FINAL HAZARDS REPORT
FOR THE RCPA ELK RIVER REACTOR

MASTER

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1. INTRODUCTION

This document is a revision of the Final Hazards Report of July 8, 1960 that:

- (1) reflects the as-built condition of the plant; and
- (2) conforms with the technical specifications set forth in Appendix A to the Elk River Reactor Operating Authorization.

The organization and content of this revision are the same as those of the original document, with four exceptions:

- (1) the charts for initial startup and initial full-power organization and the resumes of personnel named thereon have been omitted;
- (2) the startup and power testing program description has been omitted;
- (3) the methods of analysis used to obtain physics data are referenced rather than described; and
- (4) original Appendix D has been completely revised to reflect experimental data published in the last two years.

2. SUMMARY

This section outlines the design and operation of the reactor and summarizes the results of the accident analysis. A more detailed description of the reactor and its associated systems and of the bases of design and analysis appear in other sections of this report. Table 2-1 lists the principal design and operating characteristics of the reactor.

TABLE 2-1

CHARACTERISTICS OF THE ELK RIVER REACTOR*

General Characteristics

Location	Elk River, Minnesota
Reactor-system designer	Allis-Chalmers Manufacturing Company
Architect-engineer	Sargent & Lundy, Engineers
Owner	USAEC
Operator-lessee	Rural Cooperative Power Association, Elk River, Minnesota
Reactor thermal output	58.2 Mw
Superheater thermal output	14.8 Mw
Total thermal output	73.0 Mw
Gross electric output	23.8 Mw
Net electric output	22.5 Mw
Net efficiency	30.8%
Operating pressure	922 psig
Operating temperature	536 F

Reactor Vessel:

inside height	25 ft
inside diameter	7 ft
wall thickness	3 in.
base material	carbon steel, Type A 302 B
cladding material	stainless steel, Type 304
minimum cladding thickness	0.109 in.
design pressure	1250 psig
test pressure	1875 psig

*Operating characteristics listed here are for reactor operation at 58.2 Mwt. Major components have been designed for future reactor operation at 116.4 Mwt.

Core:

configuration	right cylinder
height	60 in.
diameter	60 in.
volume fractions of core materials:	
steel	4.25%
zirconium	8.25%
water	68.25%
fuel	19.25%
number of fuel-assembly positions available	164
number of fuel assemblies in a complete loading	148

Fuel Assemblies:

fuel material	sintered thorium-uranium ceramic pellets
fuel-material density	94% of theoretical density
uranium-235 content of uranium in fuel material	93.13%
fuel-pellet diameter	0.407 in.
fuel-pellet length	0.500 in.
fuel-tube material	borated stainless steel
initial natural-boron content of fuel-tube material	600 ± 10 ppm, by weight
fuel-tube inside diameter	0.410 in.
fuel-tube wall thickness	0.020 in.
fuel-tube length (excluding end plugs)	62 in.
number of fuel pellets in each fuel tube	120
uranium-235 content of each fuel tube (regular)	46.1 g
number of fuel tubes in each fuel assembly	25
arrangement of fuel tubes in each fuel assembly	5 x 5 square array
number of removable fuel tubes in each fuel assembly	1
location of removable fuel tubes	center of each assembly
uranium-235 content of entire 148-regular element core	175.043 kg
maximum number of fuel assemblies that can be installed in core	164
fuel-tube surface temperature	561 F
temperature drop across fuel-tube wall	52 F

Control Rods - General Characteristics:

type	vertically disposed, downward scrambling, neutron-absorbing
total number	13
number of shim rods	12
number of regulating rods	1
weight of each rod	228 lb
maximum withdrawal speed	30 in./min
maximum scram time	2.0 sec

Control-Rod Poison Sections:

cross-sectional configuration	cruciform
nominal length	58 in.
nominal thickness	1/4 in.
nominal width across opposite blade edges	14-7/8 in.
material	borated Type-304 stainless steel
natural-boron content	2.2% ± 0.2%

Control-Rod Follower Sections:

cross-sectional configuration	cruciform
nominal length	45 in.
nominal thickness	1/4 in.
nominal width across opposite blade edges	14-1/2 in.
material	Zircaloy-2

Control-Rod Extension Sections:

cross-sectional configuration	cruciform
nominal length	80-1/2 in.
nominal thickness	1/4 in.
nominal width across opposite blade edges	3 in.
material	Type-304 stainless steel
drive-coupling type	bayonet

Reactivity Data:

maximum reactivity worth of all control rods (fully loaded, cold core) . .	0.183
maximum reactivity worth of any one control rod (fully loaded, cold core)	0.032
maximum reactivity worth of burnable poison dispersed in fuel tubes . . .	0.058
reactivity decrease caused by:	
temperature rise from 68 F to 536 F	0.013
Doppler effect of power rise from 0 to 58.2 Mw	0.003
steam-void formation (clean core)	0.039
equilibrium xenon poisoning	0.023
equilibrium samarium poisoning	0.007
burnup	0.020
unrodded overall temperature coefficient of	
reactivity (at 536 F)	$-1.5 \times 10^{-4}/^{\circ}\text{C}$
unrodded void-coefficient of reactivity at 0% void . .	$-1.2 \times 10^{-3}/\% \text{ void}$
unrodded void-coefficient of reactivity at 58.2 Mwt .	$-1.9 \times 10^{-3}/\% \text{ void}$

Nuclear Data:

average thermal-neutron flux at beginning of core life	1.28×10^{13} n/cm ² -sec
average thermal-neutron flux at end of core life	1.51×10^{13} n/cm ² -sec
maximum-to-average flux ratio	3.45
prompt neutron lifetime	3.53×10^{-5} sec
average neutron energy	0.059 ev

Heat-Transfer Data:

total volume of primary coolant	4800 gal
volume of primary coolant in reactor vessel	425 gal
depth of water above core during operation	7 ft 2 in.
condensate flow rate	516 gpm
condensate temperature	432 F
steam flow rate (at 58.2 Mwt)	258,000 lb/hr
average power density	23.3 kw/liter of core volume
average heat flux in core	90,700 Btu/hr-ft ²
maximum heat flux in core	264,000 Btu/hr-ft ²
burnout heat flux in core	960,000 Btu/hr-ft ²

2.1 DESIGN

The Elk River Reactor is of the indirect-cycle, natural-circulation boiling-water type. Initial operation will be at ~922 psig, (536 F) and 58.2 Mwt. It will produce 258,000 lb/hr of saturated steam, which will be condensed in evaporators to generate secondary steam. The condensed primary water passes through subcoolers before re-entering the reactor vessel. The secondary steam passes through a coal-fired superheater before entering a turbine-generator set. Initial rated net output of the plant will be 22.5 Mwe.

All permanent features of the plant, such as the containment building, the reactor vessel, and the shielding, are designed to accommodate future reactor operation at 116.4 Mwt. Space is available for the additional evaporators, subcoolers, and other auxiliary equipment that will be needed to double the reactor output.

The Elk River Reactor is similar to the Experimental Boiling-Water Reactor at the Argonne National Laboratory. The principal difference is in the type of fuel element employed.

The Elk River Reactor core contains 148 fuel elements, with provision for adding 16 more if necessary. Each fuel element consists of a square array of 25 cylindrical fuel tubes on a 0.75-in. pitch. The fuel tubes are of stainless steel and enclose sintered thorium-uranium fuel pellets. The steel of the tubes contains 600 ± 100 ppm of natural boron as burnable poison. The heat generated in the fuel is removed by water flowing up among the fuel tubes as a result of boiling and steam formation within the core.

The reactor and its primary auxiliaries are housed in a cylindrical gas-tight containment building designed to withstand the internal pressure from a vessel rupture that allows all primary-system water to flash to steam. Under these conditions, the building will prevent the escape of more than 0.1 percent of its gaseous contents in any 24-hr period. A storage tank at the top of the building contains 30,000 gal of demineralized water, which is used to flood the reactor and upper biological shield cavity during fuel transfer. If the reactor vessel ruptures, 15,000 gal of the storage-tank water are available for emergency cooling of the core, and 15,000 gal can be released to the interior of the building through a spray system to condense the steam and to wash down the fission products within the building.

The superheater and its auxiliaries are outside the containment building in an enclosure adjacent to the RCPA steam plant. The turbine-generator and condenser used with the reactor are in the RCPA steam-turbine building, along with two similar turbine-generator and condenser units operated from coal-fired boilers. The control room for the reactor and the nuclear steam system is in the front part of the turbine building.

The containment building is equipped with a personnel airlock for routine access, a personnel airlock for emergency use, and a large freight door used only when the reactor is shut down.

The reactor pressure vessel, the primary-system piping, and the tube sides of the evaporators and subcoolers were designed in accordance with Sec. VIII of the ASME Boiler and Pressure Vessel Code for a pressure of 1250 psig; the maximum contemplated operating pressure is 1150 psig. All primary-system components are protected against over-pressure by a relief valve that admits steam to the emergency condenser, which is set to open automatically at 1210 psig, and by relief valves (set at 1240 and 1250 psig) that open into the containment building.

Thirteen control rods and the burnable poison (in the form of natural boron in the stainless-steel fuel tubes) offset excess reactivity. The control rods are sufficient to make the reactor subcritical at all temperatures, even if the center control rod should fail to enter the core. The rods are cruciform, with a blade width of $14\frac{7}{8}$ in. and an active length of 58 in. The blades are made of stainless steel that contains

2 weight-percent of natural boron and are connected to Zircaloy followers. The total worth of all 13 rods in the clean core at room temperature is approximately 18.3 percent. If the center rod is withdrawn, the worth of the remaining 12 rods is approximately 14.9 percent. Burnable poison reduces the initial excess reactivity by approximately 5.75 percent.

A manually actuated sodium pentaborate injection system is provided in the unlikely event of multiple control-rod failure. It incorporates a tank of sodium pentaborate kept at a pressure of 2000 psig. If all control rods do not reach their fully inserted positions within 2 sec after an all-rod scram signal, the operator may actuate the poison injection system to render the core subcritical.

In normal operation the reactor heat is removed by condensing primary steam in the evaporators. After shutdown, decay heat is removed by the emergency condenser and the decay heat cooler. In the unlikely event of a loss of primary coolant, an emergency cooling system supplies cooling water to a spray ring within the reactor vessel. A line from the 30,000-gal overhead storage tank supplies low-pressure cooling water. A total of 15,000 gal of the stored water is available for cooling. If cooling water is needed for the core when the reactor vessel pressure is too high to permit low-pressure water flow, 25 gpm of high-pressure makeup water can be added to the vessel through the spray ring.

Like other boiling-water reactors, the Elk River Reactor has an inherent ability to offset reactivity increases by steam-void formation. Fuel temperature rise decreases reactivity and, hence, also contributes to the inherent safety of the reactor. Fuel temperature rise alone can offset a rapid increase in reactivity of as much as 1.60 percent, if the increase occurs when the core is at room temperature. A sudden reactivity increase of more than 1.60 percent would be required to damage the core.

The fuel elements have been designed so that the maximum local internal temperature of the fuel material under all normal operating conditions is more than 1000 F below its melting point. The maximum local heat flux is less than one-third of the burnout heat flux.* The heat-flux safety factor before a "dry-pipe" failure would occur is even greater than 3.

The reactor core should be relatively stable, and free of oscillations or chugging. The average power density in the core coolant (excluding control-rod channels) is 39.6 kw/liter, whereas that of the BORAX-IV reactor (operated at 300 psig) was 67.5 kw/liter, and that of the EBWR-1 (operated at 600 psig) was 65 kw/liter. Loop tests at the Argonne National Laboratory and the Atomic Products Department of the General Electric Company have shown that stability increases as the reactor operating pressure increases. GE tests indicate that the power-density threshold for chugging during operation at 936 psig is about 120 kw/liter. Thus the Elk River Reactor will operate at a power density well below that at which instability might be expected.

*See Appendix H, Burnout Heat Flux in ERR.

The maximum credible accident is assumed to be the loss of primary coolant from a rupture in the worst location (i.e., below the core) that permits expulsion of all water from the reactor vessel in 10 sec. The emergency spray system is actuated by the low-water-level signal. Assuming the worst likely spray distribution, 8.62 percent of the fuel tubes would eventually melt and release fission products. A delay of as much as 6.8 min in initiating the emergency cooling water flow would not result in failure of a greater percentage of the fuel tubes.

The pressure rise in the containment building as a result of primary-system rupture and steam flashing is 20.2 psig for 116-Mwt operation. This is less than the design and leak-test pressure of 21 psig. The core contains so little zirconium (8.25 percent, by volume) that a 25-percent zirconium-water reaction would only increase the pressure by 0.5 psi.

For analysis of the maximum credible accident, it was assumed that the core had been operating for one year at 116 Mwt, and that when any fuel tube melts all volatile fission products and 5 percent of the strontium contained within the tube are released. Melting of 8.62 percent of the fuel tubes would therefore release 8.62 percent of the volatile fission products and 0.431 percent of the strontium contained in the core. No more than 0.1 percent/day of the gaseous contents of the containment building will escape at an internal pressure of 21 psig, and direct radiation from the building is attenuated by a concrete shadow shield 2 ft. thick, which lines most of the inside of the building. The doses at various distances from the building that are due to direct radiation and sky shine, calculated on the foregoing assumptions, are listed in Table 2-2.

Fission products escaping from the containment building after the maximum credible accident will expose the downwind population to radiation. To estimate the maximum possible exposure, the most unfavorable meteorological conditions (i.e., strong inversion and low wind speed) were assumed. Table 2-2 lists the resulting whole-body doses from gamma radiation, the total integrated inhalation doses, and ground-deposition concentrations.

Liquid radioactive wastes will be collected and stored within the containment building. These wastes will be cleaned up with ion exchangers and discharged from the building only after analysis indicates that the activity is within 10 CFR 20 tolerances at the point of discharge. Solid wastes will also be stored within the building; after analysis indicates that their activity conforms to applicable shipping regulations, they will be shipped off-site for burial.

TABLE 2-2

GAMMA DOSES, INHALATION DOSES, AND GROUND-DEPOSITION
CONCENTRATIONS RESULTING FROM THE MAXIMUM CREDIBLE ACCIDENT*

Whole-body gamma doses (mr):	distance from containment building			
	200 ft	1/4 mile	1/2 mile	1 mile
from direct radiation during first hour .	692	36.5	8.9	--
from sky shine during first hour	9.2	2.4	1.2	--
from escaping fission products during first 8 hours	870.0	203.0	114.0	65
<u>Inhalation doses during first hour:</u>				
from iodine-131 (rem)	--	16.67	5.00	1.46
from iodine-132 (rem)	--	0.69	0.20	0.04
from iodine-133 (rem)	--	8.74	2.64	0.77
from iodine-135 (rem)	--	2.20	0.65	0.20
total from iodine isotopes (rem)		<u>28.30</u>	<u>8.49</u>	<u>2.47</u>
from strontium-89 (mrem)	--	88.6	26.4	0.8
from strontium-90 (mrem)	--	331.0	99.2	29.3
total from strontium isotopes (mrem) . .	--	<u>419.6</u>	<u>125.6</u>	<u>30.1</u>
<u>Concentrations resulting from ground deposition at the end of first hour ($\mu\text{C}/\text{m}^2$)</u>				
from iodine-131	--	11.02	88.2×10^{-3}	30.50×10^{-5}
from iodine-132	--	13.70	104.5×10^{-3}	30.50×10^{-5}
from iodine-133	--	24.60	194.8×10^{-3}	68.11×10^{-5}
from iodine-135	--	22.36	177.3×10^{-3}	62.01×10^{-5}
total from iodine isotopes	--	<u>71.68</u>	<u>564.8×10^{-3}</u>	<u>191.12×10^{-5}</u>
from strontium-89 ($\mu\text{C}/\text{m}^2$)	--	8.071	2.358	0.6628
from strontium-90 ($\mu\text{C}/\text{m}^2$)	--	0.270	0.791	0.0222
total from strontium isotopes ($\mu\text{C}/\text{m}^2$) .	--	<u>8.341</u>	<u>3.149</u>	<u>0.6850</u>

*All values in this table are based on the following assumptions:

reactor power level	116 Mwt
reactor operating time before accident	1 year
containment-building pressure after accident	20.2 psig
containment-building leak rate	0.1 percent/day
containment-building concrete shadow-shielding thickness	2 ft
fraction of fuel tubes that melts	8.6 percent
fraction of volatile fission products that escapes from each melted fuel tube	100 percent
fraction of strontium that escapes from each melted fuel tube	5 percent
fraction of total amount of volatile fission products that escapes from the core	8.6 percent
fraction of total amount of core strontium that escapes	0.43 percent
meteorological conditions	strong inversion, low wind speed

2.2 REACTOR OPERATION

2.2.1 Operational Safety Practices

The administrative rules set forth in Sec. 14.2 of ACNP-62566 (Technical Specifications Governing the Construction and Operation of the RCPA Elk River Reactor) specify the operational safety practices that will be enforced during operation of the ERR plant.

2.2.2 Radiological Health and Safety

The rules and procedures for radiation safety at the Elk River Facility are set forth in a manual entitled "Radiological Physics Safety Manual for the Elk River Reactor."

2.2.3 Reactor Safety Committee

The reactor safety committee will consist of specialists in reactor performance, radiological physics, power-plant operation, and reactor mechanical components. To carry out its responsibility, the committee will review procedures, observe tests as they are conducted, and examine the results. The committee will be at the site on 24-hr call. At least one man from each specialty will be required to serve on the committee.

2.2.4 Organization for Normal Power Operation

The RCPA organization described in the following subsections will be used for normal power operation after the responsibility is transferred from Allis-Chalmers to RCPA.

2.2.4.1 Nuclear Plant Manager. The nuclear plant manager ensures that the nuclear plant is operated and maintained safely and efficiently in accordance with limits established by AEC and RCPA generating requirements. He reports directly to the general manager, and advises him of production, fuel requirements, improvements, and personnel requirements in the nuclear plant.

The nuclear plant manager provides the manager of the nuclear contract department with budget forecasts and operational and technical information for reports required by AEC or management.

He also reviews and approves the recommendations of the nuclear plant superintendent regarding all maintenance schedules of nuclear plant equipment and reviews and approves the recommendations of the nuclear plant superintendent regarding operation and maintenance of the nuclear plant.

The nuclear plant manager directs the operation of the nuclear plant in generating steam for the conventional plant. Reactor safety requirement always has priority over steam requirements.

He also directs all fuel-handling operations and recommends future nuclear fuel requirements to the general manager, supervises on-the-job training of nuclear plant operating and maintenance personnel, and ensures that the nuclear plant is kept clean.

The nuclear plant manager reviews nuclear plant performance, improvements procedures, equipment operation, safety practices, and operating economy.

He maintains operating procedures, reviews operating logs and makes routine plant calculations required for safe operation of the nuclear plant. He approves all schedules for reactor work, vacation, and shift changes.

He approves requisitions for supplies and equipment for nuclear plant within approved budget and management policies and notifies the manager of the radiological physics department when the services of that department are required. He cooperates with other departments when the facilities of the nuclear plant are required to fulfill their objectives.

The nuclear plant manager also controls the cost of reactor steam production (by reviewing efficiency reports and heat-balance diagrams), reviews manhour schedules and performance statistics, approves all requisitions for operations and maintenance, and installs other controls necessary to discharge his responsibilities.

2.2.4.2 Nuclear Plant Superintendent. The nuclear plant superintendent assists the Nuclear Plant Manager in achieving the objectives of reactor operation and maintenance and consults with the Conventional Plant Superintendent regarding nuclear and conventional plant integration. He assumes the duties of nuclear plant manager in the absence of the latter. He reports to the nuclear plant manager and directs the reactor specialists, the reactor operators, and the reactor auxiliary operators.

He coordinates all activities necessary to maintain and operate the nuclear plant. He ensures that the activities of the reactor specialists, reactor operators, and reactor auxiliary operators conform with the established policies and procedures.

The nuclear plant superintendent directs the operation of nuclear plant facilities and sees that established procedures are followed. He prepares or directs the preparation of all requisitions for materials and supplies for the maintenance and continuous operation of the nuclear plant. He supervises the on-the-job training program set up by the nuclear plant manager and supervises the cleaning program for the nuclear plant.

The nuclear plant superintendent prepares procedures, checklists, operating logs, and reports. He reviews completed logs and instrument charts and consults with the Nuclear Instrumentation Supervisor regarding the accuracy and malfunctions of instruments. He maintains precise records of all fuel-assembly locations in the core, fuel-storage pit, and

storage racks; of fuel burnup; of the worth of each partially used fuel assembly; and of all other accountability factors. He prepares the shift schedules for nuclear plant personnel and arranges for sick leave, vacation, and planned days off by coordinating with the superintendent of the conventional plant. He assists in recording all recommended revisions or additions to the nuclear plant operations manual. In cooperation with the Maintenance Supervisor, he coordinates the maintenance of the nuclear plant and ensures that proper maintenance records are kept.

2.2.4.3 Reactor Specialist. The reactor specialist is responsible for startup, operation, and shutdown of the nuclear plant in accordance with established operating rules and procedures. He performs nuclear plant maintenance and fuel-element changes as assigned and instructs and supervises the reactor operators and the reactor auxiliary operators. He reports to the nuclear plant superintendent.

The reactor specialist sees that the reactor operating crews keep the plant clean and that the reactor plant is operated according to the procedures in the operations and maintenance manuals, the technical specifications, and the radiological physics manual.

He directs and participates in the preparation of all logs and reports concerning operation, maintenance, fuel elements, instruments, and equipment. He operates (or directs the operation of) the controls, switches, valves, pumps, and all other reactor plant facilities.

The reactor specialist personally supervises and instructs the reactor operators on assigned shifts. He reports all suspected malfunctions of equipment or instrumentation to the nuclear plant superintendent, and participates in nuclear plant maintenance and assigned fuel-element changes. He advises the Radiological Physics Department of potentially hazardous operations and obtains the required approval, and he prevents unauthorized personnel from entering restricted areas and reports all violations.

2.2.4.4 Reactor Operator. The reactor operator starts up, operates, and shuts down all auxiliaries directly associated with the reactor, in accordance with established operating rules and procedures and under the direction of the reactor specialists. He reports to the reactor specialists.

Under the direction of the reactor specialist, the reactor operator plans the operation of the nuclear plant according to the written procedures in the operations manual, the maintenance manual, the technical specifications, and the radiological physics manual.

The reactor operator personally carries out all assigned operations according to the instructions of the reactor specialist and reports all suspected malfunctions of equipment or instrumentation to the reactor specialist. He prevents unauthorized personnel from

entering restricted areas and reports any violations. He constantly surveys the control-room instrumentation, checks the containment building, and adjusts equipment as assigned. He initiates and completes all operating logs and checklists and records all information on equipment maintenance as assigned.

He cooperates with the turbine operator in coordinating the operation of the reactor plant with that of the conventional plant. He participates in nuclear plant maintenance and fuel-element changes and advises the reactor specialist of potentially hazardous operations.

2.2.4.5 Reactor Auxiliary Operator. The reactor auxiliary operator operates the separately fired superheater and other reactor plant facilities according to established operating rules and procedures under the direction of the reactor specialist. He reports to the reactor specialist.

He consults gauges, instruments, and coal scales to determine steam flow, temperature, and pressure and to determine coal consumption and furnace pressure. He adjusts the controls of the superheater and fuel-burning equipment to secure and maintain the maximum combustion efficiency. He makes running repairs to equipment and, during each shift, checks the operation of the induced-draft and forced-draft and forced-draft fans, the air preheater, and other related superheater equipment.

The reactor auxiliary operator also assumes all the duties of the reactor operator when the latter is checking the reactor plant.

2.2.5 Rural Cooperative Power Association Organization

This section describes the RCPA operating organization as it pertains to operation of the reactor. The principal RCPA staff members directly concerned with the administration, safety, and operation of the reactor are the managers of the following departments: the radiological physics department, the nuclear contract and engineering department, the conventional power department, the nuclear plant, the planning and electrical department, and the power operations department.

2.2.5.1 Duties and Responsibilities of Radiological Physics Department Manager. The radiological physics department manager is responsible to the general manager. It is his duty to conduct the RCPA Environmental Monitoring Program continuously through the five-year AEC contract operation period. It is his responsibility to recommend and enforce the policies and regulations needed for the protection of RCPA personnel and the general public.

If radiation hazards exist, he has complete authority to limit personnel and public activities in accordance with considerations of radiation exposure, radioactive contamination, decontamination, and general maintenance. He keeps records of radiation exposure of plant personnel. His authority is limited to safety and radiation protection; he has no authority in the normal operation of the reactor plant. He is in charge of waste disposal, radiochemical analysis and procedures, personnel monitoring, laundry of contaminated clothing, survey instruments, counting room procedures, in-plant surveys, and environmental surveys. He supervises performance of subcontract work related to radiological physics activities.

2.2.5.2 Duties and Responsibilities of the Nuclear Contract and Engineering Department Manager. The manager of the nuclear contract and engineering department reports to the general manager and is responsible for administering the contract between the USAEC and RCPA. He maintains liaison between all RCPA department managers, subcontractors, and the Chicago Operations Office of the USAEC, in accordance with established procedures. He lends engineering assistance in nuclear plant operation. He obtains from the various departments technical and operating data for reports to the USAEC, including reports for source and special nuclear accountability based on data obtained from the nuclear plant manager, the conventional power department manager, and the radiological physics department manager. He supervises nuclear plant engineering, construction, and subcontract work, and directs the activities of engineers and draftsmen doing work under the operating contract.

2.2.5.3 Duties and Responsibilities of the Conventional Power Department Manager and His Staff. The manager of the conventional power department reports to the general manager and is responsible for safe and economical operation of all RCPA-owned conventional steam, diesel, and electric generating facilities at Elk River.

He directs the conventional plant superintendent, who is responsible for coordinating all maintenance and operation of the conventional steam and electric generating facilities. The superintendent is also responsible for coordinating the operation of the turbo-electric facilities with the nuclear plant.

The maintenance supervisor, the operations supervisor, and the results engineer report to the conventional plant superintendent. The maintenance supervisor directs the activities of all maintenance personnel for the conventional and reactor plants. He coordinates and schedules his work with the assistant nuclear plant manager and consults the radiological physics department manager on matters within his department's area of responsibility.

Subordinate to the operations supervisor are the turbine operators, the relief turbine-reactor operators, firemen, relief firemen, assistance firemen, auxiliary operators, and relief auxiliary operators.

The operations supervisor sees that the conventional plant equipment is operated properly. He directs the relief turbine-reactor operators only on conventional plant operation and consults with the Nuclear Plant Manager on scheduling their availability as reactor plant operators.

The results engineer sees that the plant instrumentation and its various components are functioning to obtain optimum reactor and conventional steam plant operating efficiency. By tests and analyses, he determines the safest and most efficient methods by which to operate the conventional steam plant.

2.2.5.4. Duties and Responsibilities of the Nuclear Plant Manager and the Nuclear Plant Superintendent. The nuclear plant manager reports to the general manager. He is directly responsible for the safety and operation of the reactor plant. He will direct the operation of the reactor-plant console and the other equipment that pertains to the reactor plant. He has authority to shut down the reactor, and he is to operate the reactor for the maximum time consistent with the system demands, safety, and protection of the government-owned equipment. He is also responsible for the administrative and technical matters associated with the routine operation of the reactor plant. He will consult with the nuclear contract and engineering department and the radiological physics department regarding subcontracts, reactor plant improvements, technical reporting, engineering, radiation safety, and operating and contractual requirements.

The nuclear plant manager directs the nuclear plant superintendent, the reactor specialists, and all other reactor plant operating personnel. He consults with the maintenance supervisor on maintenance work required on the reactor plant.

The nuclear plant superintendent reports to the nuclear plant manager and is subordinate to the conventional plant superintendent insofar as integration of the general reactor plant operation with the conventional steam and electric plants is concerned.

The reactor specialists, the reactor operators, the relief turbine-reactor operators, and the reactor auxiliary operators (assigned to the reactor plant) report to the Nuclear Plant Superintendent and are responsible for operating the reactor plant safely and efficiently according to established operating rules and procedures.

2.2.5.5 Duties and Responsibilities of the Planning and Electrical Department Manager.

The planning and electrical department manager reports directly to the General Manager. His objective is to plan, develop, and administer policies governing the planning, engineering, operation, and maintenance of all electrical facilities so as to ensure the maximum continuity and quality of service in the most economical manner. This department is responsible for all nuclear instrumentation and control maintenance.

2.2.5.6 Duties and Responsibilities of the Power Systems Operations Department Manager.

The power systems operations department manager reports directly to the General Manager. He coordinates the movement of power from system-owned sources and other sources to member and non-member consumers in the most efficient and economical manner so as to make the greatest use of the Cooperative's electric facilities and to ensure continuity of service.

3. DESCRIPTION OF THE ELK RIVER REACTOR FACILITY

3.1 SITE

The Elk River Reactor is built on RCPA property near the Village of Elk River, Minnesota. Elk River is a small farm community, approximately 30 miles northwest of Minneapolis and St. Paul, Minnesota. Figure 1 shows the site in relation to the cities, towns, and villages of Minnesota within a radius of 40 miles.

3.1.1 Immediate Area

The reactor is approximately 1050 ft south of the corporate limits of the village of Elk River (see Dwg. 12E-01100). It is adjacent to the rear of RCPA's steam-electric generating plant, which houses the control room and the turbine-generator for the nuclear plant.

To the west of the steam-electric generating plant is a four-lane divided highway, U. S. Highway Nos. 10 & 52, which runs parallel to the Mississippi and lies between the river and the steam plant. The highway is approximately 600 ft west of the reactor and 250 ft east of the river. The river is approximately 500 ft wide at this point.

The average volume of automobile traffic flowing past the plant in 1960 on U. S. Highway Nos. 10 & 52 was 7400 vehicles/day.

U. S. Highway No. 169 leaves U. S. Highway Nos. 10 & 52 approximately 1/2 mile south of the site and goes north. At its point of closest approach, it passes just within 1/4 mile east of the reactor.

The existing buildings on RCPA-owned property consist of the steam plant, a 60 x 100 ft operations and maintenance building, a 240 ft x 100 ft warehouse and garage, a three-bedroom house with a separate garage, a 60 x 90 ft concrete-block building, and the RCPA headquarters building. The concrete-block building, 100 ft south of the reactor, is currently being used by the Fluidyne Corporation for government experimental work involving wind tunnels.

The maximum number of employees in each building at any one time is as follows: steam plant, 32; operations and maintenance building, 20; warehouse and garage, 20; headquarters building, 61; and Fluidyne Corporation building, 6.

The reactor-plant area is completely fenced in.

A sewage treatment plant for the village of Elk River is adjacent to the northern tip of the RCPA-owned land. The center of the four-acre sewage-plant site is approximately 800 ft from the reactor. Maintenance requirements for the plant are met by one man working approximately 2 hr/day.

RCPA property rises approximately 40 ft from U. S. Highway Nos. 10 & 52 to the railroad tracks. The area surrounding the reactor site is generally flat. Ground water is approximately 10 ft above the average level of the Mississippi, and drainage in the area is to the river. Surface drainage is southeasterly from the plant and runs to the river through a ditch southwest of the plant.

