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NUCLEAR HEALTH AND SAFETY
TECHNICAL SPECIFICATIONS

December 7, 1961

The information contained in this document is a compilation of Technical Specifications applicable to the process limits for the eight existing Hanford Production Reactors. These Technical Specifications, together with other administrative procedures and controls, would define the limits of reactor operation and would establish administrative methods for obtaining approvals, reviews, and procedures for making changes. Further comments relating to these specifications should be addressed to W. D. Gilbert, 1760-D Building.

W. D. Gilbert

W. D. Gilbert
Process and Reactor Development
Research and Engineering Section

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By *CG-PR-2*

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I. FACILITY IDENTIFICATION

These Technical Specifications shall be applicable to the eight operating production reactor facilities, namely B, C, D, DR, F, H, KE, and KW (hereinafter referred to as the Hanford Production Reactors), located at the Hanford Site in the State of Washington.

II. REFERENCES TO DESCRIPTION OF FACILITIES

Basic descriptions of the reactor facilities are contained in the following reports. These reports are not to be considered a part of these technical specifications.

HDC-2225, C431-B Scope Document, 105-C Reactor Plant, H. L. Hollister, 6/1/51.

HDC-2524, Basic Design Specification for "X" Pile No. 1.10 (Twin K Reactors), G. M. Roy, 3/11/52.

HW-25892, Summary Report of Reactor Hazards for Twin 100-K Area Reactors, G. M. Roy, G. L. Locke, E. L. Armstrong, G. Sege, 10/10/52.

HW-44708, Vol. I Rev., Hazards Summary Report, Projects CG-558 and CG-600 Reactor Plant Modifications, R. E. Trumble, 3/6/57.

HW-44708, Vol. II, Hazards Summary Report, Projects CG-558 and CG-600 Reactor Plant Modifications, R. E. Trumble, 12/21/56.

III. OPERATING PROCEDURES

The Hanford Production Reactors will at all times be operated in accordance with the written procedures for Process Control. All operations will be conducted under the direct and personal supervision of technically qualified and designated personnel.

IV. OPERATING AND PERFORMANCE RESTRICTIONS

A. Reactivity

1. Maximum Loading Reactivity

The cold or hot reactivity of the water cooled loading⁽¹⁾ shall never exceed the combined negative reactivity worth associated with the

(1) The loading includes all fuel and integral fuel-target columns in the active zone of the reactor. Fuels may include both natural and enriched uranium as defined in Section F. Poison materials charged in separate columns or capable of insertion or discharge separate from adjacent fuel are not included in the loading as defined above.

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Horizontal Control Rod System, the xenon present in the reactor, inserted supplementary control devices, and poison pieces or poison columns under irradiation. That is, the reactivity in excess of that compensated by inserted supplementary and permanent poisons shall not exceed the strength of the Horizontal Control Rods, i.e., 1.5 percent $\Delta k/k$ in the B, D, DR, and F Reactors, and 2.5 percent $\Delta k/k$ in the C, E, KE, and KW Reactors.

Conditions more restrictive than the above may be required to fulfill Total Control requirements in the dry core as specified below.

2a. Total Control Criterion

The operation of the Hanford Production Reactors shall be such that in the event of any credible combination of events, including complete and permanent loss of coolant, the total available control strength of the combined reactor control systems will be sufficient to ensure that no reactivity state will develop which would significantly increase the consequences of the accident.

No reactivity state will be permitted to develop in the dry reactor after the scram and after any power excursion which may have accompanied the accident which would decrease the probability of restoring coolant to the reactor. That is, adequate control elements will be inserted or available for insertion into the reactor to ensure subcriticality immediately following a complete loss of water from the reactor core. A delayed neutron critical state is permitted after significant numbers of process tubes have been rendered useless as coolant channels. A prompt critical state shall never be permitted.

3a. Speed of Control Criterion

The operation of the Hanford Production Reactors shall be such that in the event of a power excursion resulting from any credible combination of events, including complete and permanent loss of coolant, the response of the reactor safety control system shall be sufficiently rapid to ensure that no energy state will develop during the excursion which would significantly increase the consequences of the accident.

The energy generated in a power excursion resulting from a credible accident as described in the referenced Hazards Summary Reports shall be insufficient to allow the surface temperature of the hottest fuel in the reactor to reach the melting point of uranium -- 1133 C. The excursion is considered terminated when the pile power level returns to its pre-accident value. Following this period, if coolant is not restored immediately, fuel melting would occur from heat generated by fission product decay, independent of a power excursion.

B. Reactor Control and Safety Elements

1. Horizontal Control Rods

- a. The maximum reactivity insertion rate due to the Horizontal Control Rod withdrawal will be limited mechanically to less than $5 \times 10^{-4} \Delta k/k$ per second.

