy which, due to its effect on multiple units would, by itself, prevent the reactor shutdown system from providing the protective action at the limiting safety system settings.

Class 3. Failure or malfunction of one or more components, personnel error, or procedural inadequacy which reduces the capability of the reactor shutdown system to the extent that the occurrence of a random single failure would prevent the protective action at the limiting safety system settings.

Class 4. Failure or malfunction of one or more components, personnel error, or procedural inadequacy affecting a limited number of units such that, although the degree of redundancy may be reduced, the reactor shutdown system retains, even after the application of the single failure criterion, the ability to provide the protective action required (conditions and LSSS) by the technical specifications.

11. Experiment - Any apparatus, device, or material placed near the reactor pressure vessel or in line with a beam of radiation emanating from the reactor or any operation designed to measure reactor characteristics.

12. Jordan Test - A test in which the voltage at the input grid to a sigma preamplifier is increased to simulate a high neutron flux at the safety chamber and thereby initiates a reactor shutdown.

13. Limiting Conditions for Operation - Those administratively established constraints required for safe operation of the facility.

14. Limiting Safety System Settings - Settings on instruments that initiate automatic protective action at a level such that the safety limits will not be exceeded.

15. Measuring Channel - That combination of sensor, lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a process variable.

16. Operable - Capable of performing its intended function in a normal manner.

17. Operating - Performing its intended function in the normal manner.

18. Personnel Radiation Protection System - A system of door and gate interlocks that tie into the reactor safety system and effect a reactor shutdown if violated (see Sect. 3.7).

19. Reactor Safety System - That combination of measuring channels, associated circuitry, actuators, and reactivity controlling elements that forms the automatic protective system of the reactor or provides information that requires that manual shutdown be initiated.

20. Reactor Secured - That overall condition where all of the following conditions are satisfied:

    a. Reactor is shut down.
    b. Electrical power to the control element drive or actuating circuits is switched off and switch key is in proper custody.
    c. No work is in progress involving in-core fuel handling or refueling operations.

21. Reactor Shutdown - That condition where the negative reactivity is equal to or greater than the shutdown margin.

22. Safety Limit - Limits on important process variables which are necessary for protection of the integrity of the physical barriers that guard against the release of radioactivity.

23. Shutdown Margin - The amount of reactivity which must be added to a shutdown reactor to make it critical.

24. Surveillance - Monitoring, checking, testing, calibrating, or inspecting systems or components related to verifying that operation is consistent with the technical specifications.

25. Time Intervals - In reference to surveillance or tests.

    a. Annually - To be performed once each year at intervals not to exceed 14 months.
    b. Semiannually - To be performed twice each year at intervals not to exceed 8 months.
    c. Quarterly - To be performed four times each year at intervals not to exceed 5 months.
    d. Weekly - To be performed once each week at intervals not to exceed 10 days.

26. Tried Experiment - An experiment previously performed with this reactor, or an experiment of similar size, shape, composition, and location as previously performed with this reactor.

TECHNICAL SPECIFICATIONS
TOWER SHIELDING REACTOR II

1.0 GENERAL

1.1 The Tower Shielding Facility

The Tower Shielding Facility (TSF), an integral part of the Oak Ridge National Laboratory (ORNL), is located within the well-established ORNL controlled access area in Roane County, Tennessee at a distance of 2.35 miles SSE of the main Laboratory complex. Adequate personnel- and visitor-control policies have been established so that only necessary operating personnel and persons having legitimate business are permitted within the immediate area around the TSF. Originally built in 1954 for the purpose of studying asymmetric shield configurations for the Aircraft Nuclear Propulsion Project, the facility is still in use because of its versatility for shielding studies.

1.2 The Tower Shielding Reactor II

The Tower Shielding Reactor II (TSR-II) is a spherically symmetric reactor designed and operated specifically for reactor shielding studies. Approval was requested and obtained[1] during January 1972 to operate the reactor at a power level of 1 MW. A power level of 950 kW(t) was obtained at 00:01 hours on January 22, 1972. It is with respect to operating the TSR-II at the 1 MW(t) power level that the technical specifications contained in this document are presented. ORNL/TM-2893, Vol. 2 (October 7, 1970) is the Safety Analysis Report in support of operating the reactor at the 1 MW power level and has also served as a basis for the generation of these specifications.

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

2.1.1 Safety Limits in Normal Operation

Applicability - This specification applies to the interrelated variables associated with the reactor core characteristics during normal operations. The variables are:

P_T = Total reactor thermal power.

Q = Reactor cooling water flow rate.

ΔP = The pressure drop of the cooling water flow through the reactor.

T_R = Temperature of the cooling water at the inlet to the reactor.

T_H = Temperature of the cooling water at the outlet of the heat exchanger.

ΔT = Increase in the cooling water temperature as it passes through the reactor.

Objective - The objective is to ensure that the integrity of the fuel cladding is maintained.

Specification -

P_T < 3 MW

Q > 400 gpm

ΔP > that equivalent to cooling water flow rate of 400 gpm, 6 psi

T_R < $160^{\circ}F$

T_H < $160^{\circ}F$

ΔT < $50^{\circ}F$

Bases - The criterion used to establish the safety limits was that the surface temperature of the fuel plates be maintained below the saturation temperature of water at every point in the core.[2] The design saturation temperature was set at 283°F because the minimum pressure in the core during normal operation always exceeds 36.5 psig.[3] Under normal operating conditions the cooling water flow is 800 gpm and the maximum allowable power is 1 MW. Analysis shows that if the power level were raised to 3 MW the maximum fuel plate temperature would reach only 205°F which would preclude boiling even if the core pressure dropped to atmospheric.[4] The analysis also indicates that the power could be raised to 3 MW, the cooling water flow rate dropped to 400 gpm and the maximum fuel plate temperature would reach only 258°F which is still below the design saturation temperature. For the maximum fuel plate temperature to approach the saturation temperature of 283°F under the above conditions the reactor inlet temperature would have to exceed 160°F. Although this is still a safe operating condition it could not be achieved because safety system action would terminate operation before the reactor power, cooling water flow rate or the reactor coolant inlet temperature approached the above values.