The main line of the Northern Pacific and the Great Northern Railroads runs east of the reactor in a northwest-southwest direction. The Northern Pacific Railroad operates three passenger trains each way every day; the Great Northern Railroad operates two passenger trains each way every day. The total number of passengers carried each day by the five passenger trains averages 1080. An average of eight freight trains travels each way every day. The volume of passenger travel and freight movement is heaviest during the summer. The tracks are 214 ft from the reactor at their point of closest approach.

3.1.2 Exclusion Area

The exclusion area is defined as that area upon which no residential, institutional, or other inhabited structures will be built other than those required by RCPA for its normal business operation. The exclusion area thus established includes the RCPA-owned property, the Village of Elk River sewage treatment plant, the highway right-of-way, the railroad right-of-way, and the Mississippi River.

The exclusion area is approximately 240 acres. The west side is bounded by the Mississippi River, 1/4 mile from the reactor. The east side is bounded by the eastern edge of the right-of-way for U. S. Highway No. 169, about 3/8 mile east of the reactor. The south side is bounded by the triangular section formed by the intersection of U. S. Highway Nos. 10 & 52 and U. S. Highway No. 169. The overpass at this intersection is 1/2 mile south of the reactor. The north side is bounded by the RCPA property limits and the northern boundary of the sewage-treatment plant site, which coincide with the southern boundary of the Village of Elk River.

The total area of RCPA-owned property is approximately 66 acres. The remainder of the exclusion area, including highway and railroad rights-of-way and the Mississippi River, covers about 174 acres. The minimum distance from the center of the reactor containment building to the boundary of the exclusion area is approximately 1050 ft.

3.1.3 Surrounding Area

The Village of Elk River is in the southeastern corner of Sherburne County, on the east bank of the Mississippi River.

Sherburne County is bordered on the southwest bank of the Mississippi by Wright County and on the north by Benton County. Anoka and Isanti counties form the eastern boundary. Sherburne County has a total population of 12,861, which includes 900 farms.

Farming in the surrounding area is diversified. Dairy farming has decreased; the trend is toward crop farming (principally corn, soybeans, and oats). Poultry farming is almost non-existent. Hog farming is not extensive but has increased.

The soil within a 5-mile radius of Elk River is a mixture of clay, sand, and gravel, which is subject to drought. Within the 5-mile radius the terrain is flat to the west and rolling to the east.

Elk River Township has a population of 986. Big Lake Township to the northwest of the village of Elk River has a population of 1126, and the village of Big Lake, 10 miles northwest of Elk River, has a population of 610. Elk River Township has approximately 100 farms; Big Lake Township has approximately 110 farms.

The village of Elk River had a population of 1763 in 1960. The average growth is approximately 36 people per year. The direction of residential growth is north from the north edge of town.

The municipal water supply is drawn from two wells at the south end of the town. One well is 312 ft deep; the other is 292 ft deep. Each has a pumping capacity of 750 gpm. The wells feed two storage tanks with a combined capacity of 233,500 gal.

The only industry in Elk River is the Elk River Concrete Products Manufacturing Company, which employs about 60 people. The business section of the village is limited to the usual commercial establishments.

Table 3-1 lists the population distribution within a 10-mile radius from the reactor site; Fig. 3.1 shows all major population centers within 40 miles of the site, and Fig. 3.2 is an aerial photograph of the area within a 3/8-mile radius.

The city of Anoka is the largest population center between 10 and 20 miles from the reactor site. It is located on U. S. Highway Nos. 10 & 52, 12 miles southeast of the site in Anoka County. The population of the city of Anoka is 10,562. Anoka County has a population of 85,916. Population growth in Anoka is heaviest toward the southeast; there is some growth toward the northwest but little or none toward the northeast and southwest. Growth generally follows the river and U. S. Highway No. 10.

Two of Minnesota's largest cities are St. Paul and Minneapolis, located slightly west of the eastern boundary and south of the central part of the state. St. Paul is in Ramsey County; Minneapolis is in Hennepin County. In 1960, Minneapolis had a population of 482,872 and St. Paul had a population of 313,411.

TABLE 3-1
POPULATION DISTRIBUTION*

	total population	population within x. miles from reactor site				
		x = 1/4	x = 1/2	x = 1	x = 5	x = 10
<u>Sherburne County</u>	12,861	0	10	15	305	1225
Village of Elk River	1,763	0	50	185	1763	1763
Village of Big Lake	610	0	0	0	0	250
<u>Wright County</u>	29,935					
Albertville	279	0	0	0	0	279
St. Michael	707	0	0	0	0	707
Dayton	58	0	0	0	33	58
<u>Hennepin County</u>	842,854					
Dayton	398	0	0	0	360	388
<u>Anoka County</u>	85,916	0	0	0	150	3845
Total		0	60	200	2611	8515

*Basis: 1960 Census.

3.2 REACTOR PLANT

3.2.1 Nuclear Plant Layout

The reactor is inside a cylindrical containment building 74-ft diameter by 115-ft tall. The reactor core is inside a pressure vessel, near the center of the lower part of the containment building. An 8.5-ft-thick concrete biological shield surrounds the reactor vessel. The entire primary water system is inside the containment building. The general arrangement of components is shown in Dwgs. 12E-01501 through 12E-01505.

Because the reactor is of the natural-circulation boiling-water type, the evaporators are higher than the reactor core. Two evaporators are presently installed, but there is space for two more to expand the capacity of the plant. The two subcoolers are slightly lower than the core, and there is also space for two additional subcoolers.

A fuel-assembly stainless-steel lined storage well is next to the reactor cavity in a concrete structure integral with the biological shield. If the reactor cavity is flooded with water, the reactor core can be refueled manually. A 30-ton crane and 5-ton auxiliary hook are provided for servicing and fuel handling. The crane runs on a circular rail inside the containment building.

An emergency condenser and a long exhaust duct are in the ceiling area of the containment building. The 30,000-gal water storage tank, which supplies water for shield flooding, emergency cooling, and building spray, is placed against the ceiling dome.

The primary water purification system is in the subbasement area. Control rods enter the pressure vessel through thimbles in the lower vessel head. A sub-reactor room has space for the long racks that drive the reactor control rods.

Piping between the reactor vessel and the primary water system penetrates the biological shield at only two angular positions. At these positions, the biological shield consists of removable concrete blocks and iron plates, and the primary water piping follows a tortuous path through the shielding to prevent radiation streaming.

3.2.2 Reactor Design

3.2.2.1 Core. The reactor core is contained within a cylindrical pressure vessel approximately 7 ft ID and 25 ft high (Dwg. 12D-01200). The core itself approximates a right circular cylinder 5 ft high and 5 ft in diameter. For initial operation, 148 fuel elements are installed, and there are positions for 16 more around the periphery (Dwg. 12D-04000). The 16 extra positions will be occupied by dummy fuel assemblies whenever it is not necessary to use them for fuel. The 13 control rods are in the form of crosses and move up and down within cavities in a shroud structure.

Cooling water flows in the vessel by natural circulation. It flows up through a 5-ft shroud above the core. The water level is kept at least 1 ft above the top of the shroud. Steam formed in the core disengages at this point, and the water circulates down through an 8-in. annulus outside the core to the bottom of the vessel.

The core is supported by a 6-in. core plate (Dwg. 12D-01200), which is supported and held down by a cylindrical barrel fastened to the bottom vessel head. The fuel-assembly nozzles are aligned by holes in the core plate. A Zircaloy shroud structure separates the core into cells, each of which holds three or four fuel assemblies (Dwg. 12D-04000).

Further core alignment is provided by the shroud and flat springs on the top fuel-assembly fittings. A 5-ft-high stainless-steel shroud structure above the core is fastened to the Zircaloy shroud within the core. The entire assembly is fastened to the bottom core plate.

3.2.2.2 Fuel Assembly. Drawing 12D-02000 shows a typical fuel assembly. Each assembly is made up of 25 Type-304 stainless-steel tubes in a 5 x 5 array. The stainless steel contains 600 ± 100 ppm of natural boron. The fuel tubes are approximately 62 in. long and contain thorium-uranium fuel pellets over 60 in. of their length. Each tube is held rigid by two sets of spacer straps that divide its length approximately into thirds. The straps are restrained from axial movement by small radial protrusions (called dimples) in the fuel tubes. The fuel tubes are aligned by a perforated support plate at each end. Each fuel tube slides freely in the support plates, but a nut on each end prevents it from slipping out. This arrangement permits the free expansion of each individual tube and thereby minimizes the tendency to bow that is associated with radial power gradients. The fuel-tube grid plates are welded into a top alignment box and a bottom nozzle. Alignment is maintained by the core support plate (in which the bottom nozzles are positioned) and by the flat springs in the top box. A plate inserted in the top box permits fuel-assembly handling.

The center tube of each fuel assembly is interchangeable with a borated-stainless-steel tube or a sample tube. The borated tubes permit adjusting the initial reactivity without the power distortion that would be caused by deeply inserted control rods. In the final calculations of fuel enrichment, it was assumed that 40 percent of the elements contained borated-stainless-steel tubes. The first core loading did not require boron pins; however, in eight elements the center pin was replaced with a sample tube.

Twenty-two "spiked" fuel assemblies were fabricated for the first core. The fuel tubes in a spiked assembly contain 5.23 percent enriched uranium, except for the removable center tube, which has the regular loading, 4.3 percent. This means that all removable tubes have the same loading, which avoids any confusion between spiked and regular tubes. The initial core loading consisted of 128 regular and 20 spiked elements.

Each fuel assembly has a clearly engraved serial number, prefixed by a letter indicating whether it is regular (R) or spiked (S). The identification of fuel assemblies is shown on Dwg. 12D-02000.

A fuel assembly containing lead instead of fuel was tested in a loop in which reactor flow conditions were simulated in order to investigate the stability of the tubes and their tendency to vibrate. No vibration was detected. The results of this experiment are presented in "Fuel Element Flow Test - Elk River Reactor" (June 1, 1960).

The fuel pellets consist of thoria (ThO_2), to which urania (UO_2) enriched to 93.13 percent in uranium-235 is added. The mixture forms a solid solution, which, owing to the low urania concentration, melts at a temperature very near 5900 F. The pellets are nominally 0.407 in. in OD and 0.500 in. long. They are placed in stainless-steel tubes 0.410 in. in ID and 0.020 in. thick. There are 120 pellets in each tube, in a helium atmosphere, for a total of 440,000 pellets in 148 assemblies. The initial core loading of 148 fuel assemblies contains 7800 lb of thoria and 414 lb of urania. The total metal-to-water ratio of the core is 0.465.

The fuel pellets have a density of 94 percent of the theoretical maximum. This density is obtained by pressing a ball-milled mixture of thoria and urania powders combined with a binder. After pelletizing they are sintered or fired at 1700 C in air.

The average total atomic burnup fraction is estimated at 0.22 percent, which corresponds to an average fuel exposure of 9500 Mwd/MT.

3.2.2.3 Control Rods and Drive Mechanisms. The reactor is controlled by 13 cruciform control rods that move vertically within the shroud structure.

The blades consist of a 14-7/8-by-14-7/8-in. cruciform with a poison length of 58 in. The nuclear poison is stainless steel, containing 2.2 - 0.2 percent natural boron. The poison section is fastened to a 45-in.-long Zircaloy follower, which prevents water-gap flux peaking. A 14-1/2-by-14-1/2-in. Zircaloy cruciform connects the Zircaloy follower to a stainless-steel shaft. The stainless-steel shaft terminates in a bayonet coupling. A fitting at the top of the blade permits lifting. Each control rod weighs approximately 228 lb.

The control-rod drives are attached to thimbles extending down from the underside of the reactor vessel. The arrangement of the 13 drives is shown in Dwg. 12D-01000 (sheet 7). The pressure housing and seal are attached to and supported by the thimbles. The drive motors and instrumentation are supported by platforms around the periphery of the subreactor room. A jointed shaft connects the drive unit to the shaft penetrating the pressure housing. A magnetic coupling is in the connecting line. De-energizing the magnets releases (scrams) the rods, which fall into the core by gravity.

The stability of the control rods under scram conditions was tested; the test results are presented in "Control Rod Drop Tests - Elk River Reactor." No signs of rod failure were noted after cycling for 2762 scrams.

The rod drive mechanisms are of the rack-and-pinion type (Fig. 3.3), and were supplied by ALCO Products Inc., a firm which has extensive research-and-development background on this type of drive and has applied it successfully in the APPR. ALCO has also designed and manufactured a prototype rack-and-pinion drive for the EBWR, which

was tested under various pressures and temperatures. These tests led to modifications that improved the efficiency of the drives. The ERR drives are patterned after the modified EBWR prototype drive. Thus the ERR drives are conventional and are expected to give trouble-free service.

The pinion shaft extends through a labyrinth pressure-breakdown seal. Makeup water is introduced into the seal at less than 120 F so that it will be adequately cooled and flushed at all times.

The control rod has a stroke of 56 in. and a speed of 30 in./min. The scram time from receipt of signal to the arrival of the rods within 5 in. of bottoming is 1.2 sec (max). This time comprises the 0.05-sec delay from reception of the scram signal to the start of free fall and the 1.15 sec of free fall. An additional 0.8 sec is involved in deceleration in the dashpot to give a total scram time of 2.0 sec (max). The control rods are decelerated at a rate not exceeding 270 ft/sec^2 .

There is a position indicating unit for each drive. The unit is connected to a reactor-console indicator. The unit is accurate to within 0.05 in.

As shown in Dwg. 12D-01000 (sheet 6), the center rod is used as a regulating rod, and the others as shim rods.

In the establishment of initial criticality, and during power increases, the regulating rod remains fully inserted until the twelve-shim-rod bank is fully withdrawn. The center regulating rod is then withdrawn to compensate for reactivity requirements for power increase and/or xenon buildup. To reduce power, the center rod is first inserted to its full-in position; then the twelve-shim-rod bank is inserted. The control system provides for the withdrawal of only one rod at a time. A four-rod scram will involve the insertion of rods 3, 4, 8, and 9 (Dwg. 12D-01000, sheet 6).

3.2.2.4. Reactor Vessel. The reactor vessel (Dwg. 12D-01201) is a right circular cylinder approximately 7-ft in ID and 25-ft-high, with elliptical heads. The top head is fastened with a bolted flange joint; the bottom head is welded to the shell. The vessel was designed, fabricated, and inspected in accordance with Sec. VIII of the ASME Boiler and Pressure Vessel Code for a maximum allowable working pressure of 1250 psig and a maximum allowable working temperature of 650 F.

The base material of the vessel is Type-A302 Grade B carbon steel, 3 in. thick. The vessel is internally clad with Type-304 stainless steel of 0.109 in. minimum thickness.

The vessel is designed to withstand transient stresses and local loads, as well as steady-state pressure stresses. Piping reactions of 200,000 in.-lb of torque and 1500 lb of thrust on the 8-, 10-, and 16-in. nozzles are provided for. Piping flexibility prevents

these values from being exceeded. The thermal shock of a 35 F steam-temperature change over a 15-sec period is also provided for. This is the thermal shock that would be associated with reactor scram caused by holding the secondary steam pressure constant and varying the primary steam pressure with load. Each control-rod nozzle must support a load of 1596 lb, because the 228-lb control rods are decelerated at the rate of 7 g. To minimize transient thermal stresses, the maximum vessel-wall heating and cooling rate, except during scram, is limited to 150 F/hr.

The vessel is supported by pads and local supports about 2 ft below the steam outlet nozzles. This support permits downward thermal expansion of the vessel. The nozzles penetrating the bottom shielding plug have clearances and a bellows seal to accommodate this expansion. The supports are designed to bear the weight of the vessel and its internal components (200,000 lb), the 19-ft head of water associated with refueling (76,800 lb), and the thimble loads that are due to deceleration of the control rods and the weight of the rod drives (22,100 lb). The supports are therefore designed for a total load of 278,800 lb. A box frame of I-beams resting on columns in the biological shield supports the vessel pedestals.

The vessel internal components include a thermal shield, a spray ring, a feedwater flow distribution ring, an inverted cone baffle, and a steam baffle. A thermal shield of plain Type-304 stainless steel 1 in. thick is inside the vessel 1/2 in. from the wall. The thermal shield extends from below the core nearly to the top of the upper shroud structure.

There is a 9-ft-high steam dome in the top of the vessel during normal operation. Together with the steam baffle this permits steam-water separation and minimizes carryunder.

The vessel is fitted with two 10-in. steam outlet nozzles, two 8-in. feedwater inlet nozzles, two 16-in. forced-circulation inlet nozzles, two 16-in. forced-circulation outlet nozzles, four 12-in. top-head access nozzles, 13 4-in. control-rod nozzles, three 1-1/2-in. liquid-level nozzles, one 1-1/2-in. spray-ring nozzle, and one 2-1/2-in. boric-acid injection nozzle. The 8-in., 10-in., 12-in., and 16-in. nozzles are made of carbon steel with stainless-steel internal claddings; the other nozzles are all stainless steel. The two 8-in. feedwater inlet nozzles are fitted with thermal-shock shields.

The forced-circulation nozzles are for the possible future use of pumps to recirculate the primary coolant through the core. These nozzles are capped off at the outside edge of the biological shield for initial operation. For forced circulation it will also be necessary to put a washer-like plate at the bottom of the annular downcomer and restriction caps on the control-rod shrouds and thermal shield, as provided in the design.

3.2.2.5 Vessel Insulation. Stainless-steel wool thermal insulation 3 in. thick is fastened to the outside of the pressure vessel and protruding nozzles. It is held in place by steel bands and Nelson studs attached to the vessel. This type of insulation was selected because of its good resistance to radiation damage and corrosion. The insulation will keep the vessel wall temperature approximately 40 F higher on the outside than on the inside during 116-Mwt operation.

3.2.2.6 Shielding. A 3-in. air gap separates the reactor-vessel insulation from the lead thermal shield (Dwgs. 12D-01201 and 12D-06500). The lead thermal shield is between the air gap and the biological shield. The lead attenuates the gamma leakage so that the concrete will not be excessively heated.

The thermal shield consists of a sandwich of 3 in. of cast lead with 0.75 in. of steel cladding on each face. The steel cladding and lead are bonded to eliminate air pockets. The thermal shield is in the form of a cylinder, extending from just below the bottom head to just below the vessel support pads. Stainless-steel cooling coils, 5/8 in. in OD with 0.040-in.-thick walls, are embedded in the lead.

An additional lead sandwich in the form of a plate covers the inner face of the bottom plug. Stainless-steel cooling coils 5/8 in. in OD with 0.040-in.-thick walls are also embedded in the lead of this shield.

The biological shield has a minimum thickness of 8.5 ft in the radial direction and is designed to reduce radiation levels outside the shield for operation at 116 Mwt to 2.5 mr/hr. It is composed of ordinary concrete, i.e., it has a density of 2.5 g/cc ($\sim 156 \text{ lb/ft}^3$) and a composition of 15.0 percent Portland cement, 34.2 percent sand, 43.3 percent gravel, and 7.5 percent water.

The upper and lower biological shield plugs are composed of high-density concrete. The lower plug, which is 4 ft thick, was put in place during construction. The upper plug is made up of three 16-in.-thick removable slabs, with provisions for adding one 12-in.-thick slab, if required. Both plugs limit the radiation dose at 116 Mwt to less than 2.5 mr/hr. The heavy concrete has a density of 4.55 g/cc (285 lb/ft^3) and a composition of 10.1 percent Portland cement, 31.9 percent magnetite, 53.6 percent iron punchings, and 4.4 percent water.

The allowable floor loadings permit adding more biological shielding, if necessary, by stacking concrete blocks around the face of the shield. The thickness of the radial shield could be increased by 1 ft of either ordinary or heavy concrete.

The fuel-assembly storage well can hold a complete core loading of irradiated fuel. An ordinary-concrete shield 5 ft thick surrounds the well to give ample protection from radiation. The depth of water in the well during normal fuel storage (17 ft 7-1/2 in.) provides adequate shielding to keep the radiation level directly above the pool at or below 2.5 mr/hr.

All major sources of radioactivity (such as reactor-water purification-system filters and ion exchangers) are shielded by concrete.

3.2.2.7 Fuel Handling and Storage. Fresh and irradiated fuel is transferred underwater to and from the reactor with hand tools (see Dwg. 12D-08000, sheets 1-4). The concrete shield slabs above the reactor are removable, and the cavity itself is lined with steel so that it can be flooded with water. To prevent leakage from the cavity, a flexible steel membrane connects the steel liner to the shell flange of the reactor vessel. The fuel element storage well is lined with 10-gauge stainless steel and is leak tight.

During refueling, the water level in the reactor cavity and fuel storage well is raised to within 6 in. of the top of the cavity by permitting water to flow by gravity from the 30,000-gal overhead storage tank into the reactor cavity. A 15-ft-deep canal joining the top of the reactor cavity and the fuel storage well causes the water level to be the same in each. There is a minimum of 8 ft of water above the end of any fuel assembly while it is removed from the vessel and passed along the canal. The doses during the entire transfer operation will not exceed weekly tolerance. Fuel assemblies are passed back and forth between the storage well and reactor cavity via the canal. The storage-well side of the canal contains a gate that may be closed to permit different water heights in the storage well and reactor cavity. The gate is used only when the fuel shipping casks are being filled, which is done while the reactor is operating. By using the gate, the water level in the storage well can be raised to 32 ft 6 in. to provide adequate shielding while fuel assemblies are being placed in the shipping cask.

A bridge riding on rails and driven by a hand crank runs across the top of the reactor cavity and fuel storage well. Mounted on the bridge is a jib crane, whose boom can rotate 360 deg. A load-carrying electrical cable is suspended from this boom on a trolley. One end of the cable is fastened to a hand winch on the crane and the other end is fastened to the fuel-transfer tool. By maneuvering the bridge and jib crane, the fuel-transfer tool can be suspended over any fuel-assembly position in the core or storage well. When in position, it is lowered into the core by hand cranking.

The tool engages a hole in the top of the fuel assembly and is locked in place by three stainless-steel balls, which are forced outward by an internal electromechanically actuated plunger, controlled from the bridge. When the tool is locked in place with the fuel assembly suspended from it, it is impossible to unlock the tool and drop the fuel assembly during transfer; the assembly must be resting on something before the tool can be unlocked.

The assembly is then shuttled through the canal and lowered into racks in the storage well. The racks ensure a geometric arrangement that avoids a critical mass; in addition, a 1/8-in. sheet of boral separates adjacent rows of stored fuel assemblies.

A space in the center of the storage well is reserved for the fuel shipping cask. The floor of the well is lined with stainless steel 3/8 in. thick, to prevent scuffing by the cask. Spent fuel assemblies are allowed to cool for 120 days and then are transferred underwater to a 19-assembly shipping cask. More than 17 ft of water is above the shipping cask, reducing the maximum dose during transfer to 2.5 mr/hr. A lead cover is locked on the cask top underwater, and the cask is then lifted by the 30-ton building crane out of the storage well onto a transfer dolly, which carries it through the personnel airlock to a loading platform. Here it is loaded on a truck or train for shipment to the reprocessing plant.

New fuel assemblies are stored in locked cabinets next to the reactor and pool shielding walls on the main floor. The fuel-transfer equipment can pick up a new assembly from the cabinets and transfer it directly to the core.

The reactor-vessel top head is removed for refueling, and underwater lights are lowered into the core area to aid in viewing. Head studs are loosened with a hydraulic bolt tensioner which elongates the studs, allowing them to be tightened manually. Release of hydraulic pressure results in pretensioning of the studs without the galling and excessive torque that usually accompany the impact-wrench technique.

3.2.3 Primary Water System

3.2.3.1 Flowsheet. The flow path for the primary water system is shown on Dwg. 12E-09006. During 58.2-Mwt operation, 258,000 lb/hr of steam are generated in the reactor at 936 psig and 538 F. This steam leaves the reactor vessel through two 10-in. nozzles. In the evaporators, the steam is condensed and the noncondensable gases are removed. The condensate is drained via sloped return lines, to the subcoolers, where it is cooled to 432 F. It then flows to the feedwater distribution ring in the downcomer within the vessel. The condensate is mixed with the naturally recirculating water in the vessel to subcool the water entering the core by more than 17 F.

The system incorporates an emergency condenser, to which primary steam is admitted if its pressure exceeds 1210 psig. The condenser can handle 258,000 lb/hr of steam. However, an automatic flow-control valve in the condensate return line permits only 6 percent of this capacity.

The reactor plant is designed for 1250 psig and potential operation at 116.4 Mwt. However, in order to operate at 116.4 Mwt, additional primary water system equipment, such as evaporators and subcoolers, will be necessary.

3.2.3.2 Evaporators. The two evaporators are within the containment building above the reactor core, to permit natural recirculation of condensate. They are similar to the reboilers used in the chemical industry. Primary steam enters the upper half of each evaporator head and condenses in a bundle of horizontal U-tubes. Condensate is collected in

the lower half of each head and drains by gravity to the tube side of its associated sub-cooler. Secondary feedwater from the shell side of each subcooler flows to the shell side of its associated evaporator. The secondary steam formed in the evaporators is taken off to the superheater through moisture separators.

The design, construction, and testing of the evaporators conform to Sec. VIII of the ASME Boiler and Pressure Vessel Code. The design pressures are 1250 psig for the head and tubes and 1000 psig for the shell. The tubes are made of Type-304 stainless steel, and the shell and header are made of A-212 Grade B steel with a Type-304 stainless-steel lining on the inside of the channel cover. The tubes are seal-welded to the tube sheet, and the entire unit is welded to form an integral leaktight vessel.

Each evaporator shell is 50 in. in ID and 23 ft 11 in. long. Each contains 642 U-tubes with an OD of 3/4 in. and 16 BWG walls on a 15/16-in. triangular pitch. Primary steam enters through an 8-in. nozzle and the condensate is removed through a 6-in. nozzle. Secondary feedwater enters through a 4-in. nozzle and steam leaves the shell through a 6-in. nozzle. A Wright-Austin steam separator in the outlet pipe ensures secondary steam with a quality of 99.7 percent for the superheater. A secondary steam quality greater than 99.7 percent is obtained by changing the evaporator-indicated water level according to changes in power level. Connections are provided on the shell side for hand holes, gauge glass, relief valves, vent lines, blowdown lines, and instruments. There is also a vent line on the tube side outlet channel. A small quantity of primary steam is allowed to escape through this line to sweep out the noncondensable gases formed by the radiolysis of water in the core.

Water level on the shell side is maintained by a three-element feedwater controller. An electric valve in the inlet feedwater line is automatically positioned to maintain water level.

The performance of each evaporator is based upon initial operation with a primary flow rate of 129,000 lb/hr and a secondary flow rate of 112,500 lb/hr. Primary steam enters at 536 F and condensate leaves at the same temperature. Secondary feedwater enters at 463.8 F and saturated steam leaves at 505.5 F.

3.2.3.3 Subcoolers. The saturated primary water leaving the evaporators enters the tube sides of the subcoolers through 6-in. lines at 528 F and leaves the subcoolers at 432 F. The secondary feedwater from the RCPA fourth-stage heater, under forced circulation, enters the shell sides of the subcoolers through 4-in. lines at 360 F. It leaves the subcoolers at 467 F and flows to the shell sides of the evaporators.

The subcoolers are of the shell-and-tube type and have U-tubes and two-pass shell sides. The shell is 18 in. in OD and has an over-all length of 15 ft 4-1/4 in. Each subcooler shell contains 70 U-tubes with an OD of 3/4 in. and 16 BWG walls on a 15/16 in. triangular pitch. Design, construction, and testing conform to Sec. VIII of the ASME Boiler and Pressure Vessel Code.

The tubes, channel headers, and tube sheet are made of Type-304 stainless steel. Carbon steel is used for the shells and shell-side bafflings. The design pressures are 1250 psig on the tube side and 1000 psig on the shell side. The tubes are seal-welded to the tube sheet and unit welded into an integral leaktight exchanger. Drain connections are provided for both the shell and tube sides, and there is a 1-in. vent line on the shell side.

The performance of each subcooler is based on initial operation with a tube-side flow rate of 129,000 lb/hr and a shell-side flow rate of 112,500 lb/hr.

The subcoolers are lower than the water level in the reactor vessel. Hence, the tubes are always filled with primary water.

3.2.3.4. Emergency Condenser. The emergency condenser is in the upper dome area of the containment building. An 8-in. line leads from the steam line to each evaporator and joins at a 10-in. line connected to the emergency condenser. The reactor steam goes to the tube side of the condenser, and service water is fed to the shell side. Reactor condensate returns by gravity through an 8-in. line, then splits to go to 2 ft-6 in. lines before going to the primary feedwater return lines. The condensate is of the reboiler type; in design and construction it is similar to the evaporators.

The emergency condenser serves several functions. A scram occurs when the primary steam pressure increases to 975 psig. The operator can then open a pneumatic-operated valve to pass steam to the emergency condenser. If the operator does not open the valve, it will open automatically at 1210 psig. The condenser will also be used to dissipate the decay heat from the reactor when the plant is shut down and to accept the reactor heat output during certain tests.

General service water is fed to the shell side of the emergency condenser. Vapor generated in the shell is vented to the atmosphere through a 24-in. vent line. The flow of water to the shell side is controlled automatically by a feedwater valve actuated by a signal from a liquid-level controller on the condenser.

The design, construction, and testing of the condenser conform to Sec. VIII of the ASME Boiler and Pressure Vessel Code. The design pressure for the tube side is 1250 psig. The shell side is designed for a maximum internal operating pressure of 50 psig and an external pressure of 21 psig. The shell, tube sheets, and nozzles are made of carbon steel. The water boxes, nozzles, and primary system side of the tube sheets are Monel clad. The tubes are 3/4 in. in diameter and consist of 70 percent copper and 30 percent nickel. The tubes are seal-welded into the tube sheet. Tube inlet and outlet nozzles are 10 in. in diameter. The feedwater connections to the shell are 6 in. in diameter, and the vapor vent, 24 in. There are connections for a shell-side drain, gauge glasses, a blowdown line, handholes, and instruments.

The performance of the emergency condenser is based on a flow of 258,000 lb/hr of steam on the tube side at 950 psia and a secondary steam output of 149,000 lb/hr. The secondary feedwater enters at 50 F and steam is generated at 212 F and 15 psia.

3.2.3.5 Startup Heater. A startup heater heats the primary water from 70 F to ~450 F. A temperature switch on the startup heater line is interlocked with the reactor control system to prevent rod withdrawal until the reactor water temperature reaches 425 F.

Water flows from the reactor vessel through the 2-1/2-in. boric acid injection nozzle and the two lower 16-in. forced-circulation nozzles through the decay-heat cooling pump to the startup heater and is returned to the reactor vessel through the feedwater distribution ring. Saturated steam at 600 psig is supplied to the shell side of the startup heater, where it condenses and heats the primary water. The condensate is returned to the RCPA steam plant through the high-pressure blowdown line.