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- b. If the control effectiveness of a Horizontal Control Rod is reduced by more than 10 percent due to poison burnout, the rod shall be replaced.
- c. Horizontal Control Rods withdrawn from the reactor normally are not to be included as a part of the control elements available for Total Control requirements as defined in Section A above.

2. Vertical Safety Rods

- a. For a cold startup, the reactivity insertion rate afforded by Vertical Safety Rod withdrawal shall be limited by procedural control to less than $2 \times 10^{-4} \Delta k/k$ per second.
- b. For a hot startup, the reactivity insertion rate afforded by Vertical Safety Rod withdrawal shall be limited mechanically to less than $2.5 \times 10^{-3} \Delta k/k$ per second.
- c. An interlock system shall prevent withdrawal of Vertical Safety Rods unless the Horizontal Control Rods are fully inserted.
- d. Vertical Safety Rods must meet insertion times as set forth in the Process Standards. These times, measured as the interval between first movement and 90 percent downward travel of the rod, are as follows:

<u>Reactor</u>	<u>Insertion Time Seconds</u>
B, D, F	2.4
DR, C, H	2.4
KE, KW	2.3

Rods not meeting this specification are considered inoperable.

At the power levels specified below in Section C, the number of inoperable Vertical Safety Rods shall not exceed one third of the total number of rods. In the B, D, and F Reactors, an inoperable rod shall have no more than one adjacent rod also inoperable. In the C, DR, H, KE and KW Reactors, an inoperable rod shall have no more than two adjacent rods also inoperable.

3. Ball 3X System

Small metallic or ceramic balls shall be available for insertion into the Vertical Safety Rod channels under emergency conditions wherein it becomes apparent that Vertical Safety Rod insertion may be impossible, that coolant loss is imminent, and that Total Control Strength should

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be available. The Ball 3X System, in conjunction with loaded supplementary control devices and inserted control and safety rods, shall be adequate for satisfying Total Control requirements.

4. Supplementary Control Devices

Supplementary control devices, such as poison splines, poison column control facilities, or supplementary control columns may be used to facilitate reactor operating control and may be utilized in satisfying Total Control requirements.

C. Power Level

1. The steady state power levels of the reactors shall not exceed the following values:

<u>Reactor</u>	<u>Power Level Limit - MW</u>
B, D, DR, F, H	2090
G	2310
KE, KV	4400

2. The reactors will be scrammed automatically by a high neutron flux level signal set as defined in Tables I-III.

D. Temperatures and Heat Flux

1. During steady state operation, the bulk outlet coolant temperature for all existing Hanford Production Reactors shall not exceed 95°C.
2. During steady state operation, process channel coolant outlet temperature shall not exceed 150°C.
3. During steady state operation, calculated maximum fuel cladding surface temperatures shall not exceed 175°C.
4. At the maximum steady state power levels defined in C.1 above, the calculated fuel surface heat flux at the point closest to burnout in the hottest process channel shall not exceed 70 percent of the burnout heat flux as predicted from established burnout correlations.
5. During steady state operation, the calculated maximum graphite temperature shall not exceed 850°C.
6. During reactor operation, including startup and shutdown, the graphite temperature shall be controlled such that stored energy released from the graphite will not result in substantial temperature increases in reactor core components.

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E. Reactor Scram Set Points

Nuclear and process control instrumentation shall be provided to automatically scram the reactor when preset control limits are exceeded. Tables I, II and III indicate the instrumentation, the number of monitored channels, the set points, and by-pass availability for each function capable of scrambling the reactor automatically.

F. Reactor Fuel Loadings

1. Production Scale Loads

- a. The standard fuel loadings of the reactors necessary to provide the reactivity as defined in Section A shall consist of natural uranium supplemented with 0.94 percent enriched uranium. The fuel elements shall be metallic uranium enclosed in metallic aluminum cans.
- b. Non-standard full scale loads for production of materials other than weapons grade plutonium shall not be utilized in the reactors without prior approval of the General Electric Company and the Commission. These non-standard loads will be subject to hazards reviews as required.

2. Test Scale Loads

Small test scale loads involving a minority of process channels in any reactor may be utilized for testing of fuel prototypes without prior concurrence of the AEC.

3. Shield Blanket Loadings

It shall be permissible to load the outermost ring of process channels with enriched lithium-aluminum poison pieces, and to provide additional enrichment to supply reactivity for reactor operation.

4. Limitations on Reactivity of Fuel Loadings

The combination of reactor fuel loadings as defined above and the control elements as defined in Section B shall not exceed the reactivity limitations as defined in Section A.