Thermal analysis of the Low Intensity Testing Reactor (LITR) core for 3 MW operation indicates the safety margin that the above TSR-II operating conditions would provide. The TSR-II has the same type and spacing of fuel plates as the LITR but the TSR-II heat transfer area is 1.5 times larger than that in the LITR. The maximum heat flux in the LITR at 3 MW was 52,600 Btu-ft^{-2}-hr^{-1} in a channel with a flow velocity of 0.96 fps (total core flow of 500 gpm) and a burnout ratio (burnout power/3 MW) of 16.9.[5] The maximum flux in the TSR-II fuel annulus at 3 MW is 20,930 Btu-ft^{-2}-hr^{-1} in channel 41 which has a coolant velocity of 0.93 fps (744 lbs per hr per channel of one element for a total core flow of 400 gpm).[6] The TSR-II burnout ratio should be at least as large as that in the LITR.

2.1.2 Safety Limits in Low-Flow Mode of Operation

Applicability - This specification applies to the interrelated variables associated with the reactor core characteristics during operation in the Low-Flow Mode (See Sect. 2.1.1).

Objective - The objective is to ensure that the integrity of the fuel cladding is maintained.

Specification -

$P_T < 50$ kW $Q > 25$ gpm $T_R < 160^{o}F$

$T_H < 160^{o}F$ $\Delta T < 50^{o}F$

Bases - The Low-Flow Mode of operation is needed to perform criticality checks with new core loadings. The power requirements, therefore, are modest and the coolant flow rate is minimal. The limits on temperature will preclude boiling in the core even at atmospheric pressure. The flow requirement is necessary only to make the temperature measurements meaningful.

The margin of safety for the power requirement can be inferred by comparison with boiling experiments in the LITR (See Bases for Normal Operation above). As was the case in the LITR the TSR-II has no external loop for natural circulation cooling. These experiments demonstrated that natural convective cooling constituted an adequate heat removal mechanism and that the onset of boiling occurred at 1.2 MW.[7] The burnout heat flux for the LITR in natural convection cooling[8] was determined to be 1.26×10^{5} Btu-ft^{-2}-hr^{-1}. At 50 kW the maximum heat flux in the LITR would be 875 Btu-ft^{-2}-hr^{-1} which would give a burnout ratio of 144. The safety margin in the TSR-II would be comparable to that in the LITR.

2.2 Limiting Safety System Settings

Applicability - This specification applies to the set points for the safety channels monitoring the reactor variables described in Section 2.1.1, Safety Limits in Normal Operation.

Objective - The objective is to ensure that the automatic protective action is initiated in order to prevent exceeding established safety limits.

Specification -

A. For normal operations, the limiting safety system settings will be as follows:

P_T < Neutron flux equivalent to 1.6 MW(t)

Q > 450 gpm minimum

ΔP > 8.5 psi

T_R < $140^{o}F$ maximum

T_H < $140^{o}F$ maximum

ΔT < $16^{o}F$ maximum

B. For low-flow mode operations the limiting safety system settings will be as follows:

P_T < Neutron flux equivalent to 16 kW(t)

Q > 45 gpm minimum

T_R < $140^{o}F$ maximum

T_H < $140^{o}F$ maximum

ΔT < $16^{o}F$ maximum

Bases - The limiting safety settings were chosen such that, when measurement uncertainties and anticipated transient conditions are considered, there is confidence that the criteria set forth in the safety limits are at all times satisfied. From the transient analysis it was concluded that no core damage would result from transients larger than any that could be realistically achieved.[9] If any single parameter exceeds its set points and reaches the safety limits the temperatures in the core would not reach the saturation temperature. Two parameters would have to reach their safety limit for the saturation temperature to be achieved and this would only result in nucleate boiling which is a safe operating condition.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity

Applicability - These specifications apply to the reactivity conditions of the reactor and its associated components.

Objective - The objective is to ensure that the reactor can be made subcritical and maintained in that condition at all times.

Specifications - The following reactivity conditions are mandatory for reactor operations:

a. The excess reactivity of the clean-cold configuration shall be no more than 1.9% $\frac{\Delta k}{k}$.

b. The shutdown margin shall be no less than 1.9% $\frac{\Delta k}{k}$.

c. The maximum rate of reactivity addition by the shim-safety plates in the range where criticality may be achieved will be less than 0.066% $\frac{\Delta k}{k}$ per second.

d. The maximum reactivity addition rate with the regulating plate in either manual or servo action shall be 0.11% $\frac{\Delta k}{k}$ per second.

e. The total reactivity worth of the regulating plate shall be less than 0.5% $\frac{\Delta k}{k}$.

Bases - The reactivity limits specified for excess reactivity and shutdown margin are established so that the reactor can be shut down from any operating condition and remain shut down after the cooling water temperature drops and xenon decays even if one shim-safety plate should stick in the fully withdrawn position.

The maximum reactivity addition rates specified for the shim-safety plates and the regulating plate are within the limits that comparisons with SPERT-I data indicate can safely be terminated with only intrinsic shutdown mechanisms in the TSR-II.[10] The maximum ramp reactivity addition that can be achieved in the TSR-II is 0.066×10^{-2} $\frac{\Delta k}{k}$ per second which is equivalent to a step reactivity insertion of \$1.128 and an α of 18 s^{-1}. Comparison to SPERT data indicates the TSR-II could safely experience a ramp insertion that could have an $\alpha \leqq 35\ s^{-1}$ without safety system action.

Analogue computer studies indicate that the TSR-II protection system can safely limit power excursions due to step reactivity addition of an equivalently higher magnitude than those due to the above ramp reactivity addition rates even if the intrinsic shutdown mechanisms were not also acting to limit the excursions.[11] The safety system is capable of limiting a power excursion associated with a 18.0 ms period ($\alpha \sim 56\ s^{-1}$) such that the fuel temperature does not exceed the saturation temperature of the water in the core.

The total reactivity in the regulating plate is limited to that sufficient to provide adequate control but much less than that required to make the reactor prompt critical.

3.2 Reactor Control and Safety System

Applicability - This specification applies to the safety, control, and surveillance instrumentation required for startup and operation of the reactor. It also applies to the availability of surveillance instruments during a power outage.

Objective - The objective is to ensure that an adequate complement of safety, control, and surveillance instrumentation are available during startup and operation of the reactor, and that surveillance instrumentation is available during a power outage.

Specifications -

a. The minimum complement of reactor safety and measuring instrumentation required for startup and operation in the two modes of operation shall be as specified in Table 3.1.

b. The minimum of operable shim-safety plates shall be four.

Bases - An interlock which requires a counting rate of at least two counts per second in the neutron startup channel assures that sufficient neutrons are available for proper operation of the startup channel.

The neutron detectors of the measuring channels shown in Table 3.1 provide assurance that the reactor power is adequately monitored from source to maximum power and that the information is displayed in the control room.