The startup heater is 14 in. in outside diameter across the shell and 7 ft-11 in. long. It contains 58-3/4-in.-OD U-tubes with 18 BWG walls. The water box, tube sheet, tubes, and nozzles are made of Type-304 stainless steel; the shell side is made of carbon steel. The tube-side design pressure is 1250 psig, and the shell-side design pressure, 700 psig. The tubes are seal-welded into the tube sheet. The steam inlet connection on the shell side is 4 in. in diameter, and the condensate drain nozzle is 2 in. in diameter. The design, construction, and testing conform to Sec. VIII of the ASME Boiler and Pressure Vessel Code.

The performance of the heater is based on an average steam flow to the shell side of 7700 lb/hr at 600 psig for 3 hr. It is rated for 6,520,000 Btu/hr.

3.2.4 Auxiliary Systems

3.2.4.1 Purification System. In normal operation 9 gpm of water is withdrawn through a 3/4-in. line from one of the reactor forced-circulation inlet nozzles. This steam passes through a regenerative heat exchanger and a cooler to reduce the water temperature to less than 110 F. The water then passes through a prefilter, an ion-exchange bed, and an afterfilter. A pump forces the water back through the regenerative heat exchanger to raise its temperature, and then forces it back to the reactor feedwater line. The purification system is designed to keep the conductivity of the reactor water, under normal operating conditions, at or less than 1 micro mho/cm. The general flow diagram is shown in Dwg. 12E-09010.

Each ion exchanger contains 9 ft³ of resins in a mixed bed 4-1/2 ft deep. The resins consist of 6 ft³ of cation resin and 3 ft³ of anion resin.

Effluent from the ion exchanger has a specific resistance of 12×10^6 ohms, a pH of approximately 7.0, and a silica content of 0.01 ppm or less. About 35-40 lb/yr of corrosion and erosion products are expected. The corresponding loading in the purification system would be 0.73 kilograins/day, or less. Dissolved solids should be chiefly cations from corrosion and erosion of metals in the system.

The purification system includes one regenerative heat exchanger and one cooler, but two ion exchangers, two prefilters, two afterfilters, and two pumps. Normally, one circuit will be operated and the other will be kept on standby. This permits replacing the ion-exchange resins and filter units without shutting down the purification system.

The mixed-bed ion-exchange units are 20 in. in diameter. They are designed for 1250 psig and 130 F and are in a concrete-shielded area near the bottom of the containment building. If the system becomes highly radioactive, an entire ion-exchanger or filter unit can be removed. Normally, however, it should be necessary to remove only the stainless-steel resin basket within one ion-exchanger shell. The top covers of the ion exchangers are removable for this purpose. Each resin basket has a 200-mesh screen permanently attached to it to prevent resin escape.

The upstream filters remove nonionized particles from the water and protect the ion-exchanger resins against major radioactive contamination. The downstream filters remove the minute attrition particles that may escape from the ion-exchanger beds.

All purification-system components (except the filter media and the resins) in contact with the primary reactor water are made of either Type-304 or Type-316 stainless steel. The shell side of the cooler is made of carbon steel.

The regenerative heat exchanger is of the all-welded "horseshoe" type. It is designed for 1250 psig and is made entirely of Type-304 stainless steel. It contains 31 U-tubes 1/4-in. in OD and is approximately 1 ft wide and 4 ft 2 in. long.

Performance of the regenerative heat exchanger is based on a rated duty of 1,275,000 Btu/hr. Water from the reactor centers the unit at 538 F and leaves the tube side at ~270 F. Water is returned to the reactor through the shell side, entering the unit at less than 110 F and leaving the shell at ~438 F.

The cooler is a single-shell U-tube heat exchanger of all-welded construction. The tubes are made of Type-304 stainless steel. Carbon steel is used for the shell, with Type-304 stainless steel cladding on the inside of the header. Tubes are seal-welded to the tube sheet. The unit contains 51 tubes 3/8-in. in OD with 20 BWG walls. The shell is 8-5/8-in. in diameter and 8 ft 8 in. long.

Performance of the cooler is based on a rated duty of 1,450,000 Btu/hr. Water from the tube side of the regenerative heat exchanger enters the tube side of the cooler at ~270 F and is cooled to less than 110 F at the exit. Cooling water at ~80 F is delivered to the shell side, and it leaves at ~110 F.

3.2.4.2 Makeup System. Provisions have been made for supplying a maximum of 25 gpm of demineralized makeup water from the overhead storage tank.

Demineralized water from the overhead storage tank is admitted to the cooling tank by remote manual operation and then injected into the primary system by automatic operation of the makeup injection pumps. Two positive-displacement pumps (one operating, one on standby) are provided for this service and for emergency core cooling. The pumps are motor driven; each has a capacity of 12.5 gpm and is constructed of stainless steel.

3.2.4.3 Control Rod Thimble Cooling System. In order to cool the high-pressure rotary seal on each control-rod drive and to flush out corrosion debris, demineralized water will be delivered to the 13 seals. Some of the water will enter the vessel thimbles and flow into the plenum below the core. The remaining water will pass through the seals and enter the collecting tank for recirculation by the makeup pump. The flow of water up the thimbles keeps the entire rack and thimble at a temperature near 100 F until it enters the vessel.

3.2.4.4 Decay Heat Removal System. The variation of decay heat with time after reactor shutdown was calculated by using the Way-Wigner formula (see Fig. 3.4). Immediately after shutdown, the decay heat is approximately 5.7 percent of operating power. Seven hours after shutdown, the decay heat is approximately 0.75 percent of operating power. Two separate decay-heat removal systems have been provided. One system is used during normal shutdown; the other is used if the core becomes uncovered.

3.2.4.4.1 Decay Heat Removal After Normal Shutdown. During a normal shutdown, the rods will be inserted slowly. Following rod insertion, steam is then bled to the emergency condenser by manually opening the emergency condenser outlet valve. Enough steam is bled to the emergency condenser to remove the decay heat and lower the water temperature at a rate of no more than 2.5 F/min. After approximately 7 hr, the reactor temperature will be about 300 F and further cooling is provided by a decay-heat cooler. The decay heat after 7 hr amounts to approximately 2,000,000 Btu/hr for 58.2-Mw operation. The cooler is designed to remove this heat and to cool the reactor water to 120 F. A reactor-water temperature as low as this is necessary to be able to view the fuel elements underwater without distortion.

The decay-heat cooler and pump are inside the containment building on the basement floor. Reactor water is delivered to the tube side of the cooler, and general service water to the shell side. The heat exchanger is 18 in. in diameter and 20 ft 8 in. long. It contains 218 U-tubes 1/2 in. in OD with 18 BWG walls. The tubes are made of Type-304 stainless steel; the primary channels are made of carbon steel with a stainless-steel cladding. The shell-side components are made of carbon steel. This unit is of all-welded construction and the tubes are welded to the tube sheet.

Performance of the cooler is based on a rated duty of 3,000,000 Btu/hr. This capacity is 50 percent greater than that required to remove the decay heat 7 hr after shutdown. The additional capacity provides for removal of the stored energy present at that time. Service water at ~90 F enters the shell and leaves at ~100 F. Reactor water, under equilibrium conditions, will enter the tubes at 120 F and return to the reactor at 100 F. The shell-side flow rate is 600 gpm; the tube-side flow rate is 300 gpm. The decay-heat cooling pump has a capacity of 300 gpm; it is made of stainless steel.

3.2.4.4.2 Decay Heat Removal After Emergency Shutdown. The emergency cooling system is provided for spraying water over the core to cool the fuel elements upon excessive loss of water from the reactor vessel.

Provisions for high-pressure emergency cooling of the reactor core are incorporated in the makeup system. Should the need arise, as in the case of a reactor vessel low water level, valve N-9 may be opened to provide additional water for injection into the reactor.

In normal operation one makeup pump is on "Hand" operation and the other makeup pump is on "Auto." Should the water level in the reactor decrease to the low water level scram point, the sequence of automatic operations is as follows:

- (1) Valve N-9 will open to admit water to the collecting tank.
- (2) Valve N-11 will close and valve N-10 will open, to direct water to the reactor-vessel spray ring.
- (3) The makeup injection pump on "Auto" will start (the one on "Hand" operation will start as soon as the collecting tank fills up).
- (4) A flow of 25 gpm will enter the reactor vessel through the spray ring, with both pumps running.

Valve N-9 will cycle in a narrow band to ensure that water is available to the makeup pumps without overfilling the collecting tank. A flow switch in the line to the spray ring will actuate an alarm if sufficient flow is not established. Water is available to the collecting tank for at least 10 hr in this mode of operation. The system will revert to the normal lineup when the level in the reactor is restored to clear the low level scram, except that the pump on hand will continue to pump until the collecting tank is empty or the pump is turned off.

In the case of low reactor level accompanied by loss of system pressure (pressure below 25 psig), as might occur in a primary system rupture, pneumatic valve N-3 will also automatically open and admit additional water to the spray ring from the overhead storage tank.

If the cooling is still required after the depletion of all water in the overhead storage tank, pneumatic valve N-4 may be opened remotely by the reactor operator to provide service water to the reactor spray ring for an indefinite period.

3.2.4.5 Shield Cooling System. At 116 Mwt, approximately 100 kw of heat will be generated in the lead thermal shield outside the reactor pressure vessel. To remove this heat, stainless-steel tubing 5/8-in. in OD is embedded in the lead. Duplicate coils are provided in case one should fail. Cooling water is circulated in a closed loop at 35 gpm to cool the lead. The flowsheet for this loop is shown in Dwg. 12E-09011.

The cooling circuit comprises two 35-gpm pumps (one operating, one standby), a shell-and-tube heat exchanger, and an expansion tank. Demineralized water is circulated in the shield cooling circuit and heat is transferred to general service water. An area monitor warns of excessive radioactivity. Filters at the discharge of each pump remove particles from the water and thereby confine radioactivity. Flow indicators, temperature measurement, throttle valves, and alarms are provided for regulating flow at optimum conditions.

The shield cooler is approximately 11 in. in diameter and 10 ft long and contains 88 straight 5/8-in.-OD tubes. It is of fixed-tube-sheet design and is made of carbon steel. Tube sheets are welded to the shell. Channel covers are removable to facilitate cleaning.

Performance is based on a rated duty of 350,000 Btu/hr, with 70 gpm of general service water being delivered to the tube side at 90 F and leaving at 100 F. Cooling water from the coils in the lead enters the shell side at 120 F and leaves at 100 F.

3.2.4.6 Sodium Pentaborate Injection System. A stainless-steel-lined tank of sodium pentaborate, kept at 2000 psig by an air compressor, is the principal component of the sodium pentaborate injection system. The flowsheet for this system is shown in Dwg. 12E-09010. If any control rod fails to hit bottom within 2 sec after a scram signal, an audible alarm is actuated in the control room. The operator may then decide if the sodium pentaborate should be injected into the reactor vessel. An interlock prevents injection of sodium pentaborate under any other conditions. The sodium pentaborate flows into the reactor vessel through a 2-1/2-in. nozzle in the bottom head. The nozzle is directed and baffled so that the vessel will not receive a thermal shock from the sodium pentaborate, which is kept at less than 90 F by electric heaters.

The tank contains 14.3 to 17.3 ft³ of sodium pentaborate under approximately 28 ft³ of air cushion. The solution can be injected in less than 20 sec, and will reduce reactivity at least 14.5 percent under all conditions.

A 2-1/2-in. sodium pentaborate injection line connects the sodium pentaborate injection nozzle to the sodium pentaborate storage tank. This line contains two solenoid-controlled, air-operated valves (called the sodium pentaborate injection valves) controlled by a switch on the reactor control console. A third valve (called the sodium pentaborate injection-system leakoff valve) between the other two permits detection of leakage past the first valve controlling flow from the storage tank. A level switch attached to the storage tank actuates an alarm if the level falls below an acceptable limit. Another alarm is actuated if the sodium pentaborate storage-tank pressure is too low. A relief valve protects the tank against pressures exceeding 2500 psig. Air supply and control power is monitored for failure, and an alarm is sounded in the control room for emergency operation of the sodium pentaborate injection system. The sodium pentaborate injection valves and the sodium pentaborate injection-system leakoff valve would remain operable despite failure of all control-air supply systems external to the sodium pentaborate injection system.

A 2-1/2-in. stainless-steel crossover line leads from the sodium pentaborate injection line to the startup heater system. Flow through this line is controlled by the reactor-vessel bottom-head warmup valve (Valve RA-127). Valve RA-127 is interlocked so that a four-rod scram occurs whenever this valve is open. This scram interlock is necessary because injection of sodium pentaborate might be delayed if the sodium pentaborate injection system were actuated when RA-127 was open.

3.2.4.7 Recombiner System. The gases formed by radiolysis in the reactor water circulating through the core are swept into the evaporators by the steam. They collect in the lower channel section of the evaporators and are carried off with a small flow of steam in the 3/4-in. vent line to the recombinder system. Approximately 3 scfm of H₂ and O₂

pass through the recombination chamber and are reunited to form steam. The steam then flows through a desuperheater to a heat exchanger, is condensed, and is drained to the collecting tank for reinjection into the primary system. The noncondensable gases are vented from the heat exchanger and are monitored for fission products. The gases then go to a holdup tank, where they are retained for about 1/2 hr. After the gases leave the holdup tank, they pass through a scrubber and absolute filter, where any particulates are removed. The gases are then diluted by 400 scfm of dilution air before being blown into the containment exhaust duct between the roughing and absolute filters.

If the fission-products radiation monitor indicates fission products in the noncondensable gases, the gases can be routed from the holdup tank through two carbon adsorbers, set up individually, in parallel, or in series. If the gases are hot thermally they are routed through a refrigerated cooler before passing into the carbon adsorbers.

3.2.4.8 Fuel-Assembly Storage-Well Cooling System. The storage-well cooler is a conventional shell-and-straight-tube heat exchanger with fixed tube sheets. It is 12-3/4 in. in diameter and 19-1/2 ft long. It contains 154 tubes 1/2-in. in OD with 20 BWG walls and is rated for 1,000,000 Btu/hr. Storage-well water enters the tubes at 120 F and leaves at 100 F. Service water enters the shell at 90 F and leaves at 98 F. The tubes, tube sheets, and water boxes of the cooler are made of Type-304 stainless steel; the shell is made of carbon steel. The cooler was designed, constructed, and tested in accordance with ASME Boiler and Pressure Vessel Code.

The fuel-assembly storage-well cooling-system pump is mechanically sealed, is constructed of stainless steel, and has a capacity of 100 gpm.

3.2.4.9 Reactor Flooding System. The reactor flooding system fills the cavity above the reactor vessel and the fuel-element storage well with water from the overhead storage tank before the reactor is refueled. The water flows by gravity through the feedwater inlet nozzles until it fills the pressure vessel, shield cavity, and storage well. After refueling, the water is pumped through one of the forced-circulation nozzles back to the overhead storage tank. The purification system may be used to purify the shield-cavity water before it is pumped back to the overhead storage tank, and the portable demineralizer may be used to purify the storage-well water before it is pumped back to the overhead storage tank.

3.2.5 Instrumentation and Controls

3.2.5.1 Rod Control System. One of the 13 control rods operates as a regulating rod. A separate switch controls this rod. An additional set of switches controls the remaining 12 rods in conjunction with a rod selector switch that prevents the simultaneous withdrawal of more than one rod. It is possible, however, to scram or to insert all rods simultaneously.

The 12 control rods and the regulating rod are manually positioned until the desired power level is reached. The regulating rod is then automatically positioned by a servo system to control primary-system pressure. Visible and audible signals indicate control range limits of the regulating rod.

Each drive has an electromagnetic clutch that couples the gearhead drive motor to the drive mechanism. The gearhead drive motor incorporates a self-locking worm gear that holds the drive mechanism in a fixed position when the motor is de-energized and the electromagnetic clutch is energized. When the clutch is de-energized, the rod falls by gravity to its full-in position. Total scram time is less than 2 sec. This time includes the delay of approximately 0.05 sec. from the time of the scram signal until the start of free fall and the approximately 1.15 sec. of fall until the rod is within 0.5 in. of bottoming.

A timer measures scram time from any initial position to within 0.5 in. of the full-in position for any rod selected. Each rod is timed during a reactor interlock startup check to ensure that it is not jammed. The interlocks permit withdrawal of only the selected rod and require the rod under test to be returned to its full-in position before another rod can be withdrawn for test. Separate test circuitry permits complete testing of the regulating rod if all other rods are in their full-in positions.

Scram circuitry provides for two types of scram: four-rod scrams and all-rod scrams. Four-rod scrams place the reactor on "hot standby." After a four-rod scram, power operation can be resumed by withdrawing the four scrambled rods. Interlocks operated by each rod within 0.5 in. of its full-in position prevent regaining rod-drive control power after an all-rod scram, unless all rods are within 1/2 in. of their full-in positions.

3.2.5.2. Rod Position Measurement, Indication, and Limit Switching. Rod positions are measured by synchro transmitters geared to the drive mechanisms. Each rod drive mechanism has a limit switch that turns on a light on the control console whenever the rod is within 1/2 in. of its full-in position. The same switch is also used for rod-drop timing and for the startup interlock sequence, which prevents rods being withdrawn after an all-rod scram unless all rods are within 1/2 in. of their full-in positions. Additional limit switches stop the drive motors at the "in" and "out" limits of travel and turn on indicating lights when the drives are at their "out" limits. Another light is turned on by the rod selector switch to indicate which rod drive can be operated. Warning lights throughout the plant area indicate when rod-drive control power is on, providing plant-wide warning that the reactor is operating or operable.

3.2.5.3. Safety and Alarm Circuitry. The safety and alarm circuitry actuates all-rod scrams, four-rod scrams, and control-room alarms in response to various abnormal conditions.

The causes of all-rod scrams are as follows:

- (1) high flux (greater than 125 percent of full-power flux) measured by either one or two of the three level safety channels in accordance with the setting of the scram selector switch (see Sec. 3.2.5.4),
- (2) high flux (greater than 125 percent of selected range) measured by either or both of the two linear channels, in accordance with the setting of the scram selector switch,
- (3) short period measured by either log-n channel (during startup only),
- (4) failure of the control-rod-drive mechanism magnetic-clutch power supply, and
- (5) pressing of the "all-rod scram" button on the control panel or the "all-rod scram" button in the containment building.

The causes of four-rod scrams are as follows:

- (1) high water level in the reactor vessel,
- (2) low water level in the reactor vessel,
- (3) low reactor water temperature,
- (4) high primary-system pressure,
- (5) opening of the reactor-vessel bottom-head warmup valve (Valve RA-127),
- (6) high secondary-system pressure,
- (7) opening of the emergency-and-test-condenser condensate flow-control valve,
- (8) tripping of the superheater auxiliaries,
- (9) closure of the turbine stop valve, and
- (10) pressing the four-rod scram button on the control panel.

The causes of control-room alarms are as follows:

- (1) boric acid tank - low level,
- (2) boric acid tank - low pressure,
- (3) shield cooling system - high temperature,
- (4) building area monitors - high radiation,
- (5) secondary water monitor - high radiation,
- (6) stack particulate monitor - high radiation,
- (7) stack-monitor trouble,
- (8) stack-gas monitor - high radiation,
- (9) fission products monitor - high radiation,
- (10) sewer monitor - high radiation,
- (11) recombiner - high temperature,
- (12) Evaporator 1 and 2 - downcomer high level,
- (13) Evaporator 1 and 2 - downcomer low level,

- (14) reactor spray valve - open,
- (15) reactor spray system - off automatic,
- (16) building airlocks - not secured,
- (17) mobile monitor - high radiation,
- (18) shield-cooling surge tank - low level,
- (19) boric acid tank - low temperature,
- (20) shield cooling system - low flow,
- (21) instrument cabinet blower - off high temperature,
- (22) purification system - high conductivity,
- (23) purification system inlet - high temperature,
- (24) boric acid system - power off,
- (25) decay heat cooler outlet - high temperature,
- (26) reactor building - high pressure,
- (27) uninitiated return to manual control,
- (28) evaporator blowdown line - high conductivity,
- (29) decontamination room retention tank - high level,
- (30) Retention Tank 1 - high level,
- (31) Retention Tank 2 - high level,
- (32) demineralized water storage tank - high level,
- (33) demineralized water storage tank - low level,
- (34) control air - low pressure,
- (35) emergency condenser - low level,
- (36) emergency condenser - high level,
- (37) evaporator - high level,
- (38) evaporator - low level,
- (39) superheater outlet steam - high temperature,
- (40) superheater interstage steam - low temperature,
- (41) superheater auxiliaries trip,
- (42) superheater fuel-system trip,
- (43) reactor auxiliaries trip,
- (44) main feed breaker trip,
- (45) reserve feed breaker trip,
- (46) superheater building main feed breaker trip,
- (47) superheater building reserve feed breaker trip,
- (48) reactor building main feed breaker trip,
- (49) reactor building reserve feed breaker trip,
- (50) Motor Control Center 1 feed breaker trip,
- (51) Motor Control Center 2 feed breaker trip,
- (52) Motor Control Center 3 feed breaker trip,
- (53) Motor Control Center 3 transfer switch in emergency position,
- (54) switchgear d-c power off,
- (55) engine generator oil low pressure - high temperature,
- (56) offgas system high pressure,

- (57) engine generator start failure,
- (58) 60/400 cycle inverter trip,
- (59) battery charger trip/Lo battery voltage,
- (60) superheater pressure positive,
- (61) flame failure,
- (62) reactor vessel - high level,
- (63) reactor vessel - low level,
- (64) reactor spray system - low flow,
- (65) collecting tank - high level-low level,
- (66) primary feedwater high temperature,
- (67) secondary power relief valve - open,
- (68) boiler feedwater pump trip,
- (69) turbine air ejector monitor - high radiation,
- (70) turbine air ejector - monitor trouble,
- (71) building evacuation alarm - actuated,
- (72) reactor building pressure high - spray warning,
- (73) reactor cavity monitor - high temperature/high level,
- (74) Startup Channel N-1 - short period,
- (75) Startup Channel N-2 - short period,
- (76) Log-n Channel N-3 - short period,
- (77) Log-n Channel N-4 - short period,
- (78) Linear Channel N-5 - out of range,
- (79) Linear Channel N-6 - out of range,
- (80) Safety Channel N-7 - high flux,
- (81) Safety Channel N-8 - high flux,
- (82) Safety Channel N-9 - high flux,
- (83) Safety Channel N-7 - low flux,
- (84) Safety Channel N-8 - low flux,
- (85) Safety Channel N-9 - low flux,
- (86) coal feeder shear pin failure,
- (87) regulating rod out of range,
- (88) a-c/d-c trouble on MG set,
- (89) N-5 high-flux scram,

Scram alarms:

- (90) N-6 high-flux scram,
- (91) four-rod scram,
- (92) primary steam high pressure,
- (93) primary water low temperature or RA-127 open,
- (94) reactor vessel high level,
- (95) reactor vessel low level,
- (96) turbine trip,
- (97) emergency condenser flow control valve open,

- (98) secondary steam high pressure,
- (99) superheater trip,
- (100) all-rod scram,
- (101) Log-n Channel N-3 - short period,
- (102) Log-n Channel N-4 - short period,
- (103) Safety Channel N-7 - high flux,
- (104) Safety Channel N-8 - high flux, and
- (105) Safety Channel N-9 - high flux.

The safety circuitry also enforces proper startup and operating procedures. It does so by requiring that certain conditions be met before startup can begin, before control rods can be withdrawn, and before the reactor can be placed on automatic control.

The conditions that must be met before a "reactor start" permit can be obtained are as follows:

- (1) the reactor "power on" key switch must be turned to its "on" position,
- (2) the Channel N-1 and N-2 high-voltage switches must be in their "on" positions,
- (3) the Channel N-3 and N-4 period-scram bypass switches must be in their "in circuit" positions,
- (4) all 13 control rods must be fully inserted,
- (5) the containment-building airlocks must be secured, and
- (6) all all-rod scram circuits must be clear.

Whenever the six foregoing conditions have been met and the "reactor start" button has been pressed, it is unnecessary thereafter to meet the second, third, fourth, and fifth conditions, and the "reactor start" permit is automatically revoked thereafter only in response to any all-rod scram signal or whenever the reactor "power on" key switch is turned to its "off" position.

The conditions that must be met before a "rod withdrawal" permit can be obtained are as follows:

- (1) the reactor period sensed by Channels N-1 and N-2 must be greater than 7 sec, or the reactor-period interlock associated with either Channel N-1 or N-2 must be bypassed.

(2) the reactor period sensed by Channels N-3 and N-4 must be greater than 15 sec, or the reactor-period interlock associated with either Channel N-3 or N-4 must be bypassed,

(3) the regulating-rod and shim-rod test switches must be in their "off" positions,

(4) all four-rod scram circuits must be clear, and

(5) a "reactor start" permit must be in effect.

Any "rod withdrawal" permit is automatically revoked whenever any of the five foregoing conditions no longer exists.

The conditions that must be met before an "automatic control" permit can be obtained are as follows:

(1) the regulating rod must be within its automatic-control travel range,

(2) the difference between the actual primary-system pressure and the pressure at which the pressure set-point potentiometer is set must be less than or equal to 50 psi,

(3) the regulating-rod control switch must be in its "off" position,

(4) a "rod withdrawal" permit must be in effect, and

(5) the mode-of-operation switch must be in its "automatic control" position.

Any "automatic control" permit is automatically revoked whenever any of the five foregoing conditions no longer exists.

3.2.5.4 Power and Period Measurement. The nuclear instrumentation system (Dwg. 12E-11926) indicates neutron flux at all power levels and prohibits rod withdrawal or shuts the reactor down at high rates of power increase and when the neutron flux exceeds safe levels.

A graph on Dwg. 12E-11926 presents an over-all picture of the coverage and overlapping of the nine nuclear instrument channels against a common flux scale. The scale is adjusted to account for flux attenuation between the reactor core and the detector position.

Two boron trifluoride proportional-counter channels (Channels N-1 and N-2), indicating flux to five decades from source level, are used during reactor startup. These channels are equipped with scalers and logarithmic count-rate meters. The instruments prohibit rod withdrawal when the reactor period is less than 7 sec.

Two log-n and period channels (Channels N-3 and N-4) indicate neutron flux and period in the intermediate and full-power ranges. These channels contain logarithmic amplifiers fed by compensated ionization chambers, giving a coverage of six decades of flux. They also prohibit rod withdrawal when the reactor period is less than 15 sec and shut down the reactor when the period is less than 3 sec. The period interlocks may be bypassed to avoid false scrams attributable to boiling turbulence rather than to short period.

Two linear power channels (Channels N-5 and N-6) indicate power level between eight decades below full power and 150 percent of full power. These channels use an electrometer fed by a compensated ion chamber. Adjustable level trip signals shall actuate a control-room alarm whenever they sense a reactor power level less than 5 percent or greater than 110 percent of the power-range setting and shall actuate a control-room alarm and initiate an all-rod scram signal whenever they sense a reactor power level exceeding 125 percent of the power-range setting. The trip circuits are arranged so that the operator, by turning a switch on the control console, can impose the requirement that both channels must generate a scram signal before a scram occurs. The two-out-of-two requirement reduces the likelihood of false scrams, without compromising safety. Because of their low range, these channels, as well as Channels N-1 and N-2, will be used during startup.

Three high-flux level safety channels (Channels N-7, N-8, and N-9), using uncompensated ion chambers, scram the reactor when the flux rises excessively. Each channel actuates an alarm whenever the flux exceeds 110 percent of the full-power flux and generates a scram signal whenever the flux exceeds 125 percent of the full-power flux. The trip circuits are arranged so that the operator, by turning a switch on the control console, can impose the requirement that any two, rather than any one, of the three safety channels must generate a scram signal before a scram occurs. The two-out-of-three requirement reduces the likelihood of false scrams without compromising safety. The current amplifiers in the high-flux safety channels have range-changing switches and alarm points set at 10 percent of full power to indicate instrument or detector failure.

The flux detectors for the nine nuclear instrumentation channels are in six horizontal tubes, four tangential and two radial (see Dwg. 12E-01502).

The four tangential holes are in parallel pairs on opposite sides of the reactor vessel. They extend clear through the biological shield and are tangent to the outer surface of the lead thermal shield. The axes of two of the tangential holes are approximately 12 in. above the horizontal midplane of the core; the axes of the other two are approximately 12 in. below the horizontal midplane of the core.

The axes of the two radial holes are 20 deg apart and intersect the vertical centerline of the reactor 6 in. above the horizontal midplane of the core. The holes themselves start from the eastern and southeastern faces of the biological shield and end at the outer surface of the lead thermal shield on the side of the core opposite the neutron source.

The arrangement of detectors in the instrument holes is as follows (see Fig. 3.23):

Channel N-1	radial hole on southeast face of biological shield
Channel N-2	radial hole on east face of biological shield
Channel N-3	center of upper south tangential hole
Channel N-4	center of upper north tangential hole
Channel N-5	west half of lower north tangential hole
Channel N-6	center of upper south tangential hole
Channel N-7	east half of lower south tangential hole
Channel N-8	west half of lower south tangential hole
Channel N-9	east half of lower north tangential hole

An additional detector may be installed in the west half of upper north tangential hole. The sensitivity of any detector is adjusted by moving it along the axis of the instrument hole it occupies.

The three level-safety-channel detectors and the two linear-power-channel detectors are positioned to cover each quadrant of the core, as shown in Fig. 3.5. If one level-safety-channel detector fails, the two linear-power channels and the two remaining level-safety channels will continue to monitor at least three quadrants of the core, and the failure will actuate a control-room alarm.

3.2.5.5 Temperature Interlocks. The reactor water must be heated to 425 F before the control rods can be withdrawn. The inherent ability of the reactor to offset reactivity increases by steam void formation is improved by preheating the water, because the void coefficient of reactivity is greater at higher temperatures. The time lag between power increase and void formation is also reduced. If the reactor water temperature falls below 425 F, an interlock prevents startup or, if the reactor is operating, causes a four-rod scram.

3.2.5.6 Pressure Interlocks. When the primary steam pressure exceeds 975 psig, a four-rod scram occurs. If this interlock fails to produce a scram, and if the pressure continues to rise until it reaches 1210 psig, the pressure relief valve to the emergency condenser will open and the containment-building ventilation-duct dampers will close. A set of interlock contacts on the relief valve causes a four-rod scram to shut down the reactor when the valve opens.

3.2.5.7 Reactor Water Level Measurement and Control. A level of water in the reactor is measured by a differential pressure cell and recorded. A sight glass backs up the differential pressure cell. Safety interlocks actuated by a differential pressure cell shut down the reactor if the water level is too high or too low. Additional alarm contacts from the cell warn of impending trouble before the reactor is shut down.

If the water level in the reactor should drop below the top of the shroud, an alarm is sounded in the control room. In response to this alarm, an attempt can be made to raise the water level by the injection of 25 gpm of makeup water. If, upon the addition of this 25 gpm of makeup water, the water level does not rise, the flow can be diverted so that it enters the vessel through the spray ring rather than through the feedwater inlet nozzles. This flow cools the core directly if the water level falls below the top of the core while the vessel pressure is still high. If the level continues to fall, a four-rod scram will occur and the emergency spray system (see Sec. 3.2.4.4.2) will be actuated automatically.

3.2.5.8 Temperature Measurement. Temperatures at various points in all auxiliary systems closely allied to reactor operation, such as the purification system and the shield cooling system, are measured and recorded in the control room.