G. Reactor Coolant System

1. Except for the case of total, instantaneous loss of coolant flow, the decrease of coolant flow in individual process channels shall cause the reactor to be shut down automatically to prevent melting of the fuel cladding during the shutdown transient.
2. The reactor coolant flow rate during a reactor shutdown shall be controlled so that the bulk coolant outlet temperature does not exceed 90°C, and the minimum process channel flow rates and pressures shall not be less than that required to prevent melting of the fuel cladding.

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3. The coolant reservoir capacity uniquely available to individual reactors shall be as follows:

<u>Reactor</u>	<u>Individual Reactor Reservoir Capacity 10⁶ Gallons</u>
B	15.1
C	24.6
D	14.6
DR	10.6
E	15.1
F	15.1
KE	2.3
KF	2.3

In addition to the above, emergency coolant capacity may be available through the Export Water System, or from adjacent reactors in a dual reactor area.

4. Each reactor coolant system shall consist of three independent, reliable sources of coolant supply as follows, all three of which must be in service during reactor operation, and two of which must be in service during reactor shutdown. Maintenance during operation is permissible.

a. Primary Cooling System

The primary cooling system shall:

- 1) Be capable of providing adequate process channel cooling during all phases of normal reactor operation and shutdown.
- 2) Not be expected to provide the sole source of coolant under emergency conditions.
- 3) Be designed and operated such that the backup coolant systems will not be required to provide the sole source of coolant during any phase of normal operation, excluding the reactor shutdown condition.

b. Secondary Cooling System

The secondary cooling system shall:

- 1) Be capable of providing adequate process channel cooling indefinitely immediately following a reactor scram from full level, assuming instantaneous loss of power to the primary cooling system.
- 2) Be independent of both the primary and the last ditch cooling systems.

c. Last Ditch System

The last ditch cooling system shall:

- 1) Be capable of providing shutdown flow indefinitely, assuming instantaneous loss of power to the primary cooling system and concurrent loss of flow in the secondary cooling system.
- 2) Be independent of both the primary and secondary cooling systems.

The following definitions are appropriate for the interpretation of this specification:

- a. An independent system is one with power sources and pumping systems such that failure or cause of failure in one system cannot induce failure of the alternate system. In addition, failure of any single component within a system shall not be capable of rendering that system inadequate. (In practice, common use of selected valves and piping may be qualified as exceptions in meeting the requirements of this specification.)
- b. Adequate cooling is defined, as a minimum, as that required to maintain reactor process channel components below the melting point of the fuel jackets.
3. The reactor shall be scrammed upon loss of coolant as indicated in Tables I, II and III.

H. Reactor Confinement

Each reactor shall be provided with a confinement system designed to have primary value in reducing the local contamination from a single fuel element, a single tube or other minor accident. The confinement system will not provide substantial confinement of fission products in the event of a complete loss of cooling which would inevitably be followed by melt-down of the reactor.

As a minimum, the confinement system shall consist of the following two major components:

1. A fog spray system, whose chief function is to reduce the magnitude and effects of fuel element burning in the discharge area.
2. A filter system located in the reactor ventilation exhaust ducts, whose chief function is to reduce the concentration of particulate matter which may be contained in the reactor building ventilation exhaust before being vented to the atmosphere. The exhaust air will normally be passed through the filter at all times during reactor operation or shutdown.

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I. Test Facilities

Irradiation Test Facilities may be installed in the Hanford Production Reactors and operated in accordance with the approved Administrative Procedures established by the General Electric Company.

J. Code Compliance

The intent of National Codes and Standards as specifically designated shall be observed as minimum requirements, wherever applicable, in the design, construction, alteration, maintenance, and operation of plant facilities and the related procurement. In those cases where National Codes and Standards are not applicable, the design, construction, alterations, maintenance, and operation shall be guided by safe, economical, uniform designs consistent with the use intended and with sound engineering principles.

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TABLE I

Reactor Protection Set Points - K Reactors

<u>Function</u>	<u>No. of Channels Required to Trip</u>	<u>Bypasses</u>	<u>Safety Circuit Automatically Tripped</u>	<u>Set Point Limits</u>
High startup rate	2 of 3	Manual	No. 1 FN 1	Variable FN 2a, 2b
High neutron flux level	2 of 3	Manual	No. 1	Variable FN 3
Low reactor coolant pressure	1 of 2	Manual	No. 1	R.P. = 60 FN 4a
Rapid loss of reactor coolant pressure	1 of 2	Manual	No. 3	R.P. = 160 in less than 3.5 sec, or TORP less than 25 psi FN 5a
Low flow to individual process channel containing fissionable mat'l	1 of 1	Manual	No. 1	Variable FN 6
Neismoscope	2 of 2	Manual	No. 1	Contact spacing 2.6 millimeters
In-reactor test loops	see	see	see	FN 7
Electrical power loss	see	see	see	FN 8