Table 3.1 Minimum instrumentation required for reactor startup and operation in normal and low-flow mode.

Description	Number required	
	At startup	Power operation
Safety or protective channels		
Power level safety channels	2	2
Reactor cooling water flow channel	1	1
Core pressure drop channel[a]	1	1
Core Δ temperature channel[a]	1	1
Heat exchanger outlet or reactor inlet water temperature channel[a]	1	1
Measuring channels		
Neutron counting-rate channel[b]	1	1
Log N power or picoammeter channel	1	1
Water activity monitor	1	1

[a] Channel only required for normal operation.

[b] This channel shall have an auxiliary power supply that shall be operable during a power outage.

To assure that the temperatures in the core are maintained at safe levels, power and cooling water flow channels are provided. (For one flow rate channel the differential pressure of the cooling water flow through the core is monitored rather than the actual flow rate.) Assurance that the heat removal system is operating is provided by the heat exchanger outlet temperature channel and the reactor inlet temperature channel.

The heat exchanger outlet and reactor inlet temperature channels together with channels to monitor the temperature rise of the cooling water flowing through the reactor provide independent assurance that the core temperatures are maintained at safe levels.

The combination of the temperature rise through the reactor and the flow rate of the cooling water through the reactor is the means used to monitor the actual power of the reactor.

In addition to the required safety channels there is a log N period scram to prevent the power level from increasing on a period of less than 1 second. It provides a small improvement over the level safety channels for limiting excursions in a startup accident only.[12] The action of the period scram was not considered in the transient analysis in the Safety Analysis Report.

The manual scram which can be classified as an administrative control permits the operator to shut down the reactor if an unsafe or abnormal condition arises. The reactor can also be shut down by using the startup switch or a completely independent Jordan test.

The specification of the minimum number of operable shim-safety plates ensures sufficient redundancy in the number of plates available to shut down the reactor. It should be noted that it is not possible to operate the reactor with any plate inoperable unless the plate is in its shutdown position. An auxiliary power supply assures that it is possible to monitor that the reactor is shut down during a power outage.

The water activity monitor shall ensure that the operator shall have early indication of any leakage of fission products from the fuel elements so that he may terminate reactor operation and minimize the total leakage.

3.3 Reactor Cooling System

Applicability - This specification applies to the cooling system base pressure and the pressure drop across the core.

Objective - The objective is to ensure that the design pressure of the system is not exceeded and that the rate of heat removal from the core does not increase inadvertently.

Specifications - The base pressure in the reactor cooling system shall not exceed 125 psig. The pressure drop across the core shall not exceed 35 psi.

Bases - The design pressure of the reactor pressure vessel and the detention tank was 150 psig. The design of the system is such that the actual working pressure of these items will not exceed 100 psig if the base pressure is limited to 125 psig. The system base pressure is limited to 125 psig by two pressure relief valves.

If a change occurs that reduces the cooling system pressure drop, the system flow would increase. Such a change would increase the rate of heat removal from the core and, under these conditions, a rise in power level not sensed by the power level safety channels might not cause the Δ temperature to reach its scram set point. Limiting the core pressure drop to 35 psi will assure that the information received by the safety channels for core temperature rise is meaningful.

3.4 Shim-Safety Plate Response Times

Applicability - This specification applies to the time intervals between the initiation of a reactor shutdown signal and the initial movement of a shim-safety plate (release time) and to that between the initiation of the shutdown signal and the time the plate is moved to its normal shutdown position (total insertion time which includes release time).

Objective - The objective is to ensure the proper performance of the shim-safety plates during a reactor scram.

Specifications -

a. The release time for any of the five shim-safety plates shall not exceed 70 ms.

b. The total insertion time (including release time) of any of the five shim-safety plates from its normal operating position to its normal shutdown position shall not exceed 200 ms.

Bases - The values of release time and total insertion time used in the transient analysis[13] provided a reactivity reduction of 1% $\frac{\Delta k}{k}$ in 100 ms. The above values of release time and total insertion time will ensure a reactivity reduction of the same magnitude. The standard method of determining the release time in the reactor includes time for initial movement of the shim-safety plate and actuation of a flow sensor so the actual release time is shorter than the measured release time.

3.5 Hoist Slack-Line Protection Devices

Applicability - This specification applies to the systems for preventing the lowering of a load with a tower hoist if a slack line condition occurs on that hoist.

Objective - The objective is to stop lowering with a hoist if the cable goes slack on its drum before any cable rises completely out of the cable groove.

Specifications -

a. There shall be at least two independent sensors to indicate if a cable rises in its groove.
b. Each sensor shall actuate two independent channels, each of which shall stop the lowering action of the hoist.

Bases - Two correctly spaced sensors, each actuating two separate systems to interrupt the lowering action of a hoist if the cable becomes slack, are sufficient redundancy to ensure that the cable will not become slack enough to foul and be damaged.

3.6 Instrumentation for Personnel Radiation Protection

Applicability - This specification applies to instrument channels to protect personnel inside the control building from reactor radiation.

Objective - The objective is to specify the minimum number of monitoring channels that must be operable to ensure that the radiation level in the control room from reactor operations does not exceed a preset level.

Specifications - Reactor operation shall be terminated if the radiation level seen by radiation monitors at any two of the three building exits exceeds 23 mR/h, or if the monitor at one building exit is out of service and the radiation level seen by a monitor at another exit exceeds 23 mR/h.

Bases - Radiation monitors for personnel protection are located at the building exits because that is where the radiation levels are the highest. If the dose rate levels at the monitor points do not exceed the specified values personnel at the normal work area will receive negligible radiation. All building exit monitors alarm locally and at the reactor console for high radiation or unit out-of-service. Reactor shutdown action requires alarms from 2 out of 3 stations so that incidental use of calibrating sources near one monitor would not initiate a reactor shutdown.

3.7 Personnel Radiation Protection System

Applicability - This specification applies to channels for protection of personnel entering the experimental area near the reactor or inadvertently remaining in that area during timeout prior to reactor operation.

Objective - The objective is to stipulate the location of double-tracked safety system channels that must be operable to assure that personnel cannot inadvertently enter the experimental area near the reactor during reactor operation. It is also to stipulate the location of double-tracked channels that an individual inadvertently remaining in that area may use to prevent reactor startup or to stop reactor operation.