3.2.5.9 Automatic Control. The control rods are manually positioned until the operating power level is reached. The regulating rod is then switched to automatic control, and the other 12 rods are adjusted so that the regulating rod is near the farthest-in position of its range. As the regulating rod is automatically withdrawn after startup, the 12 rods are manually positioned to keep the regulating rod within the proper operating range.

The automatic control system integrates the reactor and the turbine generator plant. It is a dual system, consisting of two interconnected loops. One control loop maintains primary steam pressure at a predetermined value for a given load, through the adjustment of the regulating rod. The second loop holds a uniform steam pressure at the turbine inlet valve by varying the governor-valve position.

3.2.6 Heat Transfer

3.2.6.1 Natural Circulation Cooling of the Core. The heat generated in the fuel is removed by natural circulation of the boiling water within the core. Recirculated water is returned to the bottom of the core by way of the annular downcomer surrounding the fuel assemblies. Natural circulation arises from the buoyancy of the steam bubbles in the core and in the shroud above the core. Flow and steam-volume distributions in the core and shroud were calculated by equating the buoyant driving force of the steam to the sum of the head losses occurring as the steam-water mixture flows up through the heated channels and as the water returns through the downcomer to the fuel-channel entrances. Calculations of the effective density of boiling steam-water mixtures at varying operating conditions and of head losses for the flow of such mixtures are based on experimental results.

The experiments and calculation procedures of Lottes and Flinn* were used to calculate steam-water densities in the core. The calculation method was modified so that the results would apply to ERR conditions rather than to those of the experiments. These conditions are:

*Lottes, P.A., and W.S. Flinn, Nuclear Science and Engineering: Vol. 1, pp.461-476(1956)

	ERR	Lottes & Flinn experiments
flow-channel cross section	area between 0.452- in.-dia. tubes on 0.75-in. square pitch	1/2 in. x 2 in. rectangle
heated height	60 in.	60 in.
shroud height	63 in.	0 in.
operating pressure	922 psig	600 psig
maximum power density in flow channel .	96.4 kw/liter	50 kw/liter
range of inlet water velocity	3.6 to 4.4 ft/sec	1.2 to 3.0 ft/sec
vertical distribution of heat generation . .	roughly cosine-shaped, but with maximum flux below center of core	constant

Some extrapolations of the experimental results are required before they can be supplied to the flow calculation.

The average steam-water density in the downcomer was determined from air-water experiments on a full-scale model of the ERR shroud, flow-baffles and downcomer (ACNP-63539, "Elk River Reactor Thermal and Hydraulic Performance," April 11, 1963). The air-water data was related to the steam-water conditions of ERR using the correlating parameters developed by Petrick (M. Petrick, "A Study of Vapor Carryunder and Associated Problems," ANL-6581, July 1962). Petrick correlated downflow air-water loop data with downflow steam-water loop data for steam pressures of 600, 1,000, and 1,500 psia and with EBWR downcomer data for operating pressures of 300 and 600 psia.

The steam void fraction in a boiling-water reactor is the ratio of steam volume to the steam-water mixture volume. The average steam void fraction in the core and the void fraction distribution depend on the distribution of heat generation in the core. The distribution of heat depends, primarily, on control-rod bank position and, to a lesser extent, on steam void distribution. The maximum heat flux in the core is limited to 313,000 Btu/(hr)(ft²) at full power (58.2 Mw) in order to obtain a large margin between burnout heat flux and peak operating heat flux. For full power operation, the peak heat flux is equal to or less than 313,000 Btu/(hr)(ft²) if the control-rod bank is withdrawn 50 in. or more from the fully inserted position.

The calculated thermal and hydraulic conditions for the initial core operating at 58.2 Mw and 920 psig with control rods withdrawn 50 in. are tabulated below. The initial core consists of 128 regular elements and 20 spiked elements.

Primary feedwater temperature, °F 450 flux peaking factors:

radial	1.60
axial	1.80
local	1.20
overpower	1.25
product	4.32

Heat flux, Btu/(hr)(ft²):

average	90,700
maximum at 100 percent power	313,000
maximum at 125 percent power	391,000

Recirculation flow rate, lb/hr: 6,050,000

Steam quality, weight fraction:

maximum at core exit	0.062
average in shroud	0.046
carryunder to downcomer	0.003

Steam void fraction:

maximum at core exit	0.48
average in core	0.20
average in shroud	0.35

Core inlet subcooling, °F: 1.9

Inlet velocity in hottest fuel element, ft/sec: 4.3

3.2.6.2 Burnout Considerations. Burnout occurs when vapor blankets the heated surface, which can occur either at high heat fluxes or when insufficient liquid is available to carry away the heat. "Dry pipe" burnout occurs only when exit qualities exceed roughly 70 percent.

The preceding table shows that the anticipated average exit quality in the reactor is only 2.8 percent. The highest exit quality at this power is 6.5 percent. There is, consequently, no danger of "dry pipe" burnout in this reactor.

Burnout at high heat fluxes is estimated to occur at a heat flux of $1,000,000 \text{ Btu}/(\text{hr})(\text{ft}^2)^{**}$ for conditions at the center of the core. In comparison, the maximum expected heat flux in the core is $313,000 \text{ Btu}/(\text{hr})(\text{ft}^2)$ at 58.2 Mw. The heat fluxes are thus well below the burnout point at all locations in the core.

3.2.6.3 Fuel-Assembly Thermal Performance. The temperatures within the fuel tubes were determined for operation at 58.2 Mw. The radial variation in power generation within the thorium-uranium fuel pellets was taken into account, although the thermal flux depression is small. Calculations were performed for tubes in the region of maximum neutron flux and in the region of average neutron flux. Thermal-conductivity values for the thorium-uranium fuel were based on the data of Kingery, et al.* In all cases, the highest temperature of the thorium-uranium pellets were below the melting point (5900 F).

A fuel tube with nominal clearances in a region of average neutron flux operates under these conditions:

heat flux	90,700 Btu/(hr)(ft ²)
boiling fluid temperature	536 F
fuel-tube outside surface temperature	549 F
fuel-tube inside surface temperature	564 F
surface temperature of thorium-uranium pellets	664 F
center temperature of thorium-uranium pellets	945 F

A fuel tube with nominal clearances in the region of maximum local neutron flux operates under these conditions:

heat flux	313,000 Btu/(hr)(ft ²)
boiling fluid temperature	536 F
fuel-tube outside surface temperature	561 F
fuel-tube inside surface temperature	611 F
surface temperature of thorium-uranium pellets	956 F
center temperature of thorium-uranium pellets	2819 F

3.2.7 Physics

3.2.7.1 Introduction. The important results of the physics study are summarized in Table 3-2 and discussed in the following subsections. The analytical methods are explained in Allis-Chalmers' Final Reactor Physics Report for the Elk River Reactor (May 23, 1960).

* Kingery, W. D., et al., J. Am. Ceramic Society, Vol. 37, No. 2 (Part II), p. 107-110 (1954).

** Basis for burnout heat flux estimate is given in Appendix A.

TABLE 3-2

PHYSICS DATA

initial enrichment, $\frac{\text{U-235 as metal}}{\text{Total Th + U as metal}}$	4.3%
enrichment of spiked elements, $\frac{\text{U-235 as metal}}{\text{total Th + U as metal}}$	5.2%
boron content of fuel-tube metal (by weight)	600 ppm
initial conversion ratio	0.396
average fuel lifetime	9500 Mwd/MT
total amount of fuel (Th + U as metal)	3970 kg
initial k_{eff} (cold, clean core)	1.1265
initial k_{eff} (hot, clean core)	1.0831
initial k_{eff} (hot core; Xe + Sm)	1.0563
reactivity worth of 148 boron pins:	
hot core	0.091
cold core	0.058
minimum reactivity decrease caused by injection of boric acid ..	0.145
initial k_{eff} :	
hot; Xe + Sm; 60 boron pins inserted	1.0179
cold, clean; 60 boron pins inserted	1.1001
total reactivity worth of control rods:	
hot core	0.284
cold core	0.183
reactivity worth of rods with center rod out (cold core)	0.149
reactivity worth of rods partially inserted into operating core:	
inserted 1/6 of core height	0.0055
inserted 1/3 of core height	0.0184
reactivity changes caused by:	
temperature change from 20 C to 281 C (zero power)	0.0128
Doppler broadening owing to power change from 0 to 58.2 Mw (0% voids)	0.0030
average void change from 0 to 17.6%:	
clean core	0.039
equilibrium Xe	0.023
equilibrium Sm	0.007
reactivity coefficients:	
unrodded temperature coefficient (moderator + fuel):	
at operating temperature	$-1.5 \times 10^{-4} / ^\circ\text{C}$
at room temperature	$+6.0 \times 10^{-5} / ^\circ\text{C}$

unrodded void coefficient:
 at 0% voids $-1.23 \times 10^{-3}/\%$ void
 at 17.6% voids $-1.90 \times 10^{-3}/\%$ void
 minimum number of fuel assemblies for criticality with the
 plutonium-beryllium source replacing one fuel assembly
 in the central position 30
 average neutron lifetime 3.53×10^{-5} sec

	core condition	
	operating, hot, clean, no boron pins	cold, clean, no boron pins
regeneration factor	1.62	1.64
fast-fission factor	1.0	1.0
thermal utilization factor	0.79	0.76
resonance escape probability ...	0.92	0.95
diffusion area	4.01 cm^2	1.57 cm^2
Fermi age	73.1 cm^2	37.1 cm^2
buckling	0.00116 cm^{-2}	0.00125 cm^{-2}
effective multiplication factor ..	1.081	1.126
<u>average flux in moderator</u>	1.280	1.443
<u>average flux in fuel</u>		
<u>average flux in fuel cladding</u> ...	1.154	1.196
<u>average flux in fuel</u>		

3.2.7.2 Initial Reactivity. The initial reactivities are listed in Table 3-2. The calculated k_{eff} of the operating core may be conservatively low for two reasons:

- (1) the resonance integral used for zirconium (3 barns) may be too great; and
- (2) calculations based on average void fractions yield a lower k_{eff} than calculations based on distributed voids.

3.2.7.3 Reactivity. Reactivity changes in the initial core because of voids, temperature, and xenon and samarium buildup have been calculated, and are reported in Table 3-2. Figures 3.6 and 3.7 show k_{eff} as a function of average void in the core (corrected for fuel temperature coefficient) and k_{eff} as a function of moderator temperature.

3.2.7.4 Boron Pins. The center pin of each fuel assembly is interchangeable with a borated stainless-steel pin. This permits the initial reactivity to be adjusted with no power distortion from deeply inserted control rods. The calculated reactivity change caused by substituting a borated stainless-steel pin for the center fuel pin of every fuel assembly in the core is 0.091 for the operating core, and 0.058 for the cold core.

3.2.7.5 Control-Rod Worth. The reactivity worth of the control rods was calculated by two- and four-group theory. The two-group calculations were done with the two-dimensional PDQ code on an IBM-704 computer. This method entailed calculations based on the assumptions:

- (1) that all the rods were fully inserted, and
- (2) that the center rod was fully withdrawn and all other rods were fully inserted.

The epithermal component of the rod worth was then estimated with four-group calculations on the Bendix G-15-D Computer.

To calculate the power distribution effects of controlling the operating core by banked movement of the control rods, the rods were assumed to be inserted $1/6$ and $1/3$ of the core height, and the VALPROD code was used on an IBM-650 computer. Figure 3.8 shows the thermal flux with the rods fully withdrawn, $1/6$ inserted, and $1/3$ inserted; Fig. 3.9 shows the estimated total rod worth as a function of insertion distance in the hot operating core.

3.2.7.6 Fuel Lifetime. Core lifetime was calculated with a code for the Bendix G-15-D Computer. Figure 3.10 shows k_{eff} as a function of fuel irradiation, on the basis of a uniform flux distribution equal to the average core flux. However, this is an ideal condition, and the expected lifetime of the core is approximately 75 percent of the calculated value, or, as shown in Fig. 3.10, about 9500 Mwd/MT. The results of related calculations are shown in Figs. 3.11 through 3.15.

Depending on initial reactivity, it may be necessary for the fuel bundles to contain boron pins during initial operation. (The number of bundles actually containing these pins will depend on the excess reactivity during operation.) For example, if 40 percent of the fuel assemblies contain boron pins, the reactor will operate about 4750 Mwd/MT (320 days) before the pins must be removed. After the pins are replaced by fuel tubes, the reactor will operate another 4750 Mwd/MT (320 days) before the available reactivity is insufficient for further operation.

3.2.7.7 Spiked Fuel Elements. The 148 normal fuel assemblies contain 4.3 percent of enriched uranium; however, 22 bundles containing 5.2 percent of enriched uranium were also fabricated. These "spiked" elements provide sufficient startup versatility to offset differences between the calculated and the actual core characteristics. If the spiked elements are not needed at startup, they will be used at the end of the normal core life and will permit a longer core lifetime than that predicted on the basis of normally enriched elements.

3.2.7.8 Reactor Shielding. The shielding calculations followed the method developed by Grotenhuis and Butler of the Argonne National Laboratory. The design basis was that the biological shield must be thick enough to reduce the dose rate at the outer surface of the shield to 2.5 mr/hr when the reactor is operating at 116.4 Mw. To reduce the dose to this level, the radial shield must consist of 9.5 ft* of ordinary concrete. Four and five-tenths feet of heavy concrete above the core, and 4 ft of heavy concrete below the core are sufficient to reduce the dose rate to below the tolerance level. Heat production in all regions surrounding the reactor core is low enough to permit reactor operation without shield-cooling problems.

Table 3-3 lists the materials and thicknesses of the radial and axial shielding of the reactor core.

* The thickness actually provided is 8.5 ft, since it is believed that the calculations are somewhat conservative. Space has been provided for the addition of 1 ft more of heavy concrete if it is needed.

TABLE 3-3
SHIELDING THICKNESSES

<u>direction</u>	<u>region</u>	<u>material</u>	<u>thickness, cm</u>
radial (proceeding outward from core)	1	water (reflector)	25.40
	2	steel (thermal shield)	2.54
	3	water	2.54
	4	steel (reactor vessel)	8.57
	5	insulation	7.62
	6	air gap	7.62
	7	steel	1.91
	8	lead (95%) and water (5)	7.62
	9	steel	1.91
	10	ferroboron plaster	1.27
	11	ordinary concrete	243.84
above the core (proceeding upward)	1	water	91.44
	2	steam	261.62
	3	steel (reactor vessel)	12.70
	4	insulation	12.70
	5	void	15.24
	6	barytes concrete	152.40
below the core (proceeding downward)	1	water and steel	4.00
	2	water and steel	5.00
	3	water	33.00
	4	steel (reactor vessel)	3.38
	5	insulation	9.00
	6	steel	0.75
	7	lead (95%) and water (5%)	3.00
	8	steel	0.75
	9	heavy concrete	48.00
	10	steel	4.00

Table 3-4 lists the calculated fast and thermal neutron fluxes and heating values in each shielding region.

TABLE 3-4

NEUTRON FLUXES AND HEATING VALUES IN SHIELDING REGIONS

<u>direction</u>	<u>region</u>	<u>face</u>	<u>fast flux, n/cm²-sec*</u>	<u>thermal flux, n/cm²-sec*</u>	<u>heating watts/cm³*</u>
radial (proceeding outward from core)	1	Inside	2.00 (13)	2.00 (13)	0.70 (0)
		Outside	7.61 (11)	5.77 (12)	0.23 (0)
	2	Inside	7.61 (11)	5.77 (12)	1.80 (0)
		Outside	4.26 (11)	2.19 (12)	1.07 (0)
	3	Inside	4.26 (11)	2.19 (12)	0.13 (0)
		Outside	3.21 (11)	1.38 (12)	0.12 (0)
	4	Inside	3.21 (11)	1.38 (12)	0.97 (0)
		Outside	5.23 (10)	6.20 (10)	0.13 (0)
	5	Inside	5.23 (10)	6.20 (10)	0.13 (0)
		Outside	4.89 (10)	6.20 (10)	0.13 (0)
	6	Inside	4.89 (10)	6.20 (10)	0.13 (0)
		Outside	4.60 (10)	6.20 (10)	0.13 (0)
	7	Inside	4.60 (10)	6.19 (10)	0.13 (0)
		Outside	3.13 (10)	6.10 (10)	0.09 (0)
	8	Inside	3.13 (10)	6.10 (10)	2.10 (-1)
		Outside	1.81 (10)	5.02 (10)	8.50 (-3)
	9	Inside	1.81 (10)	5.02 (10)	1.75 (-2)
		Outside	1.24 (10)	0.00 --	1.25 (-2)
above the core (pro- ceeding upward)	10	Inside	1.24 (10)	0.00 --	1.25 (-2)
		Outside	1.24 (10)	0.00 --	1.25 (-2)
	11	Inside	1.24 (10)	0.00 --	3.21 (-3)
		Outside	2.70 (0)	0.00 --	1.71 (-9)
	1	Inside	2.00 (13)	2.00 (13)	0.62 (0)
		Outside	9.45 (8)	5.83 (9)	9.81 (-3)
	2	Inside	9.45 (8)	5.83 (9)	1.96 (-4)
		Outside	3.64 (8)	9.98 (8)	6.34 (-5)
	3	Inside	3.64 (8)	9.98 (8)	2.71 (-2)
		Outside	3.62 (7)	3.44 (7)	1.41 (-3)

*Figure in parentheses denotes the power of 10 by which the column entry is multiplied.

Table 3-4 -- Neutron Fluxes and Heating Values in Shielding Regions (cont'd)

<u>direction</u>	<u>region</u>	<u>face</u>	<u>fast flux, n/cm²-sec*</u>	<u>thermal flux, n/cm²-sec*</u>	<u>heating, watts/cm³*</u>
above the core (pro- ceeding upward)	4	Inside	3.62 (7)	3.44 (7)	1.41 (-3)
		Outside	3.53 (7)	3.44 (7)	1.41 (-3)
	5	Inside	3.53 (7)	3.44 (7)	1.41 (-3)
		Outside	3.41 (7)	3.44 (7)	1.41 (-3)
	6	Inside	3.41 (7)	3.44 (7)	6.67 (-4)
		Outside	8.30 (0)	6.66 (1)	6.57 (-11)
below the core (pro- ceeding downward)	1	Inside	2.00 (13)	2.00 (13)	1.95 (0)
		Outside	4.06 (12)	1.16 (13)	1.16 (0)
	2	Inside	4.06 (12)	1.16 (13)	1.98 (0)
		Outside	2.74 (11)	9.95 (10)	0.32 (0)
	3	Inside	2.74 (11)	9.95 (10)	6.75 (-2)
		Outside	7.89 (7)	4.25 (8)	3.83 (-3)
	4	Inside	7.89 (7)	4.25 (8)	3.68 (-2)
		Outside	1.68 (7)	1.72 (7)	5.36 (-3)
	5	Inside	1.68 (7)	1.72 (7)	5.36 (-3)
		Outside	1.50 (7)	1.72 (7)	5.36 (-3)
	6	Inside	1.50 (7)	1.72 (7)	5.36 (-3)
		Outside	1.80 (7)	1.50 (7)	3.50 (-3)
	7	Inside	1.80 (7)	1.50 (7)	8.99 (-3)
		Outside	4.00 (6)	1.37 (7)	1.87 (-3)
	8	Inside	4.00 (6)	1.37 (7)	7.34 (-5)
		Outside	2.89 (6)	6.47 (6)	4.96 (-5)
	9	Inside	2.89 (6)	6.47 (6)	2.74 (-5)
		Outside	3.21 (-1)	5.33 (-1)	5.83 (-12)
	10	Inside	3.21 (-1)	5.33 (-1)	1.06 (-11)
		Outside	5.88 (-2)	5.20 (-2)	1.16 (-12)

*Figure in parentheses denotes the power of 10 by which the column entry is multiplied.

Figure 5 shows the dose rate vs. radial and axial concrete thickness at 116.4 Mw. This graph shows that the radial dose with 9.5 ft of ordinary concrete is approximately 2.5 mr/hr. It is believed that the calculations are somewhat conservative. Therefore, only 8.5 ft of concrete is provided in the radial direction. Space is available for adding 1 ft of heavy concrete if it is needed.

The belief that the shielding calculations are conservative is based on EBWR experience at the Argonne National Laboratory. Shielding experiments there showed that the biological shield of the EBWR was conservatively designed. Since the ERR shielding calculations were based on the method of calculation used by the Argonne National Laboratory, it is reasonable to suppose that the results are equally conservative.

3.2.8 Radioactive Waste Disposal

Because the reactor is designed so that only very small quantities of radioactive waste materials should accumulate, no facilities are provided at the site for concentrating and disposing of high-level radioactive wastes. The small quantities of wastes that will accumulate will be stored within the containment building until they can be shipped off-site for disposal.

Radioactive primary water is not expected to leak from the system at more than 3 gal/day. Normal leakage water from the control-rod thimbles will drain to a 50-gal collection tank (see Dwg. 12E-09010). Water can be returned by the makeup-system pumps from this tank to the primary system, as makeup.

Two 3000-gal retention tanks are provided in the basement of the containment building for collecting and storing waste water (see Dwg. 12E-09009). All building drains, as well as drain lines from the primary system, go to these tanks, which are manifolded to permit collecting the water in either or both. Piping and a 50-gpm retention-tank pump make it possible to pump water from one tank to the other or to discharge the water to the building drain system, which carries all effluent service water to the condenser circulating-water discharge line. This will be done only when analysis has shown that the activity of water discharged from the plant will not exceed the limits of Appendix B, Table II, of 10 CFR 20.

All service water and any water discharged from the retention tanks leave the containment building through the service-water discharge line, which contains the sewer monitor described in Sec. 3.5. If excessive amounts of activity were released from the retention tanks, they would be detected by the sewer monitor, which continuously records and indicates activity levels in the control room. The sewer monitor actuates an alarm for concentrations in excess of 2×10^{-5} microcurie/cc in the service-water discharge line, to ensure that activity discharge from reactor operation will not exceed requirements of 10 CFR 20.

To decontaminate waste water (for re-use or discharge), a portable demineralizer is provided. The piping is so arranged that water can be pumped from one retention tank through the portable demineralizer to the other retention tank. This pumping can be repeated until the water is pure. Water that cannot be easily decontaminated can be converted to a solid and shipped as such.

Small amounts of primary waste water will accumulate as a result of withdrawing samples of steam and water for boiler and radiochemical analysis. These will be collected in small shielded bottles for ultimate disposal.

Solid radioactive wastes will accumulate from purification system resins and filters. Most of the radioactive cations and anions of the primary system will eventually accumulate in the filter and resins of the purification system. If the activity level permits, the resin and filter cartridges will be removed and then placed in drums, which will be filled with concrete. The drums can then be shipped to a burial site. If the ion exchanger and filter become highly radioactive as a result of a fission-product release, provisions have been made for removing and replacing entire units. The radioactive units will be stored behind concrete blocks within the containment building, until their activity decays enough to permit shipment.

If clothing should become contaminated, it will be collected in drums in the decontamination room. The drums will be periodically sent to commercial laundry firms possessing decontamination facilities.

The only other solid wastes are the filters and adsorbers in the off-gas line. These will also be removed when they become contaminated and will be encased in concrete-filled drums for off-site burial.

The air conditioning system will pass 3000 cfm (3500 cfm max.) of filtered air continuously through the containment building. The air will enter through an air intake and pass through a fine filter. Air is discharged through a 24-in. exhaust duct in the ceiling of the containment building (see Dwg. 12E-01504). A blower within this duct ensures positive discharge of air. The air passes through a coarse filter at the inlet end of the duct, through an absolute filter, through the blower, past the stack isolation valves, and then out the stack. Stack gaseous and particulate monitors drawing a continuous sample of air downstream of the coarse filter will detect the presence of excessive activity in the stack exhaust. When high activity levels are detected, quick-closing dampers in the stack exhaust and the air intake dampers will be closed, isolating the containment building from the outside air.

All primary system vent lines and the reactor cavity vent lines enter the building exhaust duct between the coarse and absolute filters. The primary system vent lines include vent lines from all tanks and heat exchangers of the primary system and the recombiner system vent. During filling of the primary system, air will be exhausted through these lines to the exhaust duct and then out the stack. The vent lines are upstream of the stack monitors so that any activity discharged through the lines will be detected.

The vent line from the recombiner system will be the greatest potential source of stack activity. During normal operation, the only activity would be nitrogen-16, but this activity would be present in very small amounts. If a fuel assembly ruptured, however, fission products would leave the primary system and pass over to the recombiner system and then to the exhaust duct. The gaseous fission products xenon and krypton are the only fission products expected to be present in significant amounts. Solid fission products and the radioisotopes of iodine should be retained in the primary system water. In setting the amounts of activity which would cause isolation valve closure, however, it has been assumed that all fission products could be present in the stack exhaust.

Both the stack gas and stack particulate monitors actuate an alarm in the reactor control room and cause closure of the building isolation valves whenever the activity of the stack gas reaches a predetermined level. For the stack gas monitor, this setting has been taken as the count rate from a gaseous release rate of 2×10^4 microcuries/sec.

The stack particulate monitor contains two detectors. One detector gives an immediate reading of particulate activity in the stack effluent. The second reads the activity after a 6-hr decay period. It is thus possible to know whether particulate activity in the stack is due to long-lived or short-lived nuclides.

A high radiation signal from either of those detectors will actuate an audible and visible alarm in the reactor control room, and the immediate detector will actuate the stack isolation valve closure mechanisms. The high-radiation signal for the immediate detectors corresponds to a release rate of particulate activity from the stack of 20 microcurie/sec. The delayed detector produces a high-radiation signal for a release rate of 7.5×10^{-2} microcurie/sec.

The effluent from the turbine air ejector is monitored for gaseous activity. No activity would normally be present in this effluent. Unless evaporator or sub-cooler tubes fail, there is no way for activity in the primary system to enter the secondary system. The air-ejector monitor will detect the presence of gaseous fission products that could be present, however, if an evaporator or sub-cooler tube failed when a defective fuel assembly was in the reactor. This monitor will generate a high-radiation signal that actuates audible and visible alarms in the control room whenever activity exceeds a predetermined level. The high-radiation setting for this monitor corresponds to a release rate of 6 microcuries/sec.

A personnel decontamination room outside the containment building, adjacent to the airlock, contains a lavatory, a shower, and space for lockers and radiation detection equipment. Waste water from the decontamination room is collected in a 300-gal tank, which is checked periodically for radioactivity. The tank discharge valve leading to the sewer is chain-locked to prevent unauthorized disposal. In the absence of radioactivity, the water will be discharged directly to the sewer. When radioactivity is present, the water will be pumped to the containment-building retention tanks for cleanup by the portable demineralizer.

3.3 POWER PLANT

3.3.1 Layout and Flowsheet

The power generating portion of the plant is in the RCPA power-plant building (see Dwg. 12E-01500). This building contains three boilers, two 11,500-kw turbine-generators, and a 22,000-kw turbine-generator. The 22,000-kw turbine-generator will be driven by steam from the nuclear portion of the plant, and is designed to operate with either 850-psig, 900 F steam or 600 psig, 825 F steam at its inlet. The nuclear portion of the plant will generate steam at the lower pressure and temperature during initial operation (see Fig. 3.17). The reactor, auxiliary systems, and primary circuits are in the containment building, and the separately fired superheater is in a separate building of conventional construction between the power plant and the containment building (see Fig. 3.18).

Primary steam condensed in the evaporators generates dry saturated secondary steam at 700 psig. The secondary steam flows through the superheater and is delivered to the turbine inlet at 600 psig and 825 F. After a period of operation at these steam conditions, it is planned to increase the pressures and temperatures of the reactor, steam generator, and turbine inlet to the values shown in Fig. 3.17.

A condenser keeps the turbine exhaust pressure at 1.5 in. Hg abs. Condensate is collected in a hotwell at the base of the condenser. The condensate, as well as approximately 2000 gal/day of demineralized makeup water, is pumped to the low-pressure heaters and then to the deaerator. Water from the deaerator flows through the feed-water pumps to a high-pressure heater, after which it leaves the steam-plant building and returns to the subcoolers, thus completing the secondary-system cycle.

3.3.2 Superheater

A coal-fired superheater designed for indoor operation occupies a 40-ft x 40-ft area adjacent to the steam-plant building. It consists of a pulverized-coal-fired combustion zone, a superheater section, induced draft and forced-draft fans, a regenerative air heater, soot blowers, coal feeders, coal pulverizers, a dust collector, ducting, and a stack. Automatic combustion and steam temperature controls provide for superheat control, balanced operation, and shutdown requirements.

The coal feed is adjusted so that it is proportional to the load demand, and the air supply is regulated to maintain the proper air-to-coal ratio for any given load. A turn-down ratio to one-quarter load capacity is guaranteed by the manufacturer. Raw coal is automatically and continuously fed to either pulverizer to produce fires for the furnace. The pulverized coal is carried in suspension with preheated air to the burners. Individually adjustable air vanes ensure a uniform and turbulent mixture of air to the burner nozzles for efficient firing and flame control. If the steam supply

to the superheater fails suddenly, the furnace can be shut down rapidly by an interlock between the steam flow and coal feed system. The holdup time for pulverized coal in either pulverizer will not exceed 15 sec at the maximum capacity of the unit.

Controls maintain the desired steam temperature, despite variations in steam flowrate, furnace conditions (as determined by CO₂ and excess oxygen percentages), and total air input rate.

The partially superheated steam leaving the division wall of the superheater will actuate a spray desuperheater if its temperature is too high. Thermocouples in both the inter-stage and final stage sections, in conjunction with a temperature recorder-transmitter-controller, will regulate the desuperheater spray system valve to maintain final steam temperature to within ± 10 F of design limits.

3.3.3 Turbine-Generator

The turbine-generator is of standard design. It is rated at 22,000 kw at a steam power factor of 85 percent and is designed for 850 psig, and 900 F steam, or for 600 psig, 825 F steam at its inlet. The rated exhaust pressure is 1.5 in. Hg abs. The turbine speed is 3600 rpm; the exciter is direct-connected.

3.3.4 Condenser and Auxiliaries

The turbine condenser is a 22,000-ft², single-pass, divided-water-box, horizontal surface condenser. It maintains the turbine exhaust pressure at 1.5 in. Hg abs at the rated load. An air ejector removes noncondensable gases from the condenser. The steam from the ejector flows through and inter- and after-condenser, where it heats the main condenser condensate, and then flows to the main condenser. Two low-pressure heaters arranged in series heat the condensate somewhat; the condensate then flows through the deaerator to the two boiler feed pumps, from which it flows through a high-pressure feedwater heater and then back to the shell sides of the subcoolers.

3.3.5 Service Water System

General service water for the containment building and reactor auxiliary systems is taken from the Mississippi River. A 54-in. line from the river delivers water by gravity to a concrete sump pit in the basement of the steam-plant building. The water is strained by two traveling screens in the pit. Either of two 750-gpm pumps supplies general service water from the pit to the steam plant. An 1100-gpm vertical pump supplies general service water to the containment building.