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TABLE II

Reactor Protection Set Points - B Reactor

<u>Function</u>	<u>No. of Channels Required to Trip</u>	<u>Bypasses</u>	<u>Safety Circuit Automatically Tripped</u>	<u>Set Point Limits</u>
High startup rate	2 of 3	Manual	No. 1	Variable FN 2a, 2b
High neutron flux level	2 of 3	Manual	No. 1	Variable FN 3
Low reactor coolant pressure	1 of 2	Manual	No. 1	R.P. - 130 FN 4b
Rapid loss of reactor coolant pressure	1 of 2	Manual	No. 3	R.P. - 280 FN 5b
Low flow to individual process channel containing fissionable mat'l	1 of 1	Manual	No. 1	Variable FN 6
Seismoscope	2 of 2	Manual	No. 1	Contact spacing 2.6 millimeters
In-reactor test loops	...	...	...	FN 7
Electrical power loss	...	...	...	FN 8

TABLE III

Reactor Protection Set Points - C Reactor

<u>Function.</u>	<u>No. of Channels Required to Trip</u>	<u>Bypasses</u>	<u>Safety Circuit Automatically Tripped</u>	<u>Set Point Limits</u>
High startup rate	2 of 3	Manual	No. 1	Variable FN 2a, 2b
High neutron flux level	2 of 3	Manual	No. 1	Variable FN 3
Low reactor coolant pressure	1 of 2	Manual	No. 1	R.P. - 60 FN 4a
Rapid loss of reactor coolant pressure	1 of 2	Manual	No. 3	R.P. - 150 FN 5
Low flow to individual process channel containing fissionable mat'l	1 of 1	Manual	No. 1	Variable FN 6
Seismoscope	2 of 2	Manual	No. 1	Contact spacing 2.6 millimeters
In-reactor test loops	---	---	---	FN 7
Electrical power loss	---	---	---	FN 8

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FOOTNOTES TO TABLES I, II, and III

FN 1: No. 2 Safety Circuit trip consists of insertion of Horizontal Control Rods and Vertical Safety Rods.

No. 3 Safety Circuit trip consists of insertion of the poison balls into the Vertical Safety Rod channels.

FN 2a: No. 1 Safety Circuit trip results if any combination of high and/or low trips occurs in any two of three flux monitors, or if fewer than three flux monitors are in the safety circuit. High startup rate protection is provided by adjustment of the flux monitors to a range at least 100 times more sensitive during a hot startup, and at least 1,000 times more sensitive during a cold startup. Because of the low trip setting required during equilibrium operation, deliberate range adjustments are required during ensuing startup in order to prevent tripping of the No. 1 Safety Circuit. The ranges and setpoints of manual adjustments are controlled by established procedures.

FN 2b: Approach to critical is detected with subcritical monitors, backed up with a second instrument whose sensitivity is 500 watts power or less.

FN 3: No. 3 Safety Circuit trip results if any combination of high and/or low trips occurs in any two of three flux monitors, or if fewer than three flux monitors are energized in the safety circuit. High neutron flux trip points will be set as follows:

<u>Maximum Tube Outlet Temp.</u>	<u>Trip Setting % of Indication</u>
Above 95°C	110 maximum
90-95	115
85-90	120
80-85	125
Below 80	130

FN 4a: For K and C Reactors, No. 1 Safety Circuit trip results if process water riser pressure drops more than 60 psi below normal.

FN 4b: For B Reactor, No. 1 Safety Circuit trip results if process water riser pressure drops more than 130 psi below normal.

FN 5a: For K Reactors, No. 3 Safety Circuit trip results if process water pressure drops 100 psi below low Reactor Coolant Pressure set point in 3.5 seconds or less, or if the Top of Riser Pressure becomes less than 25 psi.

- FN 5b: For B Reactor, No. 3 Safety Circuit trip results if process water pressure drops 150 psi below Low Reactor Coolant Pressure set point in 5.0 seconds or less.
- FN 5c: For C Reactor, No. 3 Safety Circuit trip results if process water pressure drops 90 psi below Low Reactor Coolant Pressure set point in 5.5 seconds or less.
- FN 6: All process channels in the active zone of the reactor containing production materials are equipped with flow detection devices, the trip limits of which are variable depending upon location of the tubes in the reactor and other factors. Set point limits are set so that process tube flow reductions will not result in serious fuel element damage.
- FN 7: Special procedures are provided for operation of in-reactor test loops. Instrumentation and set points are variable depending on function and complexity of the loop. Set points will be provided to shut down the reactor under conditions established by approved Administrative Procedures or Production Test Authorizations.
- FN 8: Electrical power failure causes loss of power to the latch mechanism of the Vertical Safety Rods, resulting in insertion of the Vertical Safety Rods and reactor shutdown.

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