Specifications - During reactor operation two channels must be operable, either of which shall effect a reactor shutdown if any of the following are opened or actuated:

a. Control building north door.

b. Control building ramp gate.

c. Control building escape hatch.

d. West gate in the 600-ft-radius fence.
e. North gate in the 600-ft-radius fence.
f. Remote manual scram at pool hoist station.
g. Remote manual scram at ramp hoist station.

Bases - Procedures ensure that prior to operation with the reactor out of the handling pool personnel inside the outer exclusion fence are accounted for either in the control building or working in a remote area in accordance with special procedures. Procedures also include instructing personnel that exit from the building is prohibited during this operation. If an individual inadvertently remains outside the control building he may stop the operation by pushing a stop button in the experimental area. He may even do this while the warning horn is operating prior to actual startup of the reactor. The double-tracked channels in the personnel protection system will automatically initiate a reactor shutdown if an individual enters or leaves the control room or enters or leaves the 600-ft-radius exclusion area when the reactor is being operated when it is out of the handling pool.

3.8 Limitations on Experiments

Applicability - This specification applies to experiments utilizing the reactor as a source of radiation.

Objective - The objective is to prevent damage to the reactor or excessive release of hazardous materials in the event of an experiment failure.

Specifications - The reactor shall not be operated except under the following conditions governing experiments:

a. The reactivity worth of the experiments shall be limited to values that can be achieved within the core reactivity limits specified in Section 3.1 and a failure or malfunction shall add no more than 1% reactivity to the reactor.
b. Experiments shall be performed outside the pressure vessel. (This does not pertain to the determination of the reactivity worth of shields.)

c. Any hazardous material used in an experimental configuration shall be suitably contained and the region between the reactor vessel and the experimental configuration shall be vented to the atmostphere.

Bases - Exclusion of experiments from inside the pressure vessel places a fixed barrier and a minimum distance between the reactor fuel and the experiment. This arrangement is a most effective way to ensure that there will be no damaging interaction between the fuel elements and any components of the experiment.

The requirement that hazardous materials be suitably contained ensures that the applicable DOE, OSHA, EPA, and ORNL requirements be met before an experiment is approved. The additional requirement of a vented space between the contained experiment and the reactor vessel is further assurance that there will be no interaction between a failed experiment and the reactor.

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4.0 SURVEILLANCE

4.1 Reactivity

Applicability - This specification relates to the surveillance requirements of the reactivity in the core.

Objective - The objective is to ensure compliance with the specifications set forth in Section 3.1 relating to reactivity in the core.

Specifications -

a. The reactivity worth of the regulating plate shall be measured annually and also whenever a new fuel loading or a control mechanism housing is installed.

b. The reactivity worth of new shim-safety plates shall be determines by comparing under similar conditions, with the reactor at delayed critical, the position of the new shim-safety plates with that of the plates which have been replaced. The reactivity worth of the shim-safety plates shall be determined if the above operating positions do not agree or, if, the regulating plate calibration (see above) does not agree with previous calibrations.

Bases - The regulating plate and the shim-safety plates are located symmetrically inside the core annulus. Any change in the core which changes the reactivity worth of the shim-safety plates as a function of their separation from the fuel will change the reactivity worth of the regulating plate in a similar manner. Determination of the reactivity worth of the regulating plate coupled with the operation of the shim-safety plate at their expected operating position, at the frequency specified, provides adequate assurance that the shutdown margin and excess reactivity requirements are as set forth in Section 3.1.

The regulating plate and shim-safety plates have fixed drive speeds. The rate of reactivity addition, therefore, cannot be modified until a formal change has been reviewed and authorized. Such review is sufficient to ensure that the rate of reactivity addition specified in Section 3.1 will not be exceeded.

4.2 Reactor Control and Safety Systems

Applicability - This specification applies to the surveillance requirements for the reactor control and safety systems.

Objective - The objective is to ensure that the reactor safety systems will be in operable condition if they are needed to provide any required safety action and that the measuring instrumentation is reliable.

Specifications -

a. A channel test of the reactor safety system channels and auxiliary power supply shall be performed after maintenance and weekly when the reactor is in operation.

b. A channel check of each of the reactor safety system channels listed in Table 3.1 shall be performed daily when the reactor is in operation.

c. A channel calibration of the reactor safety system channels shall be performed semiannually and after maintenance to the reactor safety system channels that could affect the calibration of the reactor safety system channels.

d. The power measuring channels shall be calibrated against a primary system heat balance whenever the reactor is operated at 500 kW or above if the operation is sufficiently long to obtain temperature equilibrium and, in any case, at least semiannually and after maintenance to the power measuring channels that could affect calibration of the power of the measuring channels.

Bases - Redundancy is provided in all safety channels and because of this multiple system random failures have an extremely low probability of jeopardizing the ability of the channels to perform their intended function. Operating experience since 1960 confirms this contention and also confirms that the specified frequency of checks, tests, and calibrations for the reactor safety system and measuring channels is adequate to ensure a high degree of reliability.

4.3 Reactor Cooling System

Applicability - This specification applies to the surveillance requirements of the reactor cooling system.

Objective - The objective is to ensure the continuing integrity of the reactor cooling system and to ensure compliance with the specifications set forth in Section 3.3.

Specification - The pressure relief valves in the system shall be tested annually and after maintenance on the pressure relief valves that could affect the pressure setting of the valves.

Bases - The daily check of the core pressure drop is the best method to note any variation of reading with time and to ensure compliance with the specifications of Section 3.3.

Experience has shown that an annual test of each pressure relief valve, all of which are in duplicate, is sufficient to ensure that the pressure in all parts of the system remains below the design value and in compliance with the specifications of Section 3.3.

4.4 Shim-Safety Plate Response Times

Applicability - This specification applies to the surveillance requirements for the shim-safety plates.

Objective - The objective is to ensure that the shim-safety plates are operable.

Specification - The release time and total travel time of each shim-safety plate shall be measured when a new control mechanism housing is installed, when maintenance is performed on the system, and routinely semiannually.

Bases - The release and total travel times of each shim-safety plate are measured to ensure that the plates are operating freely. Experience has shown that the above frequency is enough to ensure compliance with the specifications in Section 3.4. Prudence dictates that the measurements be made when the system is installed or manipulated in a nonroutine manner.

4.5 Hoist Slack-Line Protection Devices

Applicability - This specification applies to the surveillance requirements of the hoist slack-line protection systems.

Objective - The objective is to ensure the continuing integrity of the slack-line protection systems and to ensure compliance with the specifications set forth in Section 3.5.