A typical analysis of Mississippi River is given below.

total dissolved solids	261 ppm
organic matter	28 ppm
phenolphthalein alkalinity (CaCO ₃)	8 ppm
methyl orange alkalinity (CaCO ₃)	162 ppm
hardness (CaCO ₃)	152 ppm
silica (SiO ₂)	6 ppm
iron (Fe ₂ O ₃)	0.11 ppm
calcium (Ca)	44 ppm
magnesium (Mg)	9 ppm
sodium (Na)	8 ppm
bicarbonate (HCO ₃)	177 ppm
carbon dioxide (CO ₂)	9 ppm
chlorides (Cl)	2 ppm
sulfates (SO ₄)	6 ppm
nitrates (NO ₃)	trace
turbidity	10 ppm
pH	8.0

The circulated service water is returned to a separate compartment of the sump pit; from there it flows back to the river by gravity.

The service-water system supplies cooling water for several of the heat exchangers used in the auxiliary reactor system (Dwg. 12E-09008). One main service water line enters the containment building. Water used on the shell sides of the emergency condenser, the fuel-element storage-well heat exchanger, the decay-heat-removal-system heat exchanger, the purification-system heat exchanger, the shield-cooling-system heat exchanger, and the recombiner-system condenser will be drawn from this line. The air conditioners are also cooled with service water. All service water leaves the containment building through a single return line.

3.3.6 Electrical System

Electrical power for starting up the reactor plant and the superheater is supplied by a 69 kv/480-v reserve auxiliary transformer energized by the RCPA 69-kv bus, but after startup the reactor-plant and the superheater auxiliary load is transferred to the 13.8 kv/480-v auxiliary transformer energized by the generator driven by the nuclear steam supply system. If, after the reactor-plant and superheater auxiliary load have been so transferred, the 480-v bus feeding the reactor plant or the superheater is de-energized, the auxiliary load is automatically re-transferred to the reserve auxiliary transformer energized by the RCPA 69-kv bus.

A 440-v distribution bus in Motor Control Center 3 (in the containment building) is energized automatically by a 50-kw gasoline-engine-driven generator if both sources of auxiliary power fail or are isolated from Motor Control Center 3. This bus supplies electrical power for the following equipment:

- (a) all 13 control-rod drive mechanisms (1/6 hp each),
- (b) both primary-coolant purification-system pumps (5 hp each),
- (c) both shield-cooling-system pumps (5 hp each),
- (d) the decay-heat-removal-system pump (15 hp),
- (e) the stack monitor vacuum pumps (1 hp each),
- (f) the 125-v battery-charger (20 kw), and
- (g) the 480-v/120-v distribution transformer that supplies electrical power to the failure-free bus if the motor-generator set fails (10 kva).

Electrical power for starting the 50-kw gasoline-engine-driven generator is supplied by a 12-v storage battery.

The a-c failure-free bus is ordinarily energized from the 125-v d-c distribution bus through a 125-v d-c -- 120-v a-c motor generator set; but if the motor-generator set fails, the a-c failure-free bus is automatically connected to the 440-v distribution bus in Motor Control Center 3 through a 10-kva 440-v/120-v distribution transformer.

The 125-v d-c distribution bus is ordinarily energized by the 20-kw battery-charger; but if the battery-charger fails, the 125-v d-c bus is energized by a 125-v battery. The 125-v d-c distribution bus is provided with a tie to the station battery.

The a-c failure-free bus supplies electrical power to the reactor control console, all nuclear instrumentation, radiation monitors, area monitors, primary pressure transmitters C-7 and H-51, secondary pressure transmitter H-50, fire-eye equipment for the superheater, and the evacuation siren.

Electrical power for emergency lights in the containment building and the superheater building is ordinarily supplied from the 120-v/208-v distribution bus in the containment building; but if that distribution bus is de-energized, the emergency lighting load is automatically transferred to the 125-v d-c distribution bus.

The 125-v d-c bus supplies the power to all control-rod drive mechanism magnetic clutches and the power necessary for remotely actuating the following systems:

- (a) the sodium-pentaborate injection system,
- (b) the emergency-and-test-condenser flow-control valve,
- (c) the reactor-core emergency-cooling-system flow-control valve,
- (d) the containment-building spray-system flow-control valve,
- (e) the containment-building ventilation-system inlet and outlet dampers, and
- (f) the reactor control-room annunciators.

3.4 CONTAINMENT BUILDING

3.4.1 Design, Testing, and Internal Arrangement

The containment building is a concrete-lined, gastight steel shell consisting of an upright cylinder with a hemispherical top and a semi-ellipsoidal bottom. It encloses (a) all principal components of the reactor plant except the reactor control room, the superheater and its auxiliaries; and the turbine-generator and its auxiliaries; (b) all portions of the primary system; and (c) all auxiliary systems that contain primary coolant.

The material of the shell is carbon-silicon steel conforming to ASTM Specification A201-57T, Grade B, heat-treated in accordance with ASTM Specification SA-300. The shell was designed, constructed, and tested in accordance with Sec. VIII of the ASME Boiler and Pressure Vessel Code for:

- (a) a maximum internal pressure of 21 psig at 220 F,
- (b) a minimum internal pressure of 0.33 psig,
- (c) a minimum free-air volume of 287,000 cu ft, and
- (d) a welded-joint efficiency of 90 percent.

The containment building is 115 ft high; its internal diameter is 74 ft. The bottom ellipsoidal portion of the shell is 0.70 in. thick, the cylindrical sides are 0.875 in. thick, the hemispherical top is 0.5 in. thick, except for the portion directly over the 30,000-gal water tank, which is 0.70 in thick (see Dwg. 12E-11003).

The building extends 18.5 ft below grade and 96.5 ft above grade. Corrosion protection is afforded both inside and outside by bituminous enamel. The outer surface of the shell above grade is covered with a painted layer of flex-cell insulation 2 in. thick.

The inside of the shell is lined with 2 ft of concrete on the cylindrical sides. The bottom of the building has at least 2 ft of concrete at all points. The hemispherical top has a concrete lining 2 ft thick near the base of the hemisphere, decreasing to approximately 4 in. at the edge of the water tank. The water tank is not lined with concrete. The concrete lining will afford missile protection and attenuate gamma radiation if fission products escape from the reactor.

The following test and inspection procedures were carried out before final acceptance of the containment building:

- (1) all welds were visually inspected;
- (2) all butt welds were fully radiographed in accordance with Paragraph UW-511 of the ASME Boiler and Pressure Vessel Code;

(3) all welds on doors, nozzles, and opening frames and all welds that could not be radiographed were inspected by the magnetic-particle or dye-penetrant method;

(4) all containment-shell welds were leak-tested by the soap-bubble method while the shell was pressurized at 2 psig;

(5) all containment-shell welds were leak-tested by the soap-bubble method while the shell was pressurized at 21 psig;

(6) after the containment shell had been erected, it was pressurized to 21 psig for three days while leakage was measured (the measured leak rate was less than 0.1 percent/day); and

(7) after all containment-shell equipment had been installed and after all containment-shell penetrations were made, the containment shell was again pressurized to 21 psig for three days while leakage was remeasured (the measured leak rate was again less than 0.1 percent/day).

Defective welds disclosed by soap-bubble testing, pressure testing, or radiography were repaired.

The building has a main (top) floor, an intermediate floor, a basement, and a sub-basement (see Dwgs. 12E-01501 through 12E-01505). Equipment on the main floor includes the evaporator portions of the off gas system and two 20-ton air conditioners. The air conditioners draw recirculating air from, and return it to, the basement, intermediate, and main floors. One air conditioner draws approximately 3000 cfm of fresh air from the outside. The displaced building air is exhausted through a stack at the top of the building. The exhaust air passes through one coarse and one fine high-efficiency filter. The air conditioners are used for heating when necessary.

Equipment on the intermediate floor includes the sodium-pentaborate storage tank and its associated compressor, the shield-cooling-system expansion tank, a 20-ton air conditioner, and the makeup-system pumps. The subcoolers and the decay-heat-removal-system heat exchanger are between the intermediate floor and the basement.

Equipment in the basement and subbasement includes the decay-heat-removal-system pump, the sump pumps, the motor control centers, switchgear, the startup heater, the retention-tank pump, the fuel-element storage-well pump, the purification-system pumps and heat exchangers, the shield-cooling-system pumps and heat exchanger, the ion exchangers, the retention tanks, and the control-rod drive mechanisms. The sub-basement is reached by stairs from the basement.

3.4.2 Penetrations

The containment-building penetrations are described in the following paragraphs:

The containment building is penetrated by a personnel airlock, an emergency airlock, a freight door, two ventilation ducts, two vacuum-breaker ducts, one emergency-and-test condenser vent pipe, two valve-operating shafts, 18 pipes, and approximately 274 electrical cables.

The personnel airlock is the ordinary means of access to the containment building. Its 5-ft by 7-ft doors are large enough to accommodate the fuel shipping cask. It is 24 ft 9 in. long and approximately 11 ft in diameter. The emergency airlock is approximately 10 ft long and 4 ft 9 in. in diameter; the doors are 2 ft 6 in. in diameter.

Each airlock has two gasket-sealed doors arranged so that containment-building overpressure will further compress the gasket of any closed door. The two doors of each airlock are mechanically interlocked to ensure that at least one is closed and sealed under all conditions for which containment integrity is required. Door operators are coupled to pressure-relief valves to equalize pressure before either door is opened.

The freight door is sealed with gaskets arranged so that containment-building overpressure, as well as the door-closing devices, compresses them. During reactor operation, removable concrete shield blocks are placed in front of the freight door.

Each ventilation duct is seal-welded into the containment shell and is equipped with two in-line quick-closing butterfly dampers held closed by springs whenever they are not held open by a pneumatic cylinder operated from the service-air supply system.

Each vacuum-breaker duct is sealed with a gasketed disk arranged so that containment-building overpressure, as well as the disk, compresses the gasket. Each vacuum breaker is set so as to prevent the atmospheric pressure outside the containment building from exceeding the air pressure inside the containment building by more than 0.22 psi.

The emergency-and-test condenser vent pipe is welded to the containment-building shell and is sealed from the atmosphere of the containment building by the shell side of the emergency and test condenser, to which it is welded. Valves are arranged so that the escape of gases from the containment building into the shell side of the emergency and test condenser can be prevented.

Each valve-operating shaft that penetrates the containment-building shell passes through a pipe that is welded into the shell and fitted at each end with a packing gland.

Each pipe that penetrates the containment-building shell is welded to the shell. Valves on all except five pipe penetrations are arranged so that neither the failure of any single valve to close properly nor a pipe rupture will prevent isolation of that line. The five exceptions are as follows: the containment-building ventilation inlet and outlet ducts, both vacuum-breaker penetrations, and the emergency-and-test-condenser vent pipe.

Each electrical cable that penetrates the containment-building shell falls into one of three classes: multi-conductor control cables (Class A), coaxial cables (Class B), and

all other cables (Class C). The method by which each cable in Classes A, B, and C is sealed is as follows:

(a) Each Class A cable passes through a Thomas and Betts watertight strain relief connector mounted in a 1-1/4-in. IPS tapped hole in the containment-building shell and a second Thomas and Betts watertight strain relief connector mounted in a 1-1/4-in. IPS tapped hole in a plate 3 in. from the containment-building shell. The chamber between the containment-building shell and the connector mounting plate is filled with "NOVOID C" potting compound, manufactured by the G & W Electric Specialty Company.

(b) Each Class B cable terminates on either side of the containment-building shell at an Amphenol Series 83 pressure-tight bulkhead fitting mounted in a gasketed Micarta plate. The Micarta plate in this type of penetration is of Type 20201-2 Micarta that conforms to ASTM Grade G-7. The Micarta penetration is designed so that the stresses in the plate under the specified containment-shell design conditions do not exceed 1/6 of the ultimate strength of the material.

(c) Each Class C cable passes through pipe sleeves welded into the containment-building shell. In each such penetration, the cable enters the pipe sleeve through a Type EYS fitting with an O-Z watertight cable seal inside the containment-building and leaves the pipe sleeve through an O-Z insulator bushing outside the containment-building shell. The sleeves are filled with Scotch-Cast Resin No. 4.

Terminals of electrical cables are sealed in accordance with Detail 2 of Dwg. 12E-11809.

3.4.3 Spray System

The containment-building spray system consists of a standpipe extending from the basement of the containment building into the 30,000-gal water storage tank at the top of the containment building, a motor-operated valve at the lower end of the standpipe, and pipes leading from the gate valve to spray nozzles throughout the containment building.

The containment-building spray-system standpipe extends into the 30,000-gal water storage tank far enough to ensure that the capacity of the tank below the top of the standpipe is approximately 15,000 gal.

The containment-building spray-system motor-operated valve is the only impediment to water flow from the portion of the 30,000-gal storage tank above the top of the standpipe to all the containment-building spray-system nozzles.

The spray system is designed to spray approximately 15,000 gal of water into the atmosphere of the containment building in approximately 15 min. The spray-system motor-operated valve is electrically operable from the reactor control console, and mechanically operable from the decontamination room and from a point inside the containment building.

The spray system is automatically actuated, after a 10-min delay time, by a pressure signal indicating that the containment-building pressure exceeds 2 psig, but automatic operation of the system can be prevented by manual action in the control room.

The spray system would have two main functions after release of primary steam. It would reduce the pressure and temperature built up after steam release. If the core melted and released fission products, the spraying system would wash solid fission products from the air. In 14 min the system can reduce containment-building pressure and temperature from 20.3 psig and 217 F to 8.4 psig and 168 F (see Appendix B).

3.5 RADIATION MONITORING SYSTEM

The reactor is monitored to ensure that fission products and normally produced gaseous or particulate radioactivities are not released in excessive amounts from the plant and that such reactivities do not present hazards to operating personnel. Drawing 12E-11927 shows the radiation monitoring system. The monitors shown in this drawing and discussed below are (1) the stack monitors, (2) the fission-product monitor, (3) the area monitors, (4) the sewer monitor, (5) the secondary-system water monitor, (6) the plant particulate monitor, and (7) the air ejector monitor. The measurements from all radiation monitors except the area monitors in the dispatcher's office and the control room are indicated and automatically recorded in the control room; high-radiation signals from any monitor except the area monitor in the dispatcher's office actuate audible and visible alarms in the control room.

3.5.1 Stack Monitors

The containment-building stack effluent is monitored for gamma gaseous activity and particulate beta activity by the stack monitor. This monitor incorporates three detectors: one for immediate continuous detection of beta particulate activity; one for immediate continuous detection of gamma gaseous activity; and one for delayed continuous detection of beta particulate activity. Each detector is provided with indicating and alarm devices and circuitry. The delayed particulate detector senses the activity of any trapped particles approximately 6 hr after the immediate detector senses the activity of the same trapped particles. It is therefore possible to determine whether the particulate activity consists of short-lived or long-lived nuclides.

The sampling system for the stack monitor consists of a vacuum pump which draws gas from the ventilation exhaust duct and passes it through an aperture over which a moving filter passes. A beta detector behind the aperture senses activity collected on the filter paper. A beta scintillation detector senses beta activity on the filter paper after 6 hr. The exit air from the particulate monitor is drawn into a cylindrical chamber with a volume of 0.1 cu ft, where it is monitored by a gamma detector.

3.5.2 Fission-Product Monitor

Any fission-product activity escaping into the off-gas system would be principally from xenon-135, xenon-138, and cesium-138. These nuclides emit gammas with energies near 0.475 Mev.

Normal activity in the off-gas system will be predominantly nitrogen-16, which yields high energy gammas (6.1 and 7.1 Mev). However, Compton scattering will produce a flux of low-energy gammas that would mask out the low-energy fission-product gammas.

The fission-product monitor consists of a scintillation detector with output pulse heights proportional to the energies of the incident photons. The output from the detector

is fed to a preamplifier whose output is connected to the linear pulse amplifier. The linear pulse amplifier has two outputs: One is fed to a single-channel analyzer and the other to a discriminator.

The single-channel analyzer is set to accept pulses corresponding to gamma energies of $0.475 \text{ Mev} \pm 2 \text{ percent}$. The output of the analyzer is thus proportional to fission-product activity plus the activity from Compton scattering of nitrogen-16 gamma emission.

The discriminator is set to reject pulses below those corresponding to 5-Mev gamma energies. The output will thus be proportional to nitrogen-16 activity only.

The outputs from the analyzer and the discriminator feed a comparator, where the nitrogen-16 activity count rate is subtracted from the combined count rate of nitrogen-16 and fission-product activity. The difference, which is a measure of fission-product activity, is indicated locally and recorded in the control room. The fission-product monitor is capable of detecting amounts of gaseous fission-product gamma activity of not less than twice the normal background.

3.5.3 Area Monitors

Each of the 13 area monitors incorporates a gamma-activity detector, an electrometer, and indicating and alarm devices and circuitry for immediate and continuous detection of gamma radiation fields.

One area monitor senses gamma radiation near each of the following components: the bottom plug of the reactor-vessel cavity, the startup-heating-system heat exchanger, each evaporator, each subcooler, each ion-exchange column, the shield-cooling-system heat exchanger, and the emergency condenser. One area monitor senses gamma radiation in each of the following spaces: the dispatcher's office, the reactor control room, and the main floor of the containment building.

Each area monitor can detect gamma radiation levels above 10 mr/hr . Each generates a high-radiation signal whenever its associated devices and circuitry indicate a gross gamma-radiation level more than twice the full-power background level in its vicinity during full-power operation.

A high-radiation signal generated by any of the area monitors except the monitor in the dispatcher's office actuates audible and visible alarms at the location of the monitor and in the reactor control room. A high-radiation signal generated by the area monitor in the dispatcher's office actuates audible and visible alarms only at the monitor location.

3.5.4 Sewer Monitor

The sewer monitor detects concentrations of 3×10^{-6} microcurie/cc for gamma emitters whose energy of emission is 1 Mev or greater. The monitor is in the service-water discharge line through which all the service water from the containment building must pass.

3.5.5 Secondary-System Water Monitor

The secondary-system water monitor detects concentrations of 6×10^{-6} microcurie/cc for gamma emitters whose energy of emission is 1 Mev or greater. This monitor is in the secondary feedwater return line upstream from the subcoolers.

3.5.6 Plant Particulate Monitor

A mobile particulate monitor identical with the particulate section of the stack monitor measures the activity of airborne particles at any of four locations within the containment building: three locations on the basement floor and one on the main floor.

The mobile particulate monitor senses both immediate and delayed activity. The activity sensed by each detector is indicated and recorded locally and in the control room. Audible and visible alarms are actuated locally and in the control room upon high-radiation signals.

3.5.7 Air Ejector Monitor

A monitor on the turbine air-ejector exhaust line immediately and continuously detects gaseous beta-gamma activity. The instrument is provided with indicating and alarm devices and circuitry; its minimum sensitivity corresponds to a stack release rate of 1 microcurie/sec, the same sensitivity as the gaseous section of the stack monitor.

ELK RIVER REACTOR PLANT LOCATION
Elk River, Minnesota

AERIAL VIEW OF SITE

FIG. 3-2

ERR CONTROL-ROD DRIVE

FIG. 3-3

DECAY HEAT VS. TIME AFTER SHUTDOWN

FIG. 3-4

Detector No.	Description	Sleeve Location
N-1	startup BF_3	radial
N-2	startup BF_3	radial
N-3	log-n	tang. - top
N-4	log-n	tang. - top
N-5	linear power	tang. - bottom
N-6	linear power	tang. - bottom
N-7	level safety	tang. - top
N-8	level safety	tang. - bottom
N-9	level safety	tang. - bottom
Spare		tang. - top

NEUTRON DETECTOR INSTRUMENT LOCATION

FIG. 3-5

k_{eff} AS A FUNCTION OF VOID FRACTION
(neutron temperature = 368 C; no xenon or samarium present)

FIG. 3-6

k_{eff} AS A FUNCTION OF MODERATOR TEMPERATURE AT ZERO POWER
FOR UNRODDED CORE

FIG. 3-7

AXIAL THERMAL FLUX AS A FUNCTION OF CONTROL-ROD POSITION

FIG. 3-8

k_{eff} AS A FUNCTION OF INSERTION DISTANCE, HOT CORE

FIG. 3-9

k_{eff} AS A FUNCTION OF FUEL IRRADIATION

FIG. 3-10

VARIATION OF U-235 CONTENT AS A FUNCTION OF FUEL IRRADIATION FIG. 3-11

VARIATION OF U-233 + Pa-233 CONTENT AS A FUNCTION
OF FUEL IRRADIATION

FIG. 3-12

VARIATION OF BORON-10 CONTENT AS A FUNCTION
OF FUEL IRRADIATION

(initial boron-10 content: 119 ppm by weight in clad)

FIG. 3-13

TIME IN DAYS AS A FUNCTION OF FULL-POWER FUEL IRRADIATION

FIG. 3-14

RELATIVE THERMAL-NEUTRON FLUX AS A FUNCTION OF
DISTANCE FROM VERTICAL CENTERLINE OF CORE

FIG. 3-15

DOSE RATE VS. THICKNESS OF CONCRETE

FIG. 3-16

INITIAL OPERATING CONDITIONS

point (s)	pressure psig	temperature °F	flowrate lb/hr
A	922	536	129,000
B	921	536	129,000
C	919	450	129,000
D	700	505.5	112,500
E	700	505.5	225,000
F	600	825	225,000
G	712	350	225,000
H	712	350	112,500

FUTURE OPERATING CONDITIONS

point (s)	pressure psig	temperature °F	flowrate lb/hr
A	1153	564	125,000
B	1152	564	125,000
C	1150	473	125,000
D	913	536.6	102,500
E	913	536.6	205,000
F	850	900	205,000
G	925	350	205,000
H	925	350	102,500

OVER-ALL PERFORMANCE DATA

	initial	future
reactor thermal output	58.2 Mw	53.5 Mw
superheater thermal output	14.8 Mw	15.5 Mw
gross electrical output	23.0 Mw	24.2 Mw
net electrical output	22.0 Mw	23.1 Mw
gross efficiency	31.5 %	35.3 %
net efficiency	30.2 %	33.5 %

SIMPLIFIED FLOW DIAGRAM

FIG. 3-17

CONTAINMENT BUILDING CUTAWAY,
SUPERHEATER BUILDING, AND POWER PLANT

FIG. 3-18

4. INHERENT ABILITY OF THE REACTOR TO OFFSET REACTIVITY INCREASES

4.1 INTRODUCTION

The response of the reactor to fast and slow increases in reactivity was analyzed to determine the maximum amount by which reactivity may be increased without damaging the fuel elements or the core. The results were compared to the reactivity increases associated with various accidents.

The Elk River Reactor, like the other boiling-water reactors (EBWR, BORAX, Dresden, etc.), has a large negative void coefficient of reactivity. Hence, it is inherently capable of offsetting rapid and fairly large reactivity increases. Steam-void formation alone is effective in terminating transients on stable periods as short as 100 millisecc, or reactivity increases as large as 0.65 percent. The ability of the reactor to offset rapid reactivity increases greater than 0.65 percent, which produce stable periods shorter than 100 millisecc, is more dependent on fuel temperature rise (Doppler effect) than on steam-void formation. This results from the relatively low thermal conductivity of the ceramic fuel material, and has been demonstrated in the BORAX IV reactor. If reactivity is increased relatively slowly, steam-void formation and fuel temperature rise are both effective in limiting the transient.

4.2 EFFECTS OF RAPID REACTIVITY INCREASES

Table 4-1 summarizes the effects of rapid reactivity increases. The reactivity increase which will melt the fuel tubes or the fuel material assuming shutdown by (1) fuel temperature rise, (2) steam void formation, and (3) the combined effects of both is listed for each of three sets of initial conditions: (1) room temperature, zero power; (2) operating temperature, full power; and (3) operating temperature, zero power. Because the combined shutdown effects of fuel temperature rise and steam void formation were not analyzed, the minimum reactivity increase that will melt the fuel tubes or the fuel material when both mechanisms are effective is merely listed as being greater than the larger of the two values that result from considering each mechanism singly. The values in Table 4-1 apply to the fuel tube exposed to the maximum local neutron flux.

4.2.1 Room-Temperature, Zero-Power Increases

The only credible means of increasing reactivity rapidly under room-temperature, zero-power conditions would be by inadvertently withdrawing a control rod or adding a fuel assembly during refueling, when, for some reason, the reactor was nearly critical. This would result in a fast transient that would be terminated largely by fuel temperature rise.

Calculations in Appendix D show that reactivity could increase by as much as 1.16 percent (corresponding to a 7-millisecond period) before any fuel tubes would melt, if it is assumed that the fuel-tube temperature rise is adiabatic (i.e., that no heat flows from the fuel tubes to the water).

TABLE 4-1
EFFECTS OF STEP INCREASES IN REACTIVITY

initial conditions		effective shutdown mechanism		minimum step reactivity increase required	
reactor water temp., °F	reactor power, Mwt	fuel temp. rise	steam void formation	to melt fuel tubes	to melt fuel material
68	0	x		1.16%	
68	0		x	0.65%	
68	0	x	x	>1.16%	
68	0	x			2.92%
536	58.2	x		0.90%	
536	58.2		x	0.65%	
536	58.2	x	x	>0.90%	
536	58.2	x			1.20%
527	0	x		0.93%	
527	0		x	0.65%	
527	0	x	x	>0.93%	
527	0	x			2.68%

The unavoidable formation of at least some steam voids during the transient necessitates increasing reactivity by more than 1.16 percent in order to melt the fuel tubes. A small amount of fuel material would melt if reactivity were increased by slightly more than 2.92 percent, again assuming that the fuel-tube temperature rise is adiabatic. Reactivity increases up to 3.0 percent (for which the corresponding stable period is 1.5 millisecond) would hence be terminated by the combined effect of void formation and fuel temperature rise before the fuel material melted.

4.2.2 Operating-Temperature, Operating-Power Increases

Rapid reactivity increases at operating power and temperature are not likely. However, if an increase should occur, the fuel-temperature rise would be less effective in terminating the transient. Steam void formation would offer inherent safety for rapid reactivity increases only up to about 0.65 percent. Fuel temperature rise alone would be effective in preventing fuel-tube melting for reactivity increases up to 0.9 percent. However, more than 0.9 percent would, in reality, be required because of steam void formation. The fuel material would melt after a reactivity increase of 1.2 percent if it is assumed that the fuel-temperature rise is adiabatic. However, because of steam void formation, more than 1.2 percent would actually be required.

4.2.3 Operating-Temperature, Zero-Power Increases

If reactivity increases rapidly while the reactor is at operating pressure and temperature, but at zero power, fuel-temperature rise is the principal shutdown mechanism. Steam void formation is effective as a shutdown mechanism, as before, for reactivity increases up to 0.65 percent. An increase of 0.93 percent is required to melt the fuel tubes if it is assumed that the fuel temperature rise is adiabatic. Hence, more than 0.93 percent would actually be required. An increase of 2.68 percent is required to melt the fuel material if the fuel temperature rises adiabatically. Hence, considering steam void formation, a reactivity increase greater than 2.68 percent would be necessary.

4.2.4 Discussion

Fuel temperature rise and steam void formation undoubtedly complement each other in providing inherent reactor safety. Analytical means of determining their combined effect are, however, not available. Consideration of each separately, however, ensures conservative evaluations.

The reactivity increase controllable by steam void formation was taken from BORAX IV results. Although the pressure in these tests was 300 psi, the results for 900 to 1200 psi would, based on BORAX and SPERT tests, be more favorable for the Elk River Reactor. The BORAX IV data were selected because thorium-uranium ceramic fuel pellets similar to those of Elk River Reactor were employed. The diameter of the BORAX IV pellets (and hence the thermal time constant) was smaller than that of the ERR fuel pellets, but not enough to destroy the validity of the comparison.

The probability of a rapidly increasing reactivity in the amounts discussed above is extremely slight. Fuel will be added to the reactor only when the center control rod is at cocked safety and all other control rods are fully sealed. When the reactor is

near criticality, only one fuel assembly will be added at a time. Frequent criticality checks will be made after the addition of each fuel assembly. An accident caused by the removal of a control rod is also improbable. With all rods in, the removal of any one rod cannot cause criticality, since the negative reactivity at shutdown is greater than the reactivity worth of any single rod.

4.3 EFFECTS OF SLOW REACTIVITY INCREASES

4.3.1 Room-Temperature, Zero-Power or Low-Power Increases

The inadvertent withdrawal of all control rods while the reactor is at low power or sub-critical will lead to the fastest period for normal rod motion. The control rods move at a fixed withdrawal speed of 30 in./min, which, in the cold, fully loaded core, corresponds to an average reactivity increase rate of 0.012 percent/sec per rod. In the fully loaded core, the maximum rate at which reactivity may be increased by control-rod motion is 0.06 percent/sec, which is the rate associated with withdrawal of the center control rod when it is at or near its point of highest incremental worth. If the center rod is withdrawn when the reactor is shut down, the period will be 0.125 sec when the flux rises to the flux-scam level (125 percent of full-power flux). This period corresponds to a much slower rate of reactivity increase than those considered in Sec. 4.2. Damage to the reactor from this accident is therefore not likely. No other cause of a slow reactivity increase that results in a shorter period appears credible.

4.3.2 Operating-Temperature, Operating-Power Increases

Fuel-assemblies can be damaged by large reactivity increases that occur at a relatively slow rate. This might be assumed to result, for example, from the inadvertent withdrawal of all the control rods at their fixed speed. (Actually, an improbable multiple failure of the safety circuits would also be required.) A very large reactivity increase could occur by this means without reactor instability. The void and temperature distribution would remain essentially in equilibrium with the reactivity and power increases.

The burnout heat flux for the ERR fuel assemblies is about three times greater than the operating maximum heat flux. Hydrodynamic instability would occur at exit void fractions corresponding to power increases by a factor of 3. Failure of the fuel tubes from an adiabatic temperature rise would not be a problem for slow increases in reactivity. Rather, failure of the tubes would be a result of burnout heat fluxes or excessive internal gas pressure. The limiting power at which damage would start to occur appears to result from a power increase by a factor of 1.7. The damage would consist of fuel-material melting in the hottest regions of the core. Thus, a slow increase in reactivity on the order of 2 to 3 percent while the reactor is at power could be tolerated without dangerous consequences.

5. ENUMERATION AND EVALUATION OF POSSIBLE HAZARDS

The following have been evaluated from a safety standpoint and are discussed in the following subsections:

- (1) primary-system rupture,
- (2) secondary-system rupture,
- (3) loss of turbine load,
- (4) water leakage,
- (5) shield-cooling failure,
- (6) too much or too little water in reactor,
- (7) startup accident,
- (8) sudden introduction of cold water,
- (9) change in power demand,
- (10) power failure,
- (11) instrument failure,
- (12) fuel-tube failure,
- (13) melting of fuel tubes,
- (14) control-rod failure,
- (15) metal-water reactions,
- (16) sabotage,
- (17) fire,
- (18) earthquake,
- (19) projectiles,
- (20) operator error, and
- (21) loss of boron.

5.1 PRIMARY-SYSTEM RUPTURE

The consequences of a primary-system rupture depend on the size and location of the break.

5.1.1 Large Break Above Core

A break in one of the 8-in. feedwater inlet lines would constitute the most serious above-core rupture. Vessel water would flow out to the level of the bottom of the rupture opening, and approximately 35 percent of the remaining water would flash to steam. Thus, after the rupture, the water level would drop to approximately 10 in. below the horizontal midplane of the core. The core is then cooled by the boiling of the water remaining in the core and the evaporation of the emergency cooling water which enters the reactor vessel through the spray ring. The makeup-system pumps automatically start injecting 25 gpm of water through the spray ring as soon as the water level drops to within approximately 3 ft of the top of the core. Evaporation of this water is ample for removing the decay heat. Eventually, the core is re-covered with

no melting. If the makeup pumps fail, two alternative sources of emergency cooling water are available: the building overhead storage tank and the service-water supply. Water can flow by gravity from the overhead storage tank to the spray ring whenever the reactor-vessel pressure is no more than 25 psi higher than the containment-building pressure.

5.1.2 Small Break Below Core

A small break below the level of the core could result in the eventual draining of all water from the vessel. For breaks 0.9 in. in diameter or larger, the time during which the core would remain covered is 7 min or less. Emergency cooling water from the overhead storage tank could thus enter the vessel before the core started to melt (see Appendix A). For smaller breaks, the core would remain covered for more than 7 min, and the pressure would not be reduced enough to allow emergency cooling water to drain by gravity from the overhead storage tank. For this reason, the makeup system has been designed to deliver 25 gpm to the vessel when the vessel pressure is 1200 psia (see Sec. 3.2.4.4.2).

5.1.3 Large Break Below Core (See Appendix A)

The time required to drain the water from the vessel is obviously greater for small breaks than for large breaks. Therefore, by the time some portions of the core have reached the melting point, the rate of decay-heat generation is lower in the case of a small break than in the case of a large break. The total meltdown fraction is therefore greatest for a large rupture below the core, and this is the most serious kind of rupture. If one of the 16-in. forced-circulation inlet nozzles ruptured, all water would be drained from the vessel in approximately 10 sec. This is the maximum credible accident discussed in Appendix A. Even with cooling water flow, some of the stainless-steel fuel tubes might eventually melt. If low-pressure water injection started any time within 7 min after rupture (melting would start 6.8 min after the rupture), 8.62 percent of the fuel tubes might melt. The hazards to the surrounding area resulting from this accident are evaluated in Appendix C.

5.2 SECONDARY-SYSTEM RUPTURE

A rupture in the secondary system could occur either inside or outside the containment building. Both the reactor and the turbine plant would be shut down upon loss of pressure.

The most serious consequence from the standpoint of steam release would be an evaporator-shell rupture that propagated through both the shell and the head. Both primary and secondary water would flash to the containment building. Quick-closing valves in the secondary feedwater line and check valves in the steam line to each loop would prevent the water from more than one secondary loop from flashing to steam in the atmosphere of the

containment building. The resulting building pressure would be 23.8 psig, or 2.8 psi above design pressure. With a safety factor of 4 inherent in the vessel structural design, the 2.8-psi increase over design pressure would not result in containment-building failure.

5.3 LOSS OF TURBINE LOAD

A loss of turbine load from dumping of secondary-system steam causes the reactor power and the primary-system pressure to increase. At 975 psig, a "reactor pressure high" signal causes a four-rod scram. If the pressure increases to 1210 psig, a pressure-relief valve opens automatically and dumps primary steam to the emergency condenser. The opening of this valve generates a second four-rod-scram signal, which shuts the reactor down if the high-pressure scram signal failed to do so. Two backup safety-relief valves dump steam to the containment-building atmosphere at primary-system pressures of 1240 and 1250 psig, if all other safety circuits fail. The worst consequences to the reactor of a loss of turbine load would thus be a reactor-pressure rise to 1250 psig, followed by a pressure decrease as heat is removed by the emergency condenser.

5.4 WATER LEAKAGE

Primary-system water is the only water that contains radioactivity. Its activity is kept reasonably low by the purification system. The dominant radioactivity is that of the nitrogen-16 formed by the fast-neutron irradiation of oxygen-16. The short half-life of nitrogen-16 (7 sec) prevents accumulations of activity near leaks that occur around valve stems or control-rod thimbles.

The water leakage, estimated at less than 3 gal/day, is collected in drains and discharged to one of the 3000-gal retention tanks, whose contents are periodically monitored. The water may be discharged from these tanks to the condenser circulating-water discharge line through the service-water discharge line, but this is done only if such discharge will result in average yearly concentrations of radioactivity in the circulating water not above the allowable limits of 10 CFR 20. Thus, all water-leakage activity is contained within the building until discharge is permissible. The portable demineralizer unit is used to clean the water if the tank becomes filled while the activity level is still too high to permit discharge.

5.5 SHIELD-COOLING FAILURE

Shield-cooling failure may result from failure of the shield-cooling pump, obstruction of cooling coils, or rupture of cooling coils. An alarm notifies the operator when the shield-cooling water flow is too low; and the spare pump would be turned on. Water will be directed to a spare set of cooling coils embedded in the lead of the thermal shield if operation of the spare pump does not restore the water flow.

If the operator fails to use the spare pump or spare cooling coils, the concrete will rise to a maximum temperature of approximately 500 F. The concrete might crumble, but no hazard would result.

5.6 TOO MUCH OR TOO LITTLE WATER IN THE REACTOR

The reactor will be scrammed, and control-rod withdrawal prohibited, if the water level in the vessel is too high or too low.

A high water level in the vessel is not in itself hazardous, but it decreases the inherent safety of the reactor against reactivity increases. With a high water level, expulsion of water from the core becomes more difficult, and the negative coefficient of reactivity normally associated with void formation cannot act as readily to reduce excess reactivity. Large reactivity increases must be controlled by fuel-temperature rise. At operating temperatures, fuel-temperature rise can control reactivity increases of approximately 0.90 percent without fuel-tube melting. To increase reactivity by 0.90 percent would require sudden control-rod withdrawal, which is impossible during operation because the drives move at a constant, low speed. Failure of the rods to scram on a high water level signal would therefore not constitute a real hazard.

If a low vessel water level is not remedied, the core might eventually be uncovered and some fuel tubes might melt. The results of a complete loss of coolant are discussed in Appendix A. If control rods fail to scram upon low water level, the reactor operator may inject sodium pentaborate to shut down the reactor; it is doubtful whether any fuel tubes would melt in this situation. Furthermore, removal of water from the core would in itself shut down the reactor.

5.7 STARTUP ACCIDENT

The reactor-core neutron source is permanently installed in the core and kept at high strength by irradiation during full-power operation.

No more than one control rod at a time can be withdrawn from the core. The maximum rate at which the reactivity of the fully loaded cold core can be increased by control-rod withdrawal is approximately 0.13 percent/sec, which corresponds to withdrawal of the center rod near the midpoint of its travel. The operating neutron-flux level safety channels scram the reactor when the flux is 1.25 times the operating flux. If the control rods are withdrawn during startup (assuming no shutdown caused by void formation or temperature rise and assuming continuous reactivity increase at the maximum rate obtainable by rod withdrawal), the reactor period when the flux reaches 1.25 times the operating flux will be approximately 0.125 sec. Because the maximum delay in the scram circuit is 0.050 sec, scram will occur before the flux reaches a level that constitutes a hazard.

In actual startup, a 0.125-sec period is impossible. Control rods cannot be removed until the water temperature reaches 425 F; above this temperature, the negative void and temperature coefficients will automatically compensate for the reactivity increase caused by rod withdrawal. An additional circuit used during startup will scram the reactor when the period is less than 3 sec. This circuit will not be used during power operation because reactor flux perturbations from boiling would produce spurious short-period signals.

5.8 SUDDEN INTRODUCTION OF COLD WATER

At or near operating temperature, the temperature coefficient of reactivity is negative. The introduction of water that is cooler than that already in the reactor therefore increases reactivity. One means of injecting cooler water into the reactor while at full power is the sodium pentaborate injection system. Such injection lowers the vessel water temperature only by about 10 F and, with a temperature coefficient of reactivity of about -1.5×10^{-2} percent/°C, amounts to a total reactivity increase of only 0.08 percent. This is well below the maximum allowable reactivity increase that can cause meltdown. Thus, even if reactivity is not being decreased by 15 percent concurrently with the introduction of cooler water by the sodium pentaborate system, the addition of the cooler solution after boron precipitation is not hazardous.

The makeup system can fill the reactor vessel in somewhat over an hour. If core water is replaced by cold makeup water, the operation requires at least an hour. The total reactivity increase caused by cooling the water from 536 F to room temperature is 2 to 3 percent. Therefore, the maximum reactivity increase rate possible by adding cold makeup water is less than 0.001 percent/sec. This is 10 percent of the control-rod reactivity increase rate and therefore cannot be dangerous.

If a pipe rupture is followed by flashing of the core water to steam, the core water temperature can be lowered. The pressure drop in the vessel would cause a scram. Thus, the core cannot go critical even with a large temperature drop. If the rods fail to scram, continued power generation prevents a drop in the water temperature.

5.9 CHANGE IN POWER DEMAND

The Elk River facility will be operated as a base-load plant for the RCPA power system. The reactor power output should therefore be constant except for minor, normal fluctuations.

The most serious load change is a complete loss of load. In this case, the turbine inlet valve closes completely and secondary pressure immediately starts to rise. A hot-standby scram occurs when the turbine inlet valve closes. If primary and secondary pressures continue to rise, the operator dumps primary steam to the emergency condenser; this

dumping is done automatically when the primary pressure rises to 1210 psig. Continued rise of primary and secondary pressures eventually causes the primary and secondary relief valves to discharge steam to the atmosphere of the containment building and the outside atmosphere, respectively. Four-rod-scram signals are caused by high primary pressure (>975 psig) and high secondary pressure (>650 psig). Opening of the emergency condenser condensate flow-control valve also initiates a four-rod scram.

5.10 POWER FAILURE

If power from both electrical services is lost, the superheater auxiliaries trip, which causes a four-rod scram and places the reactor on hot standby. Because the battery power supply provides power to control-rod clutches and instruments, the remaining rods will not scram. A few seconds after the power interruption, the engine generator supplies power to Motor Control Center 3, which feeds all essential reactor auxiliaries, such as the shield-cooling pumps, the purification-system pumps, the air-monitor vacuum pump, and the control-rod motors.

Depending on the cause of the power interruption, the operator may elect to shut down the reactor by inserting the remaining rods.

5.11 INSTRUMENT FAILURE

If two out of three level-safety channels or both of the linear power channels fail during power operation, the reactor is automatically scrammed. If only one channel fails, an alarm is actuated; the faulty circuit may then be repaired. During startup, the failure of a single level-safety channel or linear power channel shuts down the reactor.

5.12 FUEL-TUBE FAILURE

During operation at 116 Mwt, the core will contain a total of 4100 fuel tubes. With this many tubes, there is a finite probability that small holes will form in the tubes, owing to corrosion or faulty fabrication. A tube defect would be followed by the release, over a relatively long period, of most of the gaseous and some of the solid fission products to the core water. All solid fission products would eventually be removed from the water by the purification system. The gaseous fission products would be removed from the water by the off-gas system and discharged through a charcoal adsorber to the containment-building exhaust duct, and then out the building stack.

If many tubes fail and the adsorber does not function properly, gaseous fission-product release to the atmosphere could exceed maximum permissible concentrations for continuous exposure. The iodine-131 isotope would yield the highest dose of any gaseous fission product released. The off-gases are released from a 96.5-ft-high stack. The maximum

iodine activity release rate from this height, which would give no more than the maximum concentration permissible in unrestricted areas at any point on the ground, is 1.28 microcuries/sec. Charcoal adsorption is expected to reduce iodine concentration by a factor of 1000 or greater. With this iodine reduction factor, pinholes could form in 200 tubes without releasing iodine activity at a rate exceeding 1.28 microcuries/sec.

The reduction factor for xenon-135 absorption by the charcoal bed is expected to be lower than that for iodine. A xenon-135 activity release of 4.3×10^{-4} curie/sec from the stack would give a maximum ground concentration equal to the maximum concentration permissible in unrestricted areas. Even with no absorption of xenon-135 by the charcoal bed, defects in 83 tubes would be required to give an activity release of 4.3×10^{-4} curie/sec.

If xenon-135 concentrations are reduced by a factor of 3 or greater by charcoal absorption, the maximum number of fuel tubes that may have pinholes is determined by the amount of iodine release. Accordingly, fission-product release to the atmosphere will not be hazardous if no more than 200 tubes at a time have pinhole leaks and if the charcoal absorption bed functions properly.

If all gaseous fission products in any one fuel tube suddenly escape into the primary coolant, the reactor will have to be shut down and the assembly containing the faulty tube will have to be removed. Both the particulate and gas monitors in the building exhaust vent will actuate alarms when activity release from the stack is excessive. The building ventilation dampers will also close when the alarm is actuated and thus stop further release to the atmosphere. The reactor will be manually shut down after the radiation alarms.

5.13 MELTING OF FUEL TUBES

If some means could be devised for increasing reactivity by an arbitrarily large amount, some of the fuel tubes could be made to melt. Unless melting were accompanied by a vessel or piping rupture, there would be no hazard to the surrounding area; all activity would be contained within the primary system and the off-gas system.

The doses resulting from an 8.62-percent meltdown following a rupture are presented in Appendix C. The doses for total melting with a simultaneous power excursion and rupture would be higher than those presented in Appendix C. The amount by which melting would exceed 8.62 percent depends on the magnitude of the power excursion. Doses for simultaneous power excursion and rupture were not calculated because this double accident is extremely improbable.

5.14 CONTROL-ROD FAILURE

The failure of a control rod to enter the core under scram conditions should not produce serious results. To shut down the reactor, the rods must reduce the reactivity only enough to compensate for the reactivity in the steam voids. The rods can control more than the excess reactivity at cold, clean startup. Thus, if one or even two rods fail to enter the core, the reactor can be shut down by the remainder.

If all thirteen control rods fail to enter the core, sodium pentaborate injection system can still shut down the reactor.

5.15 METAL-WATER REACTIONS

Metal-water reactions are extremely unlikely in the Elk River Reactor core. Zirconium is used only as a structural material, not as fuel cladding. When the vessel contains water, the zirconium shroud and the zirconium control-rod followers will be at the water temperature. During a loss-of-coolant accident, emergency cooling water will flow over the shroud and prevent high temperatures. To induce a zirconium-water reaction, the zirconium must be heated to over 2000 F in the presence of water. Because heat is not generated in the zirconium, it should not reach this temperature, and thus a metal-water reaction is not considered likely.

Even if a metal-water reaction does occur during a loss-of-coolant accident, the results will not be serious. The total reaction of the 1150 lb of zirconium in the core would raise the containment-building pressure only by an additional 1.3 psi.

5.16 SABOTAGE

A would-be saboteur would be posed with the problem of how to cause a sudden reactivity increase of more than 2.0 percent. Only by doing that could he hope to cause any major reactor damage. It would take the cooperation of the operating staff and time-consuming reactor modifications to cause such an increase, and therefore such an "accident" is not considered credible.

Considerable damage might be done with explosives. A carefully timed and placed series of explosions would be necessary. If rupture of the reactor vessel below the core and demolition of emergency-coolant feed lines were followed (after sufficient time to permit core meltdown) by an explosion that ruptured the containment building and dispersed melted fuel, there would be a considerable hazard to the surrounding area. However, it is not considered credible that a would-be saboteur could place the necessary explosive charges undetected.

5.17 FIRE

Because the reactor and containment building are fireproof, hazards to the surrounding area as a result of fire are not considered credible.

5.18 EARTHQUAKE

Only mild and infrequent earthquakes have been recorded in the Minneapolis area, and new earthquake areas seldom develop. The worst consequence of a mild earthquake would be control-rod jamming or piping rupture. The sodium pentaborate injection system will shut down the reactor if control rods jam, and the worst piping rupture is considered in Appendix A.

5.19 PROJECTILES

The possibility of a chemical explosion within the reactor is very small, owing to the low probability of a zirconium-water reaction (Sec. 5.1.5). A 25-percent metal-water reaction would release 6×10^7 calories, compared with a release of 80×10^7 calories for a 25-percent reaction in the EBWR. The potential energy of such an explosion is thus smaller than in the EBWR by a factor of 13. The reactor is surrounded by 5 to 8 ft. of concrete, so reactor-vessel fragments would have little chance of escaping. The concrete lining the containment building should be adequate protection against any projectiles that do escape from the reactor.

The top shield plug over the reactor weighs 124,300 lb and is an effective barrier to projectiles from the reactor-vessel cavity. Release of steam to the reactor cavity during a piping rupture is not expected to make a projectile of the top shield plug. For most conceivable ruptures, the rate of steam release to the cavity is so low that the shield plug would be displaced only enough to relieve steam pressure.

5.20 OPERATOR ERROR

The judicious selection and rigorous training of reactor operators should minimize the possibility of an accident from operator error. Two persons, at least one of whom is an AEC-licensed ERR operator, will be in the control room whenever the reactor is critical, whenever reactivity is changing or likely to change, or whenever reactor components are being modified while fuel is in the reactor. The selection of operators is based on the following requirements:

- (1) they must be 25 to 60 years old, in general good health, and with no record of diseases that could affect their ability to operate the reactor (such as epilepsy or a heart ailment);

- (2) they must be reliable, mature, emotionally stable, and well coordinated;

- (3) their education must include at least high school;
- (4) they must have shown aptitude in physics and mathematics while in school and interest in mechanics, engineering, and science; and
- (5) they should be experienced in the operation of conventional power plants and the maintenance of electrical instrumentation and mechanical systems (such experience is not essential).

The operator training program consists of a three-month lecture series, three months of training at the Argonne National Laboratory, and three months of training at the Elk River Reactor and its components. This training includes two phases: elementary and advanced. The elementary phase consists of a brief, simple review of reactor fundamentals, with emphasis on the differences between nuclear and conventional power plants. The advanced phase consists of instruction on the components and operation of the Elk River Reactor itself.

Regardless of the quality and training of operators, the possibility of operator error will always exist. The most probable error is the failure of an operator to take action when it is required. However, all situations that require immediate remedial action to prevent a serious accident cause an automatic scram. The conditions that cause a scram are listed in Sec. 3.2.5.

Operator action is required, however, if a control rod fails to enter the core after a scram signal. In this case the operator must decide whether sodium pentaborate solution should be injected. At a temperature not less than 425 F (approximately the temperature at which the reactivity available in the cold core is entirely offset), and with a clean core, the control rods can make the reactor 2 percent subcritical, even if the center rod fails to enter the core. Thus, the center rod and at least one other rod could fail, and the reactor would still be made subcritical at a temperature of 425 F or more. As the temperature decreased below 425 F, the reactor would eventually become critical if the center rod and one of the other rods of highest worth failed. Thus, for the hardly conceivable case of control-rod jamming, the operator would have to inject sodium pentaborate solution only after the reactor water temperature fell below 425 F. A delay in operator action that did not exceed the time required for cooldown to 425 F would therefore have no serious consequences.

Conditions that cause alarms are also listed in Sec. 3.2.5. Alarms indicate that a potentially dangerous situation exists, but not one that is immediately hazardous. The failure of an operator to take immediate action in such a case will not result in an accident, but the operator should nevertheless correct the condition as soon as possible.

5.21 LOSS OF BORON

The stainless-steel fuel tubes contain 600 ppm of natural boron. The center fuel tubes in the fuel assemblies can be replaced with stainless-steel tubes containing 1.2 to 1.5 percent boron (by weight). The loss of boron could cause a reactivity increase.

The reactor is initially loaded so that its excess reactivity in the presence of boron is 8.66 percent. The worth of the burnable poison is 8.09 percent in the initially operating reactor. This worth decreases as the boron-10 isotope is consumed. A hazardous situation can develop only if the rate of reactivity increase from boron burnup exceeds the reactivity decrease due to fuel burnup and fission-product buildup. Figure 13 shows that k_{eff} decreases with core life, so boron burnup will not create a hazard.

6. ELK RIVER REACTOR EVACUATION PROCEDURES

The Operations Manual (Vol. 1, Sec. E) contains detailed evacuation procedures for all on-site personnel involved in reactor startup and operation. The description of those procedures, which includes names and telephone numbers for working-hours or off-hours emergencies and describes specific job functions of key personnel, is only summarized here.

6.1 GENERAL EVACUATION PLANS

6.1.1 General Notification

This section presents notification procedures to be used in the event of an accident involving potential radiation exposure of personnel on or off the site.

When an accident occurs, the reactor shift instructor will notify the RCPA dispatcher (via the plant page network or dial telephone). The dispatcher will then immediately notify the nuclear plant manager and the radiological physics department manager.

These managers, who comprise the Emergency Evaluation Team, will meet in the reactor control room as soon as possible after notification, where they may:

(1) initiate the orderly shutdown of the plant and the evacuation of the conventional plant, and

(2) request the RCPA dispatcher to notify the following persons:

- AEC site representative,
- steam plant superintendent,
- instrumentations engineer,
- electrical apparatus superintendent,
- chief dispatcher,
- power system operations,
- conventional power department manager,
- planning and electrical department manager,
- general manager,
- electrical engineering section manager,
- education and publicity department manager, and
- nuclear contract and engineering department manager.

6.1.2. Control Center

All personnel listed in Sec. 6.1.1 will report as soon as possible after notification to the control center. The dispatcher will designate the location of the control center when notifying personnel. This location will be either in his office or at the RCPA radio communication tower (approximately 2 miles north of Elk River).

6.1.3 Additional Notification

The Radiological Physics Department manager may then notify any or all of the following, as required:

- University of Minnesota,
- State Board of Health,
- AEC Radiological Assistance Team, and
- medical aid.

6.1.4 Radiological Physics Equipment

The instruments listed below are at the site and are normally used for routine operation. Survey instruments are in the personnel decontamination room; laboratory instruments are in the Radiological Physics Laboratory in the headquarters building. Additional survey instruments for emergency use will be in the reactor control room, the dispatching office, and the headquarters building; other emergency instruments can be readily supplied by the University of Minnesota.

- Geiger-Mueller portable count-rate survey meters
- ionization-chamber portable survey meters
- stationary count-rate meters
- alpha portable survey instrument
- proportional counters
- Geiger-Mueller scaler system
- multi-channel pulse height analyzer
- high-volume air sampler
- low-volume air sampler
- pocket chambers
- external beta-gamma monitoring system

The hazard-evaluation personnel will be protected against radioactive contamination by self-contained breathing apparatus and protective clothing (coveralls, hoods, gloves, and shoe covers).

6.1.5 Criteria for Evacuation

If it appears that on-site personnel will be exposed to radiation doses in excess of 25 rem, or that the general public will be exposed to radiation doses in excess of 5 rem, evacuation procedures will be initiated.

6.1.6 Evacuation

Evacuation will begin on orders from the Emergency Evaluation Team. If the Team is unavailable, the shift instructor may order an evacuation of reactor personnel and a plant alert of all personnel. The criteria defined in Sec. 6.1.5 will be used to determine which of the following three evacuation plans is applicable:

Plan A: orderly plant shutdown, with plant personnel reporting to plant conference room after shutdown.

Plan B: orderly plant shutdown, with site evacuation; or

Plan C: orderly plant shutdown, with site and surrounding area evacuation.

6.1.6.1 Plan A. The Emergency Evaluation Team may order the shutdown of the plant and may order nonessential personnel to report to the plant conference room immediately. Personnel essential for plant shutdown will be equipped with respirators for use during shutdown and will report to the conference room as soon as shutdown is completed.

6.1.6.2 Plan B. In addition to completing procedures for Plan A, the RCPA steam plant, the headquarters building, the warehouse, Fluidyne Engineering Company, and the Elk River Municipal Sewage Disposal Plant will be alerted for evacuation. The individual in charge of evacuating these buildings will be notified by the radiological physics department manager (or his authorized delegate) in the control center to proceed with the building evacuation and will be advised of the precautionary steps to take and the direction in which to evacuate.

6.1.6.3 Plan C. In addition to completing procedures for Plans A and B, the surrounding area will be evacuated. If it appears that people outside the 1/4-mile exclusion area will be affected or that highway travel past the site will be affected, the sheriff of Sherburne County will be notified. He will be advised of the hazards and told what areas will be evacuated or restricted, and will call the State Highway Patrol, the sheriff of Wright County, the railroads, the local police, and local fire and civil defense groups, as necessary. He will then report to the control center. The responsibilities of these parties are outlined below.

State Highway Patrol:

Regulating traffic on state and federal highways passing the site, expediting the arrival of outside help, and evacuating all personnel from the roadside park near the RCPA property.

Sheriff of Wright County:

Evacuating rural families in his county, as necessary, and regulating traffic on county highways into Elk River.

Elk River Police and Fire Department:

Regulating local traffic, controlling evacuation routes, and controlling fires.

Elk River Civil Defense:

Notifying Elk River residents, evacuating them according to existing civil defense procedures, and supplying additional help for sheriff and local police.

Sheriff of Sherburne County:

Notifying the above agencies, regulating county highway traffic into Elk River, and stopping railroad traffic past the site.

Sheriff of Anoka County:

Evacuating rural families in his county, as necessary, and regulating traffic on county highways into Elk River.

6.2 EVACUATION OF SPECIFIC SITE BUILDINGS

6.2.1 Containment Building

Evacuation of the containment building may be started by any person who recognizes that conditions in the building make occupancy hazardous. The evacuation siren is actuated by pushing any of three buttons labeled "Evac Alarm." These buttons are in the fuel handling gallery, next to the interior airlock door, and in the reactor control room.

Upon hearing the evacuation siren, all personnel in the containment building will leave through the main airlock or the emergency airlock, whichever is nearest. When the reactor annunciator indicates that the evacuation siren has sounded, the reactor operator will announce this over the plant page network.

Evacuees will close the outer airlock door after leaving the airlock, report their names and the containment-building pressure to the control room via the plant telephones, and go to the personnel decontamination room. If the portal monitor and frisker monitor are reading on scale, normal decontamination procedure will be followed, if not, evacuees will go to the control room.

Evacuees will take portable radiological physics survey meters to the control room. Personnel monitors will be retained by evacuees and not returned to the badge board as in normal procedures.

Members of the operating crew will go to the control room; all other evacuees will go to the conference room.

6.2.2 Conventional Plant

In the event of a nuclear-plant accident that would affect the health and safety of conventional-plant personnel, the conventional plant will be evacuated after an orderly shutdown. The shutdown and evacuation will take place in two phases:

- (1) plant alert and nonessential-personnel evacuation, and
- (2) orderly shutdown and essential-personnel evacuation.

The reactor shift instructor may initiate a plant alert. When a plant alert is given, all personnel in the plant, except essential operators, will go immediately to the conference room. The essential operators are the turbine operator, the fireman, the assistant fireman, and the auxiliary operator. Visitors present in the plant will be directed by the tour guide to the conference room.

The essential operators will remain at their stations, where they will put on adsorber respirators and shut down the plant.

Plant shutdown may be ordered by a member of the emergency evaluation team after the alert.

The essential operators will go to the conference room as soon as plant shutdown is complete.

Personnel will remain in the conference room until further instructions are received from the emergency evaluation team or the control center. If evacuation criteria dictate that the conference room be vacated, instructions will be given as to a safe route and destination.

6.2.3 Headquarters Building

When the decision is made to evacuate the site buildings, the headquarters building receptionist will be notified from the control center and will immediately instruct personnel and visitors in the building (via the public address system) to go to room 132 or 133. The receptionist will remain at the switchboard.

Personnel and visitors will remain in rooms 132 and 133 until further instructions are received. If evacuation of these rooms is necessary, they will be advised of a safe route and destination.

6.2.4 Operations and Maintenance Building

When the decision is made to evacuate the site buildings, the dispatcher will be so notified from the control center. He will immediately designate an individual in the warehouse to notify all personnel in the building to congregate in the electrical shop.

All personnel will immediately close all doors, windows, and ventilation systems and go to the electrical shop when directed to do so by the designated individual. The dispatcher will remain in his office.

Personnel shall remain in the electrical shop until further instructions are received. If evacuation from this area is necessary, they will be advised of a safe route and destination.

6.2.5 Fluidyne Engineering Company

When the decision is made to evacuate the site buildings, personnel at the Fluidyne Engineering Company building will be notified from the control center. They will be told of the radiological hazards and advised of what action to take, including a safe route and destination.

6.2.6 Elk River Municipal Sewage Disposal Plant

When the decision is made to evacuate the site buildings, the personnel at the Elk River Sewage Disposal Plant will be notified from the control center. (This plant is not continuously staffed and requires only periodic maintenance.) They will be told of the radiological hazards and advised of proper action and a safe route and destination.

6.2.7 Warehouse and Garage

When the decision is made to evacuate the site buildings, personnel at the Fluidyne Engineering Company building will be notified from the control center. They will be told of the radiological hazards and advised of proper action and a safe route and destination.

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1. FOR ASSEMBLY PROCEDURE SEE
PREOPERATIONAL TESTING MANUAL (APPENDIX B)

GENERAL ASSEMBLY
PLAN VIEW
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NOZZLE SCHEDULE

CODE	DESCRIPTION	NO.	NOM. SIZE
I	FORCED CIRCUL INLET	2	16"
II	LIQUID LEVEL	1	1 1/2"
III	LIQUID LEVEL	1	1 1/2"
IV	FORCED CIRCUL. OUTLET	2	16"
V	EMERGENCY COOLING INLET	1	1 1/2"
VI	STEAM OUTLET	2	10"
VII	VENT LINE	1	1"
VIII	HI-PRESS. POISON INLET	1	2 1/2"
IX	CONTROL ROD	13	4"
X	REACTOR ACCESS	4	12"
XI	FEED WATER INLET	2	8"

SECTION A-A

NOTES

1. DIM. MARKED * SHOW CLEARANCE BETWEEN NOZZLE O.D. & SLEEVE I.D. OF THERMAL SHIELD AT ROOM TEMP. DIMENSIONS ARE GIVEN AS GUIDES TO BE USED WHEN INSTALLING THERMAL SHIELD & SHIMMING. (SEE DETAIL Y)

Z VENT-LINE NOZZLE (NOZZLE VII) HAS BEEN PLUGGED.

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ELEVATION THROUGH REACTOR
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SCALE: 1"=1'-0"

DETAIL Y
SCALE: 3"=1'-0"

DETAIL V
TYP. 8 PLACES
SCALE: 3"=1'-0"

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SCALE: 3"=1'-0"

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DETAIL U
SCALE: 3"=1'-0"
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SECTION T-T
SCALE: 3"=1'-0"

1. FILL WELL WITH FLUX
2. HEAT WELLS WHILE FEEDING SILVER SOLDER ALLOY TO FILL WELL
3. DIP THERMOCOUPLES IN FLUX & PREHEAT SLIGHTLY
4. INSERT THERMOCOUPLES IN WELL WHILE MAINTAINING SOLDER MOLTEN UNTIL THERMOCOUPLES ARE SEATED AT BOTTOM OF WELLS & SURFACES OF SOLDER BECOMES QUIESCENT
5. WASH ACIDIFIED ZINC CHLORIDE FLUX & SOLDER RESIDUES OFF WITH HOT WATER TO PREVENT CHLORIDE CORROSION ON STEEL.