Specifications -

a. Prior to use of the hoists the operation of the slack-line protection devices on each hoist shall be checked after maintenance, alterations, or shutdown periods exceeding one month.
b. When the hoists are used routinely, operation of each sensor in the slack-line protection systems shall be checked weekly.

Bases - The slack-line protection system was designed to have sufficient redundancy to ensure that the cable could not get out of its groove without detection. Experience has indicated that the frequency of checking is sufficient to ensure the operability of the systems if they are needed. Prudence dictates performing the checks after maintenance, alterations, or shutdown periods exceeding one month.

4.6 Instrumentation for Personnel Radiation Protection

Applicability - This specification applies to the surveillance requirements of the radiation monitoring equipment required in the control building during reactor operation.

Objective - The objective is to ensure that the radiation monitoring equipment is operating and to verify appropriate alarm settings.

Specifications - The operation of the radiation monitoring channels and their associated alarm setting shall be checked weekly during periods of reactor operation. The radiation monitoring equipment shall be calibrated annually and after maintenance on the radiation monitoring equipment that could affect the calibration of the equipment.

Bases - Experience has shown that the above frequency of check and calibration of the radiation monitors is sufficient to ensure that the instrumentation is operable and will meet the requirements of the specification in Section 3.6.

4.7 Personnel Radiation Protection System

Applicability - This specification applies to the surveillance requirements for the Personnel Radiation Protection System.

Objective - The objective is to ensure that the systems function as required to prevent inadvertent entry into the area around the operating reactor or so that anyone inadvertently left in the area can prevent startup of the reactor.

Specification - The operation of each channel shall be checked weekly during periods when the reactor is in operation and after maintenance on the radiation monitoring equipment that could affect the calibration of the equipment.

Bases - Experience has shown that the weekly checks are sufficient to ensure that the systems are operable in case they are challenged inadvertently or operated when they are needed.

4.8 Limitations on Experiments

Applicability - This specification applies to the surveillance of the limitations on experiments.

Objective - The objective is to assure that damage to the reactor or excessive release of hazardous materials shall not occur.

Specifications -

a. Measurements shall be performed to determine the reactivity worth of the experiment. The information describing the reactivity worth of the experiment shall be recorded in the Operations Log Book.

b. Integrity of containment of hazardous materials will be verified by Inspection Engineering personnel prior to use and periodically while the material is maintained for use in the experimental program.

<u>Bases</u> - Specification 1 ensures that the reactivity worth of an experiment is known before the experiment is conducted and that the worth is determined under conditions which can be directly controlled by the reactor operator.

Specification 2 ensures that the integrity of the containment will remain in a satisfactory condition for use in the experimental program.

5.0 DESIGN FEATURES

5.1 Reactor Site

The Tower Shielding Facility is located on a knoll with an elevation of 1069 ft, 2.35 miles south-southeast of ORNL, 6 to 13 miles from the city of Oak Ridge, and 17 to 25 miles from the city of Knoxville. The TVA Melton Hill Dam is located 0.8 mile south of the TSF on the Clinch River, which forms a natural boundary of the restricted area. The nearest ORNL facilities, the Health Physics Research Reactor (HPRR) and the High-Flux Isotope Reactor (HFIR), are over 6000 ft from the TSF and are separated from it by an offshoot of Copper Ridge and by the highest point of Copper Ridge, respectively.

The TSF and the HPRR are situated within a general exclusion area which is enclosed by a nominally 6-ft-high chain-link fence topped with three strands of barbed wire (called the "perimeter fence"). Additional security for the TSF is provided by a nominally 8-ft-high chain-link fence topped with three strands of barbed wire which is located on a circle of 600-ft radius from the reactor. The TSF is separated from the other reactor, HPRR, in the general exclusion area by a nominally 5-ft-high field wire fence.

Two reinforced concrete underground buildings adjacent to and north of the towers provide a shielded working area for personnel during reactor operation. The buildings are shielded against radiation by an 18-in.-thick concrete roof covered with 3 1/2 ft of earth.

5.2 Reactor Fuel

The TSR-II core consists of 60-mil-thick curved aluminum-clad uranium-aluminum alloy plates cooled and moderated with light water. The plates are shaped and arranged so that the assembled core is a spherical fuel annulus.

Each fuel plate is 0.060-in. thick and consists of a sandwich of uranium-aluminum alloy clad in aluminum. Fuel plates are peened and welded 0.120 in. apart into aluminum side plates to form elements. Three types of elements are used: annular elements, so called because they form a cylindrical fuel annulus when assembled together; central elements, which are used in the upper and lower sections of the core; and one 3-in.-diam cylindrical "plug" element, which is centered in the lower central elements and which may contain an antimony-beryllium source. Four central elements are mounted inside the lower end of a bottomless aluminum cylinder which is suspended inside the reactor pressure vessel. Neutron-absorbing, shim-safety control plates and their operating mechanisms are mounted in a spherical control mechanism housing which is mounted above the lower elements. Four upper central elements are mounted above the housing. Twelve annular elements are mounted on the central cylinder in the region between the central cylinder and the reactor tank. Four 1/8-in.-thick fuel-loaded, lune-shaped aluminum covers are mounted on the control mechanism housing and form a spherical shell that is located 1/4 in. inside the elements in the fuel annulus. To limit the temperature in the lune-shaped fuel plate the mass of fuel in these plates is limited so that the total power generated in them will not exceed 4% of the power generated in the core.

The fuel elements will be fabricated in accordance with ORNL "Specifications for TSR-II Fuel Assemblies."

5.3 Fuel Storage and Handling

A two-section concrete pool provides shielding during the removal and storage of fuel elements and the changing of reactor shields. The pool is located midway between the west tower legs. Its large section is 20 ft by 20 ft and 25 ft deep, and its small section is 4 ft wide by 12 ft long by 22 ft deep. The large section of the pool may be covered with three 2-ft-thick slabs of reinforced concrete.

Irradiated fuel may also be stored in concrete-lined silos which are 4 ft in diameter, 20 ft deep, and which may be closed with 4 ft of reinforced concrete.

5.4 Reactor Cooling System

The reactor cooling system consists of a main pump for pumping demineralized water from a detention tank through aluminum pipe and neoprene hose through the reactor vessel and then through a forced draft air cooler. The reactor vessel is a cylindrical aluminum tank with a hemispherical bottom. The vessel was designed, fabricated, inspected, and tested in accordance with the latest published ASME code for unfired pressure vessels.[14]

A fill and pressure pump operates in conjunction with a variable pressure regulating system to maintain about 5 to 10 psi in excess of the minimum necessary to keep the system full of water as the height of the reactor is varied. Pressure relief valves are connected to ensure that the system base pressure remains within specified limits (see Sect. 3.3).