TYP. INSTALLATION OF THERMOCOUPLES

VIEW E-E
CONTROL ROD DRIVE ARRANGEMENT

GENERAL ASSEMBLY
SECTION E-E
CONTROL ROD DRIVE ARRANGEMENT
Sheet 6 of 7
Revision C
DWG. 12D-01000

SECTION F-F

GENERAL ASSEMBLY
SECTION F-F
CONTROL ROD DRIVE ARRANGEMENT
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Revision C
DWG. 12D-01000

ELK RIVER SITE EXCLUSION AREA
Revision B
DWG. 12E-01100

ORIGINAL DWG NO. 120-0120

PROPERTY PLAT
Revision A
Dwg. 12E-01500

MAIN FLOOR PLAN ELEV. 938'-0"

PLAN AT ELEV. 977'-7 1/2"

GENERAL ARRANGEMENT PLAN
MAIN FLOOR
Revision B
DWG. 12E-01501

INTERMEDIATE FLOOR PLAN ELEV. 919'-0"

GENERAL ARRANGEMENT PLAN
 INTERMEDIATE FLOOR
 Revision B
 DWG. 12E-01502

GENERAL CROSS SECTION "B-B"

REACTOR VESSEL AND
INTERNAL COMPONENTS ASSEMBLY
Sheet 2 of 2
Revision D
DWG. 12D-04000

REACTOR AND STORAGE POOL SHIELDING
Revision B
DWG. 12D-06500

FUEL TRANSFER PROCEDURE

1. PREPARE FOR FUEL TRANSFER BY:
 - (a) RAISE WATER LEVEL IN REACTOR VESSEL TO EL. 927'-5"
 - (b) REMOVE TOP SHIELDING PLUGS
 - (c) REMOVE CANAL SHIELDING PLUG
 - (d) REMOVE REACTOR VESSEL HEAD & HEAD INSULATION ASSY
 - (e) REMOVE CANAL GATE
 - (f) RAISE WATER IN STORAGE WELL & REACTOR CAVITY TO EL. 947'-0"
2. REMOVE ELEMENT FROM CORE BY:
 - (a) POSITION BRIDGE APPROX. 6" BEHIND ELEMENT TO BE REMOVED
 - (b) SWING JIB CRANE OVER ELEMENT
 - (c) LOWER FUEL TRANSFER TOOL IN ELEMENT WITH HAND WINCH
 - (d) LOCK LATCH
 - (e) RAISE ELEMENT FROM CORE
3. TRANSFER ELEMENT TO STORAGE WELL VIA CANAL BY:
 - (a) SWING JIB CRANE TO CENTER
 - (b) MOVE BRIDGE TO STORAGE WELL SIDE
4. PLACE ELEMENT IN STORAGE RACK BY:
 - (a) POSITION BRIDGE APPROX. 6" BEHIND DESIRED STORAGE POSITION
 - (b) SWING JIB CRANE OVER DESIRED STORAGE POSITION
 - (c) LOWER ELEMENT & REST IN STORAGE RACK
 - (d) RELEASE LATCH
5. REPEAT 2, 3 & 4 UNTIL CORE IS REFUELED
6. PREPARE FOR REACTOR OPERATION BY:
 - (a) LOWER WATER LEVEL IN STORAGE WELL TO EL. 932'-1 1/2" & WATER LEVEL IN REACTOR CAVITY TO EL. 927'-5"
 - (b) INSTALL CANAL GATE
 - (c) INSTALL VESSEL HEAD & HEAD INSULATION ASSY.
 - (d) INSTALL CANAL SHIELDING PLUG
 - (e) INSTALL TOP SHIELDING PLUGS
 - (f) LOWER WATER IN REACTOR VESSEL TO OPERATING LEVEL

2 3 4 LOCATE HOLES FOR 1/2" BOLTS FROM "ARM, UNDER WATER LIGHT" (12D-08010) 24 PLACES SEE LOCATION BELOW

NOTES:

1. TO REMOVE A DUMMY FUEL ELEMENT FROM THE CORE OR STORAGE WELL ATTACH "PLUNGER" (12D-08005) TO "FUEL TRANSFER TOOL"
2. TO REMOVE THE START-UP SOURCE FROM THE CORE OR STORAGE WELL ATTACH "START-UP SOURCE TOOL" (12D-08011) TO THE FUEL TRANSFER TOOL

FUEL TRANSFER SYSTEM
STORAGE POSITIONS OF MISCELLANEOUS
EQUIPMENT DURING FUEL TRANSFER
Sheet 3 of 4
Revision D
DWG. 12D-08000

REACTOR ENCLOSURE PLAN AND SECTION
Revision K
DWG. 12E-11003

NO.	DESCRIPTION	LOCATION	REMARKS
1	1000	1000	1000
2	1000	1000	1000
3	1000	1000	1000
4	1000	1000	1000
5	1000	1000	1000
6	1000	1000	1000
7	1000	1000	1000
8	1000	1000	1000
9	1000	1000	1000
10	1000	1000	1000
11	1000	1000	1000
12	1000	1000	1000
13	1000	1000	1000
14	1000	1000	1000
15	1000	1000	1000
16	1000	1000	1000
17	1000	1000	1000
18	1000	1000	1000
19	1000	1000	1000
20	1000	1000	1000
21	1000	1000	1000
22	1000	1000	1000
23	1000	1000	1000
24	1000	1000	1000
25	1000	1000	1000
26	1000	1000	1000
27	1000	1000	1000
28	1000	1000	1000
29	1000	1000	1000
30	1000	1000	1000
31	1000	1000	1000
32	1000	1000	1000
33	1000	1000	1000
34	1000	1000	1000
35	1000	1000	1000
36	1000	1000	1000
37	1000	1000	1000
38	1000	1000	1000
39	1000	1000	1000
40	1000	1000	1000
41	1000	1000	1000
42	1000	1000	1000
43	1000	1000	1000
44	1000	1000	1000
45	1000	1000	1000
46	1000	1000	1000
47	1000	1000	1000
48	1000	1000	1000
49	1000	1000	1000
50	1000	1000	1000
51	1000	1000	1000
52	1000	1000	1000
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58	1000	1000	1000
59	1000	1000	1000
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62	1000	1000	1000
63	1000	1000	1000
64	1000	1000	1000
65	1000	1000	1000
66	1000	1000	1000
67	1000	1000	1000
68	1000	1000	1000
69	1000	1000	1000
70	1000	1000	1000
71	1000	1000	1000
72	1000	1000	1000
73	1000	1000	1000
74	1000	1000	1000
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76	1000	1000	1000
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78	1000	1000	1000
79	1000	1000	1000
80	1000	1000	1000
81	1000	1000	1000
82	1000	1000	1000
83	1000	1000	1000
84	1000	1000	1000
85	1000	1000	1000
86	1000	1000	1000
87	1000	1000	1000
88	1000	1000	1000
89	1000	1000	1000
90	1000	1000	1000
91	1000	1000	1000
92	1000	1000	1000
93	1000	1000	1000
94	1000	1000	1000
95	1000	1000	1000
96	1000	1000	1000
97	1000	1000	1000
98	1000	1000	1000
99	1000	1000	1000
100	1000	1000	1000

[illegible]

2. LOCATIONS ARE SHOWN IN LOWER RIGHT HAND CORNER OF EACH BLOCK:
 PD - PULSE SUPPLY
 SR - STARTUP SIGNAL
 CR - CURRENT SIGNAL
 1. CAPACITORS REQUIRED FOR THE SYSTEM ARE SHOWN ADJACENT TO EACH BLOCK.

NOTE:

APPENDIX A

MAXIMUM CREDIBLE ACCIDENT

A1. SUMMARY

The maximum credible accident is assumed to be a rupture that (1) completely opens one of the 16-in. forced-circulation lines at a point outside the biological shield and (2) occurs after a significant period of operation at 116 Mwt. Ten seconds after the rupture, all water will have been expelled from the reactor vessel. It is assumed that emergency cooling water from the 30,000-gal overhead storage tank starts flowing into the vessel no more than 6.83 min after the rupture, the time at which the stainless steel of the fuel tubes in the central portion of the core starts to melt.

With these assumptions, the 8.62 percent of fuel tubes might eventually melt. Since data on the release of fission products from thorium-uranium fuels are limited, it is assumed that 100 percent of the noble gases and halogens and 5 percent of the strontium* contained in a fuel tube are released when the tube itself melts. This corresponds to the release of 8.62 percent of all volatile fission products and 0.431 percent of all strontium in the core at the time of rupture.

The containment-building pressure and temperature variations resulting from the accident are discussed in Appendix B. Fission-product leak rates from the containment building, and the resulting doses to the surrounding population, are discussed in Appendix C.

A2. OPERATION OF EMERGENCY COOLING SYSTEM AFTER THE MAXIMUM CREDIBLE ACCIDENT

A2.1 Controls

The following events take place after a large rupture: A four-rod scram occurs when the water level is 3 ft 2 in. above the core. At this point, the solenoid valves that permit high-pressure cooling water to flow through the spray ring are automatically opened, and 12-gpm flow begins. A flow of 25 gpm (20 gpm minimum) from the overhead storage tank will also enter the reactor vessel when the difference between the containment-building pressure and the reactor-vessel pressure is 26 psi or less. A check valve prevents back flow. If the reactor-vessel pressure is low and cooling water fails to enter the core, an alarm will alert the operator, and steps can be taken to provide backup cooling water. A total of 25 gpm of makeup water can be introduced into the vessel via the spray ring from the service-water system.

A2.2 Overhead Storage Tank

The overhead storage tank contains 30,000 gal of demineralized water, 15,000 gal of which are available for emergency core cooling. This water flows by gravity to the spray ring. The bottom of the storage tank is approximately 60 ft above the spray ring. The pressure within the storage tank is the same as the containment-building pressure at

*Creek, G.E., et al., "Experiments on the Release of Fission Products from Molten Reactor Fuels" (ORNL-2616).

all times, so a minimum head of 60 ft of water (26 psi) is always available to drive the cooling water. The system consists of approximately 75 ft of 1.25-in. Schedule-40 piping, two globe valves, and two check valves. When all pressure drops (including the pressure drop in the spray ring) are accounted for, the system delivers at least 25 gpm of 70 F cooling water. The initial flow rate will be slightly higher, since the water level in the tank is initially more than 60 ft above the spray ring.

A2.3 Spray Ring

The spray ring is 6 ft in diameter and 3 ft above the top of the shroud. Water is directed at the shroud by 24 nozzles equally spaced about the ring. The nozzles are of stainless steel; the spray leaves each nozzle as a 50 deg cone, with uniform water flow across the base of the cone.

The spray distribution in the shroud sections was computed for the optimum nozzle setting; the computed results were confirmed by tests. Each quadrant of the core contains eleven shroud sections, seven of which have a different flow input. These are labeled A, B, C, D, E, F, & G in Dwg. 12D-04000. Table A-1 lists the fraction of the total water flow that enters each shroud section. The flow rates are based on a total flow of 25 gpm.

TABLE A-1

FLOW DISTRIBUTION TO TYPICAL SHROUD SECTIONS

<u>shroud section</u>	<u>fraction of total flow</u>	<u>flow rate (gpm)</u>
A	0.0351	0.878
B	0.0325	0.813
C	0.0295	0.738
D	0.0255	0.638
E	0.0200	0.500
F	0.0054	0.135
G	0.0063	0.158

A3. ASSUMPTIONS

A3.1 Assumptions Relating to Size and Location of Break

One of the major decisions to be made in analyzing a loss-of-coolant accident is the size and location of the rupture. A sudden rupture of the vessel wall because of material defects was deemed impossible. Vessel technology and reactor-vessel standards are so

rigorous that a vessel rupture without a large overpressure is hardly credible. A below-the-core rupture would have more serious consequences than a rupture above the core in a loss-of-coolant accident. It was decided, therefore, to determine the largest rupture possible below the core. Below the core, piping consists of thirteen 4-in. control-rod thimbles and two 16-in. forced-circulation nozzles (which are capped off). The probability of rupture increases as pipe size decreases, but in order to analyze the worst possible case it was decided to assume a complete rupture of one of the 16-in. forced-circulation lines at a point outside the biological shield. No account, therefore, can be made for flooding of the reactor cavity in the biological shield. With the vessel at an initial pressure of 1200 psig, it was estimated that a minimum of 10 sec would be required to expel all water from the vessel.

A3.2 Assumptions Relating to Initial Fuel-Tube Temperatures

After a rupture in a 16-in. forced-circulation line, water rushes from the vessel. The water remaining in the vessel boils at a rate that varies inversely with vessel pressure. The rate of heat removal during blowdown would probably be sufficient to cool the fuel tubes nearly to the final temperature of the water (217 F). It was assumed, however, that only enough heat is removed to reduce the fuel-tube surface temperature to the water temperature at full-power operation (about 550 F).

If the temperature of each fuel tube were 2000 F lower than the average temperature of the pellets within the tube, 1.5×10^6 Btu would have to be removed to bring all tubes to a uniform temperature. This amount of heat would be removed by boiling 7 percent of the water initially present in the vessel. During 10 sec of normal operation, approximately 4.5 percent of the vessel water boils. With the increasing boiling rate during blowdown, it seems safe to assume that at least 7 percent of the vessel water would boil.

A3.3 Assumptions Relating to Fuel-Tube Temperature Rise

During the accident, fuel-tube temperatures rise owing to decay-heat generation. It was assumed that temperatures are uniform across the tubes during this transient rise. This is not strictly true, but it is a conservative assumption. Actual temperatures would be highest at the pellet centerline and lowest in the stainless steel. The criterion for fission-product release is melting of the stainless steel. A uniform temperature distribution would give higher cladding temperature, or greater fission-product release, than the actual temperature distribution.

A3.4 Assumptions Relating to Cooling Mechanisms

Water from the spray nozzles is directed at the top of the shroud sections. If the water were finely atomized, it would probably fall uniformly through the core and produce uniform cooling. It is not certain that this will be the case. If the spray consisted of

heavy droplets, all the water would hit the shroud sections and then flow into the core. Most of the water would probably flow down the shroud surfaces, and the remainder would splash and fall directly on the fuel assemblies. It was conservatively assumed that all the cooling water flows down the shroud surfaces and cools directly only the fuel tubes adjacent to the shroud and the shroud itself.

It was assumed that the inner fuel tubes are cooled only by radiation. Thus, to cool an inner row of tubes, heat must be absorbed and re-radiated by each successive surrounding row of pins. No credit is taken for heat removal by steam convection. Water evaporated by radiant heat will produce steam, which, in turn, would flow downward through the core and remove additional heat. Also, no credit is taken for heat radiated to the steam atmosphere. All heat absorbed by the steam would be removed by the flow of steam. Thus the cooling-mechanism assumptions are very conservative.

A3.5 Assumptions Relating to Spatial Variation of Decay Heat Generation

The decay heat was assumed to have the same radial variation as the neutron flux in the clean operating reactor. This is a somewhat conservative assumption. The buildup of fission products and depletion of fuel with time tends to flatten the flux distribution and, to an even greater extent, flatten the power distribution.

The rate of decay heat generation was assumed to be constant in the axial direction. For the calculation, this constant value is the maximum axial value of heat generation for a fuel tube. This assumption was made in order to make the results as conservative as possible.

A3.6 General Assumption

The general assumption that double accidents will not occur has been made. Such a double accident would be a simultaneous rupture and failure of the control rods to enter the core after a scram signal.

A3.7 Summary of Assumptions

The foregoing assumptions may be summarized as follows:

- (1) Fuel-tube temperature 10 sec after the rupture are constant at 550 F.
- (2) The only means of cooling the fuel tubes in any shroud section is by radiation to the outermost row of tubes, which is at the same temperature as the cooling water.
- (3) The cooling water starts to flow sometime within the first 7 min after the rupture, but heat is not transferred by radiation until the pins are at a temperature just below the melting point of stainless steel (2550 F).

(4) The variation of decay heat with time is based on the Way-Wigner formula for a reactor operating for an infinite time at 116 Mw. The radial variation of decay heat is the same as the initial hot operating flux distribution, whereas decay heat generation is constant axially at the maximum axial value.

(5) The decay-heat generation rate is constant for fuel tubes within a given shroud section and equal to the heat generation of the central fuel tube.

A4. ACCIDENT ANALYSIS AND RESULTS

Ten seconds after the rupture all water has left the vessel and the temperature of the fuel tube is 550 F. Assuming no cooling water and no radiative heat transfer, decay-heat generation causes the fuel-tube temperatures to rise. If this condition continues indefinitely, all of the fuel tubes would eventually melt. With cooling water flow, however, enough heat can be removed by radiation to the cooled shrouds to prevent melting most of the tubes.

It was assumed that, even with the cooling-water flow, heat is not transferred by radiation from the fuel tubes until the tubes in any given shroud section have reached a temperature just below the melting point of stainless steel (2550 F). However, with the assumption of a constant heat generation per tube within each shroud section, all tubes within a given shroud section will reach this temperature at the same time. Since the decay heat varies radially and is also a function of time, the tubes in different shroud sections will reach this temperature at different times. The tubes in shroud sections A would reach a temperature just below 2550 F within a few seconds after the rupture. The tubes in sections B through G would require 485, 508, 720, 890, 1550, and 1190 sec. respectively, to reach this temperature. The average heat generation rate per tube at the time the tube material reaches a temperature of 2550 F is different in each shroud section because of the radial variation of decay heat. The decay heat generation rate is computed by the Way-Wigner formula. The average heat generation rate per tube is given in Table A-3.

When a temperature close to 2550 F is reached in each shroud section, it is assumed that the tubes begin to radiate heat. The tubes within a typical shroud section form a 10 x 10 square array. For purposes of analysis, this array is considered to be a series of coaxial, isothermal square cylinders. Since fuel-tube temperatures were assumed to rise adiabatically, all tubes (or square cylinders) in a given shroud section would be at the same temperature (just under 2550 F) initially. As soon as radiative heat transfer starts, however, the temperature of some of the cylinders in a given shroud section will start to rise above 2550 F and the temperature of other cylinders will start to fall. The direction of the temperature change will depend on the steady-state temperature distribution obtained by assuming a constant heat-generation rate per tube equal to the value of heat generation at the time radiation begins. In other words, if the steady-state temperature distribution within shroud section A for a heat-generation rate of 1.052 Btu/sec per tube was such that the four center cylinders had to have a temperature above 2550 F to transfer heat at that rate by

radiation, whereas the outer cylinder required a temperature below 2550 F to transfer heat at the same rate, the tubes of the four center cylinders would presumably melt, while those of the outer cylinders would be adequately cooled.

The analysis involved determining the steady-state temperatures within a shroud section for the constant heat-generation rates listed in Table 4-3. If the steady-state temperature of a cylinder was above 2550 F, the temperature at the start of cooling would start to increase. For this case, the cylinder is assumed to melt. Conversely, a steady-state temperature below 2550 F would mean that the cylinder would not melt. Since the heat-generation rate decreases with time, temperatures should decrease with time after this point. The steady-state temperatures in shroud sections A, B, C, D, E, F, & G are shown in Table A-4. The cylinders are numbered 2 through 5, where cylinder 5 consists of the four tubes at the center of a shroud section and cylinders 4, 3, and 2 are the coaxial outer cylinders. It has been assumed that the row of tubes nearest the shrouds is cooled directly by water and hence is at the temperature of the water. Table A-4 shows that all cylinders above 2550 F steady-state temperature will melt. The total number of melted tubes is 8.62 percent of the total number of tubes (see Table A-5).

It should be emphasized that the temperatures shown in Table A-4 will never actually be reached since the heat generation rate per tube decreases with time. These temperatures were determined so as to indicate the direction of temperature change in those tubes that are initially just under 2550 F. It should also be emphasized that the assumption that there is no cooling by radiation before a temperature just below 2550 F is reached is pessimistic. Cooling water will start to flow less than 7 min after a rupture, and hence the initial temperature rise will not be adiabatic.

It has been tacitly assumed that the heat radiated to the outer row of tubes could be adequately removed by the cooling water. An analysis of the cooling capacity of the water was made to confirm this assumption. Cooling water enters the core at 70 F and is converted to steam at the saturation temperature in the containment building after the rupture (217 F). The total heat-removal capacity per section can be determined from the water flow to each shroud section in Table A-1. The total heat-generation rate per shroud section as a function of time is listed in Table A-2, from which it may be seen that the heat-removal capacity of the cooling water equals or exceeds the heat-generation rate within all shroud sections, long before the tubes in any shroud section could reach the melting point under adiabatic conditions.

A4.1 Heat Generation and Removal per Shroud Section

The fission-product decay-heat generation rate per tube can be expressed as a function of radial position and time. The time variation is obtained from the Way-Wigner formula for decay power:

$$P = 6600 t^{-0.2}$$

where,

- P = total decay-heat generation rate for 116-Mw operation, Btu/sec
- t = time after shutdown, sec

Or, the average power per tube is:

$$\bar{p} = 1.615 t^{-0.2}$$

where,

- $\bar{p}$ = average decay-heat generation rate of 3900 tubes, Btu/sec
- t = time after shutdown, sec

The decay-heat generation rate is assumed to vary initially as the core flux in the hot operating condition. The radial variation of flux can be described approximately by the equation:

$$\phi = 1.57 \bar{\phi} J_0 \frac{1.926 r}{R_0}$$

where,

- ϕ = flux at radial distance r from core centerline
- $\bar{\phi}$ = average core flux
- R_0 = core radius
- r = radial distance

Assuming a constant axial flux equal to the maximum axial flux, the heat-generation rate per tube in Btu/sec is:

$$\bar{p} = 3.64 t^{-0.2} J_0 \frac{1.926 r}{R_0} \quad (A-1)$$

A4.2 Adiabatic Temperature Rise

The decay-heat distribution is described by Equation A-1. The thermal capacitance of the stainless-steel-thoria-urania fuel tubes is 0.256 Btu/°F. The temperature rise in a fuel tube at time t after shutdown, assuming an adiabatic temperature rise, is:

$$\Delta T = \frac{3.64}{0.256} J_o \frac{1.826 r}{R_o} \int_{10}^t t^{-0.2} dt \quad (A-2)$$

$$= 17.77 J_o \frac{1.826 r}{R_o} (t^{0.8} - 6.3) \text{ } ^\circ\text{F}$$

where $\Delta T = T_t - T_{\text{initial}} = T_t - 550 \text{ F.}$

For a ΔT of (2550-550) F, time t can be determined as a function of position. This t is then used in Equation A-1 to find the heat-generation rate at time t . The results for each shroud section are presented in Table A-3.

TABLE A-2
HEAT GENERATION AND REMOVAL RATES

shroud section	heat-generation rate per shroud section* as a function of time, Btu/sec	rate of heat removal by cooling water, Btu/sec	time required for heat removal rate to equal or exceed heat- generation rate, sec
A	$246 \times t^{-0.2}$	126	28.3
B	$213 \times t^{-0.2}$	117	20.0
C	$183 \times t^{-0.2}$	106	15.3
D	$154 \times t^{-0.2}$	92	13.2
E	$128 \times t^{-0.2}$	72	17.8
F	$61 \times t^{-0.2}$	19.4	302.0
G	$51 \times t^{-0.2}$	22.8	53.0

*Based on the average flux within each shroud section.

TABLE A-3

HEAT-GENERATION RATES WHEN TUBES OF A SHROUD SECTION
HAVE REACHED 2550 F

<u>shroud section</u>	<u>time to reach 2550 F, sec</u>	<u>heat-generation rate at this time, (Btu/sec)/tube</u>
A	410	1.052
B	485	0.883
C	508	0.838
D	720	0.588
E	890	0.473
F	1550	0.269
G	1190	0.352

A4.3 Steady-State Temperature Distribution

Shroud sections A, B, C, D, and E are almost identical geometrically. Sections F and G have a different geometry, but it can be shown that, if they meet the requirements for non-melting with a geometry assumed similar to the other sections, no portion of them will melt.

The equation for radiative heat transfer is:

$$q''_n = A_n \frac{\sigma}{\left(\frac{1}{\epsilon_n} + \frac{1}{\epsilon_{n-1}} - 1\right)} (T_n^4 - T_{n-1}^4) \quad (A-3)$$

where,

q''_n = heat radiated from the $\underline{n}$ -th row of tubes to the $\underline{(n-1)}$ th row of tubes,
Btu/hr

n denotes the row of tubes, is equal to 1 for the outside row, and increases to 5 for the four central tubes nearest the center

A_n = area of n -th row, in.²
 σ = Boltzmann constant = 0.1731×10^{-8} Btu/ft²-hr-°R⁴
 ϵ_n = emissivity of n -th surface = 0.5 for stainless steel
 T_n = temperature of n -th row, °R

Assuming a heat-generation rate per tube of q' Btu/hr, the total heat generated in the tubes of the n -th row and all inner rows is $= (12 - 2n)^2 \times q'$. Then:

$$T_n^4 = T_{n-1}^4 + \frac{\left(\frac{1}{\epsilon_n} + \frac{1}{\epsilon_{n-1}}\right) (12 - 2n)^2 q'}{\sigma A_n}$$

or,

$$T_n^4 = T_n^4 + \frac{q'}{5.71 \times 10^{-10}} \sum_{k=2}^n \frac{(12 - 2k)^2}{A_k} \quad (A-4)$$

Equation A-4 is then used to find the steady-state temperature of cylinders 2 through 5, where cylinder 2 is the second cylinder from the outside and cylinder 5 is the cylinder nearest the center. It is assumed that cylinder 1 is at the temperature of the cooling water (217 F). The values of q' used in each shroud section are the values of heat-generation rate per tube when the tubes within a shroud section have reached 2550 F.

TABLE A-4
STEADY-STATE TEMPERATURES

shroud section	fuel-tube cylinder no.			
	2	3	4	5
A	2180 F	2590 F	2820 F	2960 F
B	-	2460 F	2680 F	2810 F
C	-	-	2640 F	2770 F
D	-	-	2370 F	2500 F
E	-	-	-	-
F	-	-	-	-
G	-	-	-	-

(Melting temperature of stainless steel: 2550 F)

A4.4 Meltdown Fraction

The total numbers of melted tubes in each shroud section, computed from the data in Table A-4, are listed in Table A-5.

TABLE A-5
NUMBER AND LOCATION OF MELTED FUEL TUBES

<u>shroud section</u>	<u>cylinders melted</u>	<u>number of tubes melted in each section</u>	<u>number of sections of same type in core</u>	<u>total number of tubes melted</u>
A	3, 4, 5	36	4	144
B	4, 5	16	8	128
C	4, 5	16	4	64
D	-	-	-	-
E	-	-	-	-
F	-	-	-	-
G	-	-	-	-
total				<u>336</u>

The total number of fuel tubes in the core is 3900. Therefore the meltdown fraction is 336/3900 or 8.62 percent.

APPENDIX B

CONTAINMENT DESIGN

The containment-building design pressure (21 psig) was determined by calculating the building pressure rise resulting from rupture of the primary system. It was assumed that all primary-system water flashes to steam instantaneously and that no heat is lost to the outside air or stored in the concrete lining of the containment building or in other materials. These assumptions are conservative, because no conceivable primary-system rupture would permit instantaneous flashing of all primary-system water and because the steam will inevitably lose heat to internal components and to the outside air. It was also assumed that the released steam and containment-building air mix thoroughly, so that the heat capacitance of the air is accounted for. On this basis the containment-building pressure rise is 20.2 psig. Operation was assumed to be at 116 Mwt, with four evaporators, and at the highest contemplated operating pressure (1168 psia). Under these conditions, the primary system contains 25,611 lb of water and 2168 lb of steam. This amounts to a total energy content of 17.09×10^6 Btu. Releasing this quantity of energy to the containment building results in a final equilibrium pressure of 20.2 psig.

The energy release from a chemical reaction between zirconium and water was not considered in calculating pressure buildup within the containment building. Zirconium is used only as a structural material in the shrouds, spacers, and control-rod followers.

The total amount of zirconium in the core is only 1150 lb. If all the zirconium reacted with water the energy release would be 0.96×10^6 Btu, with an increase in containment-building pressure of 1.3 psi. Reaction of 25 percent of the zirconium would raise the containment-building pressure only 0.5 psi. Even if included, a zirconium-water reaction would not affect the design of the containment building. However it is doubtful whether a zirconium-water reaction would actually occur during a loss-of-coolant accident. This zirconium must be melted and, probably, finely subdivided before a violent reaction can take place. Since zirconium is not an integral part of the fuel elements, it would have to be heated by radiation from the fuel elements.

Most of the cooling water from the spray ring will run down the shrouds, as shown in Appendix A. This water flow is sufficient to remove all heat radiated from the fuel pins. The shrouds will therefore not melt and a zirconium-water reaction should not occur.

The assumption of a zirconium-water reaction accompanied by a hydrogen reaction would be even more pessimistic than the assumption of a zirconium-water reaction alone. At atmospheric pressure with no steam present and at a temperature of 230 F, at least 4 percent hydrogen by volume must be present in the containment-building air even to propagate an upward flame.¹ The hydrogen content must be at least 20 percent for an explosive reaction. The presence of steam and lower temperatures would increase these limits. When air contains more than 32 percent steam, by volume, combustion cannot be supported with any mixture of hydrogen. This limitation on steam content would exclude any possibility of a hydrogen reaction within the containment building. If all

¹/WAPD-SC-545

the zirconium reacted with water, a total of 25.5 lb-moles of hydrogen would be produced. The free volume in the containment building would be occupied by a mixture of hydrogen (1.9 percent), steam (48.8 percent), and air (49.3 percent). Thus, the hydrogen reaction could not occur.

If one of the evaporator shells ruptures, the primary-system water and a portion of the secondary-system water could flash to steam. This would be the case for a large crack that penetrated both the head and shell of the steam generator. Quick-closing valves in the secondary feedwater lines and check valves in the secondary steam lines from the evaporators would allow no more than the water (4200 lb) and the steam (560 lb) in the secondary side of one evaporator to flash into the containment building. This would cause an increase of 3.5 psig in the containment-building pressure for a total pressure of 23.8 psig, as compared to the design pressure of 21 psig. The design of the evaporators and subcoolers is such that it is difficult to conceive of a crack propagating through both the shell and head. Such a failure would still not endanger the integrity of the containment building, however, because the building has been designed according to the ASME Code, which requires a safety factor of four. A pressure of 23.8 psig falls far below the 84 psig which the vessel can withstand before failure.

The containment building was pressurized to 26.25 psig before it was leak-tested at 21 psig. The design pressure of 21 psig seems adequate to ensure the integrity of the building under any conditions. The contributions to pressure buildup in the containment building caused by a zirconium-water reaction, hydrogen flammability, or steam-generator failure were not included in setting a design pressure since their chances of occurrence are deemed very slight. Even if included in the analysis, they would not raise the containment-building pressure more than a few points per square inch above the design pressure of 21 psig. The resulting maximum pressure would still be less than the test pressure of 26.25 psig.

The overhead water tank has a capacity of 30,000 gal. Of this, 15,000 gal are available for use in the building spray system. Water can be delivered to the building spray system at the rate of 1000 gpm, which will decrease the pressure and temperature within the building after the initial flashing of primary-system water. The variations of pressure and temperature with time were calculated over a period of 15 min.

Again it was assumed that no heat is lost to the outside air, or stored by the concrete or internal components. It was assumed that the low-pressure cooling system is operating and that it delivers 33.3 gpm of water at 70 F to the vessel. Since the water tank is at the top of the enclosure, it will receive heat from the sun reflecting on the roof of the building. It was therefore assumed that the temperature of the water in the overhead tank is 100 F. Decay heat from the reactor was also accounted for in the calculations.

The resultant variations of pressure and temperature for the time after rupture are shown in Fig. B-1, which shows that the pressure is reduced from 20.3 psig to 8.4 psig and that the temperature is reduced from 217 F to 168 F during the 15 min immediately following rupture. After 15 min, flow from the overhead tank stops. Thereafter, pressure and temperature will decrease slowly as heat is lost from the building.

CONTAINMENT BUILDING PRESSURE AND TEMPERATURE
VS. TIME AFTER MAXIMUM CREDIBLE ACCIDENT

FIG. B-1

APPENDIX C

HAZARDS TO THE SURROUNDING AREA

C1. INTRODUCTION

The maximum credible accident releases to the containment building 8.62 percent of the noble gases and halogens and 0.431 percent of the strontium contained in the core at the time of the accident. This appendix evaluates the hazards to the surrounding area as a result of this accident.

It is assumed that the containment building remains intact. The hazards, therefore, are due only to gamma radiation and fission-product leakage from the building. Radiation outside the building is due to direct and scattered gamma rays. The water tank at the top of the building is not lined with concrete and, shortly after the accident, would not contain water. Radiation could penetrate through the containment building at this point and give a dose at ground level from "sky shine."

Hazards from fission-product leakage are principally the result of whole-body gamma doses, total integrated inhalation doses, and ground deposition of fission products. It is pessimistically assumed that all leakage is at ground level and occurs at a single point. The escaping plume would give an external gamma dose to someone in or near the plume. A person in the plume would also receive an internal dose from inhalation of the fission products. Fission products in the plume would also be deposited on the ground.

C2. BUILDING RADIATION

The portion of the containment building above ground is a cylinder 74 ft in diameter and 59.5 ft high, topped by a hemisphere of 37-ft radius. The steel is 0.875 in. thick in the cylindrical portion, 0.5 in. thick in the hemispherical portion below the water tank, and 0.70 in. thick in the hemispherical portion above the water tank. The effective thickness of the concrete lining of the cylindrical portion is 2 ft at any distance from the base of the building. For distances up to 200 ft from the base of the building, the effective thickness of the concrete lining of the hemispherical portion is also 2 ft.

Sky shine is caused by air scattering of the gamma radiation that escapes through the water tank. Direct radiation is the gamma radiation which penetrates the concrete and steel shells of the containment building. For the calculation of direct radiation and sky shine, the gamma-source term (representing the total gamma energy released in the first hour after shutdown) was calculated by integrating the Way-Wigner equation over the first hour after shutdown, assuming previous operation at 116 Mwt. The source term thus calculated was adjusted to reflect the fact that during the first hour after shutdown only 20 percent of the total gamma energy is due to volatile fission products and to reflect the assumption that only 8.62 percent of the volatile fission products are released. On this basis, the total fission-product gamma-energy release to the containment building is 1.89×10^{20} Mev.

C2.1 Sky Shine

In calculating the sky shine, it was assumed that all the radiation emanates from a point source lying on the centerline of the containment building at a distance from the ground equal to one-half the building height above grade. The conservative assumption was made that the Klein-Nishina differential cross section for the scattering of energy is constant and is equal to its value at 90 deg. The equation used for sky-shine dose is*:

$$D = \frac{SN}{4x} (\Psi_2 - \Psi_1) \left(\frac{\pi}{2} - \phi \right) \sum_{i=1}^3 (f_i/C_i) \left[\sigma(\mu, E) \exp(-T/\lambda_i) \right] \quad (C-1)$$

where,

- D = sky-shine gamma dose during first hour, r
- S = gamma energy released during first hour, Mev ($= 1.89 \times 10^{20}$ Mev)
- N = number of electrons per cubic centimeter of air ($= 3.6 \times 10^{20}$)
- $\sigma(\mu, E)$ = Klein-Nishina differential cross section for the scattering of a photon of energy E, at a scattering angle whose cosine = μ , cm^2
- f_i = fraction of total gammas in the i-th energy group
- C_i = conversion factor for i-th energy group, Mev/r-cm^2
- T = thickness of steel shell, cm ($= 0.70$ in.)
- λ_i = relaxation length for gammas of i-th energy group in steel, cm
- h = building height above grade, cm
- x = distance from centerline of building, cm
- $\phi = \tan^{-1} h/x$, radians
- $\Psi_2 - \Psi_1$ = angle subtended at the source by the edges of the water tank, radians

All gamma rays were assumed to fall into three energy groups: 0.33, 1.1, and 2.1 Mev.

Table C-1 lists the sky-shine doses calculated in accordance with Equation C-1.

*MIT-5007, p. 85

TABLE C-1
SKY-SHINE DOSES DURING FIRST HOUR

<u>distance from building centerline</u>	<u>first-hour dose, r</u>
200 ft	0.0092
1/4 mile	0.0024
1/2 mile	0.0012

C2.2 Direct Radiation

To calculate the direct radiation, the portion of the containment building above grade level was divided into two parts: the hemispherical head and the cylindrical lower portion. The dose from each section was added to obtain the total dose. For the cylindrical portion, it was assumed that the total source could be replaced by an equivalent line source along the centerline. The equation used to obtain the total dose in the first hour after shutdown is*:

$$D = \frac{BSK}{4\pi hx} \int_0^{\theta_2} \exp \left[\left(- \sum_{i=1}^2 T_i / \lambda_i \right) \sec \theta \right] d\theta \quad (C-2)$$

where,

- D = direct gamma dose from cylindrical portion of building during first hour, r
- B = buildup factor, dimensionless
- S = gamma energy released during first hour, Mev. ($= 1.89 \times 10^{20}$ Mev)
- K = conversion factor, r/Mev-cm²
- h = height of cylindrical portion of building above grade, cm
- x = distance from center of building, cm
- T = thickness of *i*-th attenuating material, cm
- λ_i = relaxation length of gammas in *i*-th attenuating material
- $\theta_2 = \tan^{-1} h/x$, radians

*Reactor Shielding Design Manual, Theodore Rockwell, III, ed., D. Van Nostrand Co., Inc., 1st ed., p. 348.

Two feet of concrete and 0.875 in. of steel attenuate radiation from the building. A gamma energy of 2 Mev was assumed in calculating the attenuation and the buildup factor. However, an average gamma energy of 0.7 Mev was used to obtain K. The buildup factor was assumed to be that for 2-Mev gammas through 2 ft of concrete and was calculated from data presented on p.423 of the Reactor Shielding Design Manual.

The thickness of the concrete lining the hemispherical head varies with height. The hemisphere was therefore divided into four sections, the uppermost of which had no concrete and the other three of which had a concrete thickness equal to the average over the particular section. At 200 ft, all effective concrete thicknesses are 2 ft except that of the water-tank section. The equation for the direct-radiation dose from the hemisphere is thus:

$$D = \frac{4 BSK}{9\pi R^2} \sum_{m=1}^4 f_m \left[E_1(b_m) - E_1(b_m \sec \theta) \right] \quad (C-3)$$

where,

- D = direct gamma dose from hemispherical portion of building first hour, r
- B = buildup factor, dimensionless
- S = gamma energy released during first hour, Mev. ($=1.89 \times 10^{20}$ Mev)
- K = conversion factor, r/Mev-cm²
- R = radius of hemispherical portion of building, cm
- f_m = fraction of hemispherical volume occupied by m-th section, dimensionless

$$b_m = \left[\sum_{i=1}^2 T_i / \lambda_i \right]_m, \text{ dimensionless}$$

$$E_1(b_m) = \int_{b_m}^{\infty} \frac{\exp(-t)}{t} dt, \text{ dimensionless}$$

$$E_1(b_m \sec \theta) = \int_{b_m \sec \theta}^{\infty} \frac{\exp(-t)}{t} dt, \text{ dimensionless}$$

- T_i = thickness of i -th attenuating material in m -th hemispherical section, cm
 λ_i = relaxation length of gammas in i -th attenuating material in m -th hemispherical section, cm
 $\theta = \tan^{-1} \frac{R}{\sqrt{x^2 + (h')^2}}$, radians
 h' = height of cylindrical portion of building above grade plus vertical distance from top of cylindrical portion to bottom of m -th hemispherical section, cm

Table C-2 lists the total direct radiation doses from the cylindrical and hemispherical portions of the containment building, calculated in accordance with Equations C-2 and C-3.

Figures C-1 and C-2 show the dose from sky shine and direct radiation from the building as functions of distance from the building. It can be seen that the total doses are well below the 25-r maximum emergency dose.

TABLE C-2
DIRECT RADIATION DOSES DURING FIRST HOUR

<u>distance from building centerline</u>	<u>total first-hour dose, r</u>
200 ft	0.692
1/4 mile	0.0365
1/2 mile	0.0089

C3. FISSION-PRODUCT LEAKAGE

The containment-building leak rate is less than 0.1 percent of the contained volume per day at the design pressure of 21 psig. The leakage flow is assumed to be turbulent. Hence, the leak rate varies as the square root of the pressure difference. The doses calculated from fission-product leakage account for the decrease in leak rate owing to the pressure decrease resulting from actuation of the containment-building spray system. None of the fuel tubes melts until 6.83 min after shutdown. Ten minutes after shutdown, the building pressure has been reduced by the spray system to 10.55 psig. The initial leak rate for fission-product release is thus 0.0708 percent/day. Fifteen minutes after

shutdown, the pressure is 8.4 psig, and it decreases more slowly thereafter as heat is radiated from the building. A constant leak rate of 0.0633 percent/day is assumed for all periods more than 15 min after shutdown. These leak rates expressed as fractions of the total building volume per second are presented in Table C-3 and are equivalent to the fraction of total activity that escapes per second from inside the building.

TABLE C-3

FISSION-PRODUCT LEAK RATES VS.
TIME AFTER MAXIMUM CREDIBLE ACCIDENT

<u>time after accident,</u> <u>min</u>	<u>leak rate,</u> <u>fraction of total activity/sec</u>
0-10	-0-
10-15	8.19×10^{-9}
more than 15	7.32×10^{-9}

C3.1. Meteorology

Meteorological data needed to determine the fission-product distribution and concentration downwind were obtained from the Special Projects Group of the United States Weather Bureau. On this group's recommendation, a strong inversion, accompanied by low wind velocity, was used in making the meteorological calculations. These very pessimistic conditions are rare at the Elk River Reactor site.

The parameters employed in Sutton's equation for the atmospheric diffusion of the released activity are as follows:

<u>parameter</u>	<u>symbol</u>	<u>value</u>
isotropic diffusion constant	C^2	0.002
vertical diffusion constant	C_z	0.02
average wind speed	$\bar{u}$	1 meter/sec
stability constant	n	0.5
settling velocity for vapor	V_g	0.02 meter/sec
settling velocity for particulate matter	V_p	0.001

The Special Projects Group also recommended the use of a reduction factor to account for fluctuations in the centerline wind direction, in calculating inhalation doses. This reduction factor is supported by experimental work reported in HW-56605. It is the ratio of the dose at a given point with fixed centerline wind direction to the dose at the same point with truly random wind-direction fluctuations. It is assumed that the probability for wind to blow in a specified direction is described by a Gaussian distribution over a 22.5 deg arc. The ratio of the probability at the center to the probability at the edge of the arc is taken to be 1000. The resultant centerline dose reduction factors that are due to wind fluctuation around that point are listed in Table C-4.

TABLE C-4

WIND VARIABILITY REDUCTION FACTORS

<u>distance from containment building</u>	<u>wind variability reduction factor, dimensionless</u>
1/4 mile	3.77
1/2 mile	4.46
1 mile	5.30

A strong inversion is assumed in the dose calculations for radioactive leakage from the containment building. There is thus very little lateral diffusion of the cloud as it moves away from the site, which raises the intensity of radiation along the centerline but greatly reduces the probability of exposure by making the cloud width small.

For a continuous point source, the equation for cloud width (given on p. 49 of Meteorology and Atomic Energy) is:

$$2 Y_o = 2 \sqrt{\ln \frac{100}{P}} C_y x^{(2-n)/2}$$

where,

$2 Y_o$ = cloud width (meters)

x = distance downward from containment building, meters

P = radiation level at cloud edge as percent of centerline value, dimensionless

$C_y = (C^2)^{1/2}$ = lateral diffusion constant, dimensionless (=0.044)

For $P = 1$ percent, the results are as follows:

<u>distance from containment building</u>	<u>cloud width, ft</u>	<u>percent of circumference</u>
1/4 mile	56.4	0.69
1/2 mile	93.8	0.57
1 mile	156	0.47

It should be noted that pessimistic assumptions in addition to those of a strong inversion and low wind speed have been made in using the diffusion equations. It has been assumed that all fission-product release occurs from one point at ground level. Since most of the containment-building penetrations are near ground level, this assumption is not overly pessimistic, but it is extremely unlikely that all release will occur from one point. It was also pessimistically assumed that the escaping plume does not rise in spite of its greater-than-atmospheric temperature. Plume rise would increase the height of release and substantially reduce doses at all distances from the containment building.

C3.2 Whole-Body External Gamma Radiation

Whole-body gamma doses were computed for an 8-hr period after shutdown. The nomograph developed by J.Z. Hooland*, based on Sutton's equation for atmospheric diffusion, was used for this calculation. This nomograph takes into account decay of gamma energy during travel of the plume. However, an additional reduction factor must be introduced to account for the fission-product decay in the elapsed time before the leakage occurs. This reduction factor was computed at 200 ft, 1/4 mile, 1/2 mile, and 1 mile and for times of 10 min, 15 min, and 1 hr through 8 hr. Leakage was assumed to occur at ground level at the rates listed in Table C-3. Whole-body gamma doses at distances of 200 ft, 1/4 mile, 1/2 mile and 1 mile are listed in Table C-5. Figure C-3 shows whole-body gamma doses vs. distance, for an 8-hr exposure.

C3.3 Inhalation Doses

Inhalation doses are due almost completely to the inhalation of the isotopes of iodine and strontium. Inhalation doses were computed for the isotopes iodine-131, iodine-132, iodine-133, iodine-135, strontium-89, and strontium-90, at distances of 1/4 mile, 1/2 mile, and 1 mile from the containment building. Doses were computed by equation C-4.

*"Meteorology and Atomic Energy," Fig.83, p. 108.

TABLE C-5

WHOLE-BODY GAMMA DOSES FROM ESCAPING FISSION PRODUCTS
DURING FIRST EIGHT HOURS

<u>distance from containment building</u>	<u>dose during first 8 hr, r</u>
200 ft	0.870
1/4 mile	0.203
1/2 mile	0.114
1 mile	0.065

$$TID = k \left(\frac{X}{Q} \right) \frac{BZ}{f} \left[Q_1 \Delta t_1 + Q_2 \Delta t_2 \right] \quad (C-4)$$

TID = total integrated inhalation dose, i.e., the total lifetime dose received by a person inhaling a given amount of an isotope for 1 hr after the accident, rem

k = conversion factor, hr/sec. (=1/3600)

X/Q = concentration factor, sec/meter³

B = breathing rate, meter³/hr

Z = TID per microcurie inhaled, rem/microcurie

f = wind variability reduction factor, dimensionless

Q₁ = leak rate from 10-15 min (see Table C-7), microcuries/sec

Q₂ = leak rate after 15 min (see Table C-7) microcuries/sec

Δt₁ = 300 sec (= 5 min)

Δt₂ = 2700 sec (= 45 min)

The concentration factors at different distances from the containment building were computed by Sutton's equation*:

$$X/Q = \frac{2}{\bar{u} \pi C^2 x^{2-n}} \exp \left(-h^2 / C^2 x^{2-n} \right) \quad (C-5)$$

with h = 0 for ground release. The concentration factors are listed in Table C-6.

*"Meteorology and Atomic Energy," p.47.

TABLE C-6

AIRBORNE FISSION-PRODUCT CONCENTRATION FACTORS

<u>distance from containment building</u>	<u>concentration factor, X/Q, dimensionless</u>
1/4 mile	3.98×10^{-2}
1/2 mile	1.41×10^{-2}
1 mile	4.96×10^{-3}

Activity leak rates are based on T.J. Burnett's* values of source strength in curies/kw for 400 days operation. These values were corrected to account for an 8.62 percent release of iodine and a 0.431 percent release of strontium. Total activity release to the containment building for 116-Mwt operation and corresponding leak rates are listed in Table C-7.

TABLE C-7

IODINE AND STRONTIUM ACTIVITY LEAKAGE RATES

<u>isotope</u>	<u>activity in containment building, microcuries</u>	<u>Q₁ leak rate from 10 to 15 min, microcuries/sec</u>	<u>Q₂ leak rate from 15 min on, microcuries/sec</u>
iodine-131	2.60×10^{11}	2.13×10^3	1.90×10^3
iodine-132	3.95×10^{11}	3.24×10^3	2.89×10^3
iodine-133	5.82×10^{11}	4.77×10^3	4.26×10^3
iodine-135	5.29×10^{11}	4.33×10^3	3.88×10^3
strontium-89	2.19×10^{10}	1.80×10^2	1.61×10^2
strontium-90	0.72×10^9	0.59×10^1	0.525×10^1

A breathing rate of 1 meter³/hr was assumed. Values of Z were also obtained from T.J. Burnett's article, and values of f are given in Table C-4. The time periods considered were for 10 to 15 min and 15 min to 1 hr after the accident. The total integrated fission-product inhalation doses received during the first hour after the accident are listed in Table C-8.

Total iodine and strontium inhalation doses vs. distance are plotted in Figs. C-4 and C-5.

*Nuclear Science and Engineering, Vol. 11, p. 386, 1957.

TABLE C-8

TOTAL INTEGRATED FISSION-PRODUCT INHALATION DOSES
DURING FIRST HOUR

isotope	dose x miles downwind from containment building, rem		
	<u>x = 1/4</u>	<u>x = 1/2</u>	<u>x = 1</u>
iodine-131	16.67	5.00	1.46
iodine-132	0.69	0.20	0.04
iodine-133	8.74	2.64	0.77
iodine-135	2.20	0.65	0.20
total iodine	<u>28.30</u>	<u>8.49</u>	<u>2.47</u>
strontium-89	0.0886	0.0246	0.0008
strontium-90	<u>0.3310</u>	<u>0.0992</u>	<u>0.0293</u>
total strontium	0.4196	0.1256	0.0301

C3.4 Ground Deposition

The ground deposition of iodine and strontium isotopes as a result of the release of fission-product activity has been calculated at distances of 1/4, 1/2, and 1 mile downwind from the containment building. The rate of ground deposition is given by*:

$$R = \frac{2 Q_o V_g}{\bar{u} \pi C_x^2 x^{2-n}} \exp - \left(\frac{4 V_g x^{n/2}}{n \bar{u} \pi^{1/2} C_z} \right) \quad (C-6)$$

where,

R = rate of deposition, microcuries/meter²-sec
 Q_o = continuous point-source release in microcuries/sec

In determining deposition rates, the iodine isotopes were treated as vapors and the strontium isotopes as particles. Total depositions for 1 hr after the maximum credible accident are listed in Table C-9.

*"Meteorology and Atomic Energy, p. 93.

TABLE C-9

GROUND DEPOSITION CONCENTRATIONS AT END OF FIRST HOUR

isotope	ground-deposition concentrations x miles downwind from containment building, curie/m ²		
	x = 1/4	x = 1/2	x = 1
iodine-131	1.102 x 10 ⁻¹¹	0.882 x 10 ⁻¹³	3.050 x 10 ⁻¹⁶
iodine-132	1.370 x 10 ⁻¹¹	1.045 x 10 ⁻¹³	3.050 x 10 ⁻¹⁶
iodine-133	2.640 x 10 ⁻¹¹	1.948 x 10 ⁻¹³	6.811 x 10 ⁻¹⁶
iodine-135	2.236 x 10 ⁻¹¹	1.773 x 10 ⁻¹³	6.201 x 10 ⁻¹⁶
total iodine	7.168 x 10 ⁻¹¹	5.648 x 10 ⁻¹³	19.112 x 10 ⁻¹⁶
strontium-89	8.071 x 10 ⁻⁶	2.358 x 10 ⁻⁶	6.628 x 10 ⁻⁷
strontium-90	0.270 x 10 ⁻⁶	0.791 x 10 ⁻⁶	0.222 x 10 ⁻⁷
total strontium	8.341 x 10 ⁻⁶	3.149 x 10 ⁻⁶	6.850 x 10 ⁻⁷

Ground depositions of iodine and strontium at the end of the first hour vs. distance are plotted in Figs. C-6 and C-7.

SKY-SHINE DOSE VS. DISTANCE FROM CONTAINMENT BUILDING

FIG. C-1

DIRECT RADIATION DOSE VS. DISTANCE FROM CONTAINMENT BUILDING

FIG. C-2

WHOLE-BODY DOSE VS. DISTANCE FROM CONTAINMENT BUILDING

FIG. C-3

TOTAL INTEGRATED IODINE DOSES VS. DISTANCE FROM CONTAINMENT BUILDING

FIG. C-4

TOTAL INTEGRATED STRONTIUM DOSE VS. DISTANCE FROM CONTAINMENT BUILDING

FIG. C-5

GROUND DEPOSITION OF IODINE ACTIVITY VS. DISTANCE FROM CONTAINMENT BUILDING

FIG. C-6

GROUND DEPOSITION OF STRONTIUM ACTIVITY VS. DISTANCE FROM CONTAINMENT BUILDING

FIG. C-7

APPENDIX D

INHERENT SAFETY OF THE REACTOR

Since the inherent shutdown mechanisms of the ERR were first analyzed (see Appendix A of the original Final Hazards Report, July 8, 1960), two particularly relevant series of tests have been completed. The results of these tests permit conclusions regarding the inherent safety of the ERR to be based on experimental data to a greater extent than was possible at the time this appendix was first prepared. The newer data show that two assumptions underlying the original analysis were unnecessarily pessimistic, and, therefore, that the conclusions regarding the inherent safety of the ERR were more conservative than the observed behavior of similar reactors and prudent judgment would require.

The first test series was the BORAX IV experiments.* The core of the BORAX IV reactor was composed of urania-thoria pellets, 0.230 in. in diameter, inserted into aluminum tubes. The space between the pellets and the aluminum was filled with lead to improve heat transfer.

The BORAX IV series studied step reactivity increases from low power at room temperatures and at 207°F. Periods as short as 83 millisecon were terminated by inherent mechanisms, without core damage. With the small-diameter pellets and relatively long periods, the formation of steam accounted for part of the shutdown reactivity. There was, however, a definite contribution from the prompt reactivity coefficient associated with Doppler broadening of the uranium-238 absorption resonances.

The second test series was the SPERT I experiments.** The core of the SPERT I reactor was composed of Type-304 stainless-steel fuel tubes previously used in the N.S. Savannah critical assembly. The tubes were 6 ft long and 0.500 in. OD. They were filled with low-enrichment urania powder, swaged-compressed to an effective density of approximately 87 percent of the theoretical density of urania. All SPERT I data discussed here apply to the "constrained" core, because that core more nearly resembles the ERR core in which the fuel tubes are rigidly restrained at two axial positions between the end fittings.

The larger stainless-steel SPERT I fuel tubes more nearly resemble those of the ERR. In the SPERT I test series, self-limiting power excursions initiated by step increases in reactivity were induced on asymptotic periods ranging from 30 sec to 3.2 millisecon. Excursions on relatively long periods were terminated by the various heat-transfer-dependent mechanisms effecting loss of moderation from the core and by Doppler broadening of the uranium-238 absorption resonances. Excursions on short periods (i.e., on periods short relative to the thermal time constant of the fuel rods) were terminated principally by Doppler broadening alone.

* Maxon, B.X., O.A. Schulze, and J.A. Thie, "Reactivity Transients and Steady-State Operation of a Thoria-Urania-Fueled Direct-Cycle Light Water Boiling Reactor, BORAX IV" (ANL-5733), February 1959.

** Spano, A.H., J.E. Barry, L.A. Stephan, and J.C. Young, "Self-Limiting Power Excursion Tests of a Water-Moderated Low-Enrichment UO₂ Core in SPERT I" (IDO-16751), February 28, 1962.

In the excursion on the shortest asymptotic period (3.2 milliseC), the maximum power was about 7500 Mw. The total nuclear energy released during the excursion was about 107 Mw-sec, and the maximum steam pressure rise was about 8 psi. No damage occurred as a consequence of the test.

The results of the short-period tests were essentially those postulated in the original ERR analysis, with two important differences. In the original analysis, it was pessimistically assumed that it would be necessary to offset the initial reactivity increase entirely before the power level would begin to decrease. The tests demonstrated that it was necessary to compensate for the prompt reactivity alone in the initial portion of the transient. After that, steam voids would begin to form to assist in compensating for the delayed reactivity.

The second assumption shown to be pessimistic was that the melting point of stainless steel would determine the upper limit of the reactivity increase that could occur without damaging the core. However, at the instant of peak power during the excursion on the shortest period (3.2 milliseC), the temperature of the fuel tubes had risen only 8 C. Even though the fuel-tube temperature continued to rise after the instant of peak power, the maximum fuel-tube temperature measured was only about 250 C.

In the SPERT I reactor, it would be necessary to increase reactivity at a ramp rate of about \$8/sec to duplicate the results of the step increase that resulted in a 3.2-milliseC period. (As stated previously, no damage occurred as a consequence of that test.)

Comparison of the calculated Doppler coefficient of the ERR ($-\$0.29/^{\circ}\text{C}$) to the measured Doppler coefficient of the SPERT I ($-\$0.25/^{\circ}\text{C}$) reveals that ramp reactivity increases (starting from low power at room temperatures) at rates as high as \$8/sec could be tolerated without serious damage to the core.

APPENDIX E

THE GEOLOGY AND HYDROLOGY
IN THE VICINITY OF THE ELK RIVER, MINNESOTA, NUCLEAR REACTOR SITE,
WITH REFERENCE TO THE ACCIDENTAL SPILLAGE OF RADIOACTIVE MATERIAL

Prepared at request of U. S. Atomic Energy Commission

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E1. SUMMARY

The objective of this report is to determine the geologic and hydrologic conditions that would affect the movement of a hypothetical, spilled liquid at the Elk River, Minnesota, nuclear reactor site and to evaluate the possibility of such a spill reaching a major water facility downstream along the Mississippi River.

The report was prepared by the Water Resources Division of the United States Geological Survey for the Atomic Energy Commission. The study was under the direct supervision of Robert Schneider, district geologist, Ground Water Branch, St. Paul, Minnesota. Leon R. Sawyer, district engineer, Surface Water Branch, compiled most of the streamflow data for the Mississippi River.

The Elk River nuclear reactor site is on the east bluff of the Mississippi River about 35 miles northwest of Minneapolis and St. Paul.

The area is underlain by about 70 to 180 ft of glacial drift, including, at the top, as much as 120 ft of outwash deposits (valley train) of the glacial Mississippi River. The underlying Cambrian bedrock consists of marine sedimentary formations, including artesian sandstone aquifers.

A spilled liquid at the reactor site could follow one or both of two courses: (1) it could flow over the land surface and through an artificial drainage system to the river in a matter of minutes; (2) part or nearly all of it could seep downward to the water table and then move laterally to the river. This seepage might take from a few months to a number of years.

The St. Paul and Minneapolis water-supply intakes, 21 and 25 miles downstream, respectively, are the most important supplies to be considered in the event of a spill. Based on stream flow and velocity data for the Mississippi River near Anoka, the time of travel of the maximum concentration of a contaminant from the reactor site to the St. Paul intake was computed to be about 8 hr for the median maximum daily discharge. For this discharge the maximum concentration at the intake would be 0.0026 microcuries per cubic foot for the release of 1 curie of activity.

About 1-1/2 weeks were spent collecting basic data, such as well records and drillers' logs, and in a reconnaissance of the geology. Water-level altitudes were determined with an aneroid barometer, and most of the surficial geology was delineated in the office, using aerial photographs. No topographic quadrangle map of the area is available.

The only report on the hydrology and geology of the Elk River area that had been prepared earlier was written by Simpson (1958) for the Atomic Energy Commission.

E2. ACKNOWLEDGMENTS

The following well drillers spent considerable time and effort in furnishing well logs and discussing ground-water conditions in the area: McAlpine Brothers, Dayton, Minn.; McAlpine Brothers, Rogers, Minn.; and Kate Thomas, Elk River, Minn. Special thanks are due E. J. Nelson, Rural Cooperative Power Association (R.C.P.A.) plant superintendent, R.W. Anderson, Platting Engineer of Anoka County, and R. Halter, Superintendent of Utilities, Village of Elk River, for giving freely of their time.

E3. LOCATION

The area under investigation is roughly encompassed by a circle of two-mile radius, centering at the Rural Cooperative Power Association steam power plant at Elk River, Minn. (pl. 1). The plant is situated on the east bluff of the Mississippi River, about one-quarter mile southeast of the Village of Elk River. The village is about 35 miles northwest of Minneapolis and St. Paul at the confluence of the Elk and Mississippi Rivers in southeastern Sherburne County. The area is traversed by State Highways 101 and 218 and U.S. Highways 169, 52, and 10. The Great Northern and Northern Pacific Railroads serve the area, with their main tracks situated directly east of the plant site.

Additional information was compiled on major water facilities located downstream along the Mississippi River as far south as the Minneapolis municipal water-supply intake at Fridley, Minnesota (pl. 2).

E4. GEOLOGY

E4.1 Bedrock Formations

The deepest wells in the Elk River area penetrate sandstones, shales, siltstones, and dolomites of the St. Croixan series (see Table 1). In the Minneapolis-St. Paul area St. Croixan rocks are underlain by about 1,000 to 3,000 ft of Precambrian sandstones and shales, which may have been formed by the erosion of Precambrian lava flows (Schwartz, 1936). These flows are not present at the Twin Cities but are exposed in the gorge of the St. Croix River at Taylor Falls, Minnesota, about 35 miles to the northeast. The oldest rock in the Minneapolis-St. Paul area is Precambrian granite, which occurs at an altitude of about 1,350 ft below sea level.

Regionally the bedrock strata are on the northwest rim of a shallow structural basin whose center is near the Mississippi River in Minneapolis. The basin is slightly elongated north-east-southwest and is formed by rocks of Cambrian and Ordovician age, including several important artesian aquifers.

E4.2 Glacial Deposit and Alluvium

During Pleistocene time, glaciers advanced and retreated over the area several times, depositing till (heterogeneous, unsorted materials ranging in size from clay particles to boulders) directly from the ice, and stratified, sorted deposits (outwash) in meltwater streams within and in front of the ice mass.

In the Elk River area, the surficial deposits are largely outwash sand and gravel deposits (valley train) of the glacial Mississippi River. These deposits are found in a broad belt along the present Mississippi River and include the hill of sand and gravel upon which the R.C.P.A. plant is situated (