The forced draft air cooler has two large variable-pitch fans that blow air across aluminum tube and fin radiators to remove the heat from the water. The pitch of the fans and the position of the radiator louvers are controlled to maintain a fixed temperature for the water leaving the cooler.

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6.0 ADMINISTRATIVE CONTROL

6.1 Organization

The Oak Ridge National Laboratory, which is owned by the United States Department of Energy (DOE) and operated under contract by the Nuclear Division of Union Carbide Corporation, shall be responsible for operation and supervision of the facility. The Operations Division shall be directly responsible for the operation of the facility. The relationship of the reactor operating staff to the Laboratory's structure is shown in Fig. 1.

6.2 Personnel Qualification

The reactor shall be operated by personnel examined and certified under the general provisions of DOE Manual Chapter 0540, Appendix 8401-II, and IAD-8401-6, and approved by the Operations Division Director.

6.3 Minimum Staff Requirements

a. A Senior Reactor Operator and one other TSF staff member shall be present in the control building when the reactor is operated.

b. Either a Senior Reactor Operator or a Reactor Operator shall be in a position to take remedial action as necessary during reactor operation.

c. A Senior Reactor Operator and at least two other members of the TSF staff shall be present whenever fuel elements, a control mechanism housing, or shields are removed from or inserted into the reactor pressure vessel.

6.4 Facility Modifications

It shall be the responsibility of the Division Director to ensure that changes in technical specifications or modifications to the plant protection system, reactivity control systems, or engineered safety features, or changes that involve a safety question not reviewed in the safety analysis report shall receive

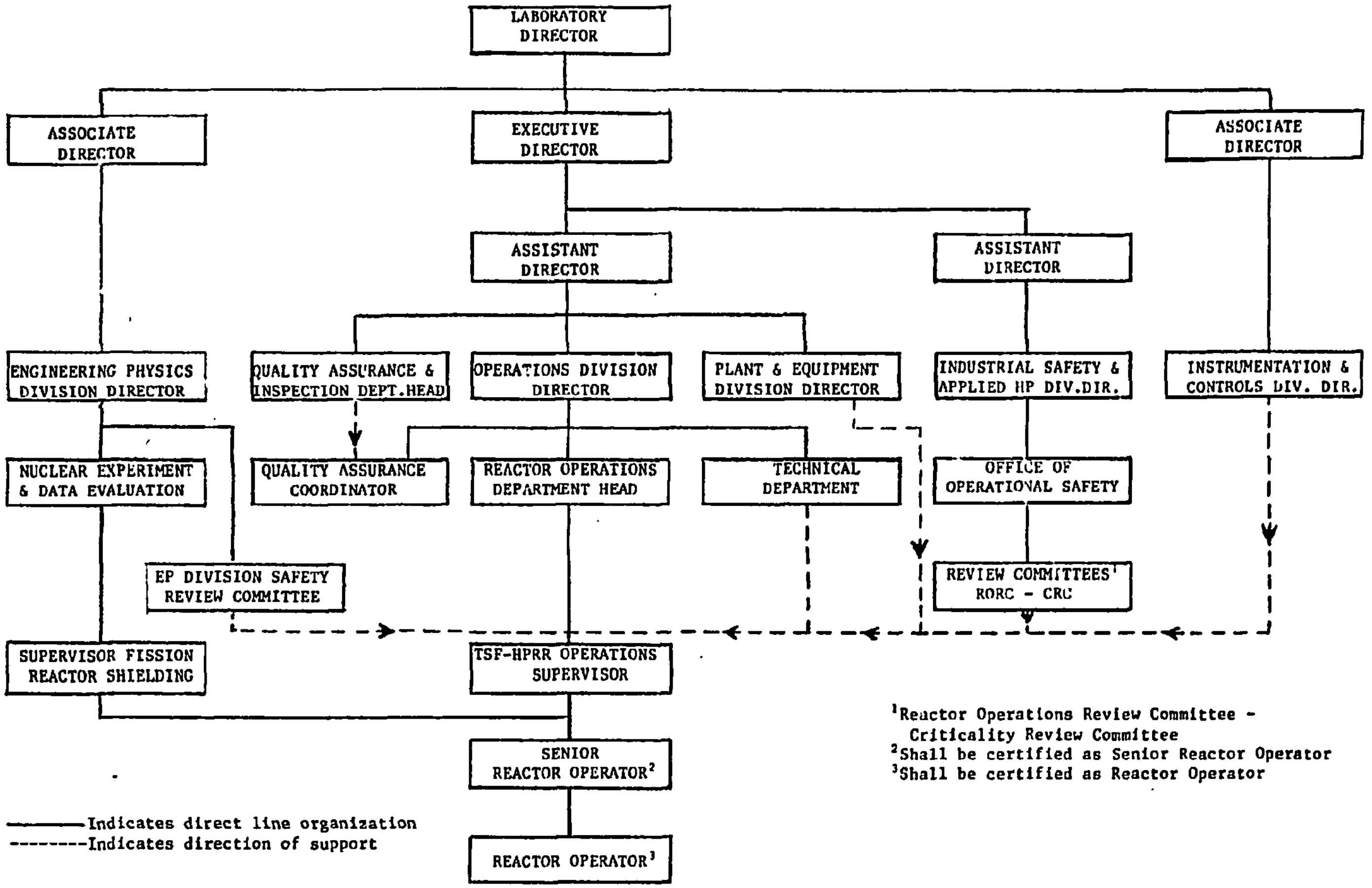

Fig. 1 TSR-II Organization Chart

prior review and authorization by the RORC and/or DOE in accordance with the requirements of ORNL Standard Practice Procedure 18-B and ORO Manual Chapter 0540.

b. Certain mechanical and instrumentation and control design changes may be made by the contractor provided the effect of the change does not involve a change in a technical specification or an unreviewed safety question. Formal procedures shall be established for documenting important mechanical and instrumentation and control design changes.

c. When operation in the low-flow mode is to be used a formal configuration change shall be processed to do the following:

(1) Inhibit the scram action from the main flow scram.
(2) Inhibit the scram action from the core pressure differential switch.
(3) Activate two scram circuits that actuate if the cooling water flow rate is below 45 gpm.
(4) Change the scram set point on each neutron flux level safety channel from 1.6 MW to 16 kW.

When normal operation is to be restored a formal configuration change shall be processed to restore the system to its normal condition.

6.5 Reactor Operating and Maintenance Procedures

a. The reactor shall be operated in accordance with documented operating procedures. In no instance will the operating procedures designate authorization to operate the reactor in excess of any specification listed in this document. The procedures shall be adequate to ensure safe operation of the reactor but should not preclude the use of independent judgment and action should the situation require such. Detailed written procedures shall be provided for, but not limited to, the following:

(1) Emergency and abnormal conditions including evacuations.
(2) Reactor startup, operation, and shutdown.

(3) Installation and removal of fuel elements, control rods, and other components inside the reactor pressure vessel.
(4) Reactor safety system checks and calibrations.
(5) Maintenance of equipment important to the safe operation of the reactor.

b. A standard method shall be used to change operating procedures as necessary to ensure that all persons concerned are notified of the change and that a permanent record is made. Permanent procedure changes must be formally written and approved by at least two of the following senior staff members:

(1) Operations Division Director.
(2) Reactor Operations Department Head.
(3) TSF-HPRR Operations Supervisor.
(4) Senior Reactor Operator.

Temporary procedure changes that do not alter their original intent shall be made, when required, by issuing special operating instructions. Such special operating instructions shall be approved by two senior reactor operators.

c. Radiation control procedures shall be maintained and made available to all operations personnel.

d. There shall be a personnel head-count badge system in use covering all personnel entering the TSF area. Prior to operation it shall be established, in accordance with approved procedures, that all personnel in the area are in a safe location.

6.6 Action to be Taken in the Event a Safety Limit is Exceeded

In the event a safety limit is exceeded:

a. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the DOE.
b. An immediate report shall be made to the Office of Operational Safety.
c. A report shall be made no later than the next work day to DOE.

d. A report shall be made which shall include an analysis of the causes and the extent of possible resultant damage, effectiveness of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be sent to the Reactor Operations Review Committee and a similar report submitted to the DOE when authorization to resume operation of the reactor is sought.

6.7 <u>Action to be Taken in the Event of an Abnormal Occurrence</u>

In the event of an abnormal occurrence (see Glossary of Terms) the following action shall be taken:

a. The TSF-HPRR Operations Supervisor and other appropriate management personnel shall be notified and corrective action taken prior to resumption of the operation involved.

b. A report shall be made that shall include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence, in accordance with DOE Manual Chapter 0502.

c. Where required, a report shall be submitted to DOE.

6.8 <u>Actions to be Taken in Regard to Potential Degradations of a Reactor Shutdown System</u>

a. Immediate remedial actions required:

(1) Upon experiencing a Class 1 degradation of the reactor shutdown system, the reactor shall be shut down immediately by manual scram or other emergency backup means that may be necessary.

(2) Upon the discovery of a Class 2 degradation of the reactor shutdown system, the reactor shall be shut down immediately in an orderly (nonemergency) manner, except when the situation warrants more urgent shutdown action be taken.

(3) Upon the discovery of a Class 3 degradation of the reactor shutdown system, the contractor shall take the

applicable course of action below:

(a) For coincident logic systems, the degraded unit shall be promptly placed (or kept) in the tripped state until operability is restored, except for the brief time necessary to determine the operability of the redundant channels. If the action above would result in an automatic scram, this requirement should be satisfied instead by the prompt initiation of an orderly shutdown.

(b) For one-out-of-two or -three systems, where a bypass of a channel for short time periods is authorized in order to permit testing or repair, the bypassed condition may be retained. If operability cannot be regained by the end of the period authorized for bypass, the reactor shall be immediately shut down in an orderly manner, except when the situation warrants more urgent shutdown action be taken.

(4) Upon the discovery of a Class 4 degradation of the reactor shutdown system, the contractor shall ascertain the root cause of the degradation and implement appropriate corrective action designed to correct the specific degradation and to reduce the probability of similar occurrences.

b. Authorization for restartup of the reactor:

Following the occurrence of either a Class 1 or Class 2 degradation of the reactor shutdown system, authorization from DOE is required for restartup of the reactor.

c. Notification to DOE:

(1) In the event of a Class 1 degradation, ORNL shall verbally notify DOE immediately and provide a written report within five calendar days.

(2) In the event of a Class 2 degradation, the contractor shall verbally notify DOE as expeditiously as practical but within 24 hours and provide a written report within five calendar days.

(3) In the event of a Cla[illegible] degradation, the contractor shall provide a written report to DOE within five calendar days.

(4) In the event of a Class 4 degradation, the contractor shall include the occurrence in a written report to DOE provided no later than 30 days following the occurrence.

6.9 <u>Additional Reporting Requirements</u>

a. A report shall be made no later than the next work day to the Safety and Environmental Control Division, DOE, Oak Ridge Operations of the following conditions:

(1) Any release of radioactivity to the environment above the permissible limits specified in DOE Manual Chapter 0524.

(2) Any violation of a safety limit (see Sect. 2.1).

(3) Any exposures to personnel in controlled and uncontrolled areas that exceed the standards in DOE Manual Chapter 0524.

b. A report shall be made within three work days to DOE-ORO of any violation of the technical specifications.

6.10 <u>Plant Operating Records</u>

In addition to the requirements of applicable regulations, and in no way substituting therefor, records and logs shall be prepared of at least the following items and retained for a period of at least six years:

a. Normal plant operation.

b. Principal maintenance activities.

c. Abnormal occurrences.

d. Equipment and component surveillance activities required by the technical specifications.

e. Fuel inventories and transfers.

f. Experiments performed with the reactor.

g. Updated, corrected, and as-built drawings of the facility which shall be retained for the lifetime of the facility.

6.11 Review Committees

a. Reactor Operations Review Committee (RORC)

There shall be a Reactor Operations Review Committee (RORC) responsible for periodically conducting an independent safety review of the reactor facility. The members of the RORC shall be appointed by the Director of the Laboratory and shall not be directly involved in the operation of the reactor. The committee members shall collectively possess expertise in all areas of reactor operations and safety.

The RORC shall meet with the operating personnel as frequently as it deems necessary to keep informed of any operational problems or potential hazards. The committee shall conduct at least one formal review each year and the minutes of this review shall be reported in writing to the Director of the Laboratory. In compliance with the requirements of IAD-8401-7, the RORC shall review any proposed modifications that have safety significance. The RORC, which has the overall responsibility for reviewing experiments utilizing the TSR-II, may initiate an experiment review after it receives notice of action by the Engineering Physics Division Safety Review Committee (see below). A detailed description of the RORCs function is presented in Reference 15.

b. Engineering Physics Division Safety Review Committee (EPDSRC)

There shall be an Engineering Physics Division Safety Review Committee responsible for reviewing all new shielding experiments to be performed with the TSR-II. The committee shall be appointed, by memorandum, by the Director of the Engineering Physics Division. One member of the committee shall be a member of the Operations Division Technical Assistance Staff. The committee shall be responsible to the

Director of the Engineering Physics Division and shall review experiments from the standpoint of personnel and equipment safety. The committee shall, as it deems necessary, place limits upon any material, system, components, effluents, or operations that may present a hazard to personnel or to the reactor. The committee may recommend approval to the Head of the Nuclear Experiment and Data Evaluation Section of the Engineering Physics Division provided that it meets the specifications in Section 3.8. EPDSRC forwards information copies of action taken to the RORC which has overall responsibility for reviewing experiments using the TSR-II (see Sect. 6.11,a).

c. Criticality Committee

There shall be a Criticality Committee responsible for the review and approval of operations which involve handling, storage, transportation, and disposal of significant quantities of fissile material. The committee shall, on request, serve as a consulting group and provide assistance in problems involving criticality. The committee shall conduct an annual review of all areas containing significant amounts of fissile material to ensure that approved procedures are being followed. A detailed description of the committee's functions and method of review is presented in Reference 16.

6.12 Limitations on Reactor Operation

The reactor shall be operated so that the radiation levels to uncontrolled areas from routine reactor operation or under MCA conditions will always be within acceptable values. To meet this requirement the following limitations shall be in effect:

a. The reactor operation shall be scheduled so that the radiation dose at the boundary of the uncontrolled area shall be kept at a practical minimum, shall be within the Radiation Protection Standards, DOE Manual Chapter 0524, and shall be less than 100 millirems in any seven consecutive days. Dosimeters will be placed at monitoring stations at the boundary of the uncontrolled area to record the accumulated dose.

b. The total operation for any one set of fuel elements in the TSR-II shall be limited to 3000 MWh.

b. Integrated operation of the reactor during a 5-day period at power levels above 100 kW shall be limited to 75 MWh. If the limit of 75 MWh is reached in a 5-day period the reactor shall not be operated above 100 kW for 48 hours. If a limit of 72 MWh is reached in a 12-day period the reactor shall not be operated above 100 kW for 24 hours.

The afterheat analysis[17] indicates that the hottest point on the fuel plates would be 133^{o}F below the melting temperature even if the reactor suffered a Maximum Credible Accident after continuous operation at 1 MW to accumulate 75 MWh provided that the background afterheat is less than 1 kW. The delay required for the background afterheat to decay to 1 kW after 75 h at 1 MW is approximately 42 h. The delay required to reach 1 kW after 9 cycles of 8 h at 1 MW and 16 h off is approximately 21 h.

Even though melting will not occur under MCA conditions the 3000 MWh limitation would allow the low population zone boundary to be inside the minimum distance to the TSF exclusion fence even if 3.6% of the core were to melt.[18] The hottest portion of the core comprises only 0.39% of the core.[19]

6.13 <u>Hoist Operation Requirements</u>

To preclude the possibility of an accident that might result from overloading the hoisting equipment the hoists shall be operated by a certified reactor operator and the size and positioning of the loads moved with the hoisting equipment shall be in conformance with those outlined by the architect-engineer who was responsible for design and erection of the tower structure.[20] Conformance to the load limitations outlined in the referenced engineering report should preclude the possibility of an accident which might result from overloading the hoisting equipment. Operating procedures shall also specify that the reactor will not be elevated from a ground position when winds exceed 40 mph and that the hoists shall not be used to raise the reactor more than 50 ft above the ground unless it has been less than five years since the system has been load checked.

6.14 Tower Integrity

To ensure that the tower structure, guy cables, and foundations continue to meet design specificactions they shall be checked and inspected on a routine basis.

The checks and inspections shall include but not be limited to the following:

1. On a yearly basis the alignment of the towers shall be checked, the elevation of the tower bases and guy anchors shall be checked, the resistance to ground of lightning protection grounding system shall be measured to determine that it has not changed, and the condition of the towers, the tower guys, the tower bases, guy anchor bases, the electrical grounding system, and all connections shall be checked visually.

2. On a five-year basis the tower guy shall be checked for internal breakage or wear with eddy-current and magnetic induction measurements.

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REFERENCES

1. Correspondence from H. M. Roth, AEC, to A. M. Weinberg, ORNL, *Operating Authorization for the TSR-II*, (January 20, 1972).

2. J. Lewin and L. B. Holland, *TSR-II Heat Transfer*, ORNL/TM-1779 (July 26, 1967), p. 4.

3. Ref. 2, p. 27.

4. L. B. Holland, *Tower Shielding Reactor II Design and Operating Report: Vol. 1, Description*, ORNL/TM-2893, vol. 1, Rev. (February 16, 1971), p. 5.11.

5. F. T. Binford and C. C. Webster, *The Low Intensity Testing Reactor - Safety Analysis*, ORNL/TM-1924 (February 1968), pp. 69 and 78.

6. Ref. 2, pp. 7 and 9.

7. Ref. 5, pp. 69 and 71.

8. Ref. 5, p. 78.

9. L. B. Holland, *Tower Shielding Reactor II Design and Operating Report: Vol. 2, Safety Analysis*, ORNL/TM-2893, vol. 2, (October 7, 1970), pp. 3.2 and 3.3.

10. Ref. 9, pp. 3.2, 3.3, and Appendix B.

11. Ref. 9, p. 3.3 and Appendix A.

12. J. R. Tallackson, *Performance Tests of the ORNL Fast Safety System*, ORNL-3393 (August 28, 1963).

13. Ref. 9, Appendix A.

14. *ASME Boiler and Pressure Vessel Code: Section V111. Rules for Construction of Pressure Vessels*, The American Society of Mechanical Engineers, New York, 1956.

15. Office of Operational Safety, *Charter of the Reactor Operations Review Committee (RORC) of Oak Ridge National Laboratory*, (October 7, 1977).

16. Office of Operational Safety, *Charter of the Criticality Committee of Oak Ridge National Laboratory*, (October 1978).

17. Ref. 9, pp. 6.7 and 6.8.

18. Ref. 9, p. 7.1.

REFERENCES (Continued)

19. Ref. 9, p. 6.8.

20. Knappen-Tibbetts-Abbett-McCarthy, *Loading Criteria and Analysis for the Tower Shielding Facility*, (1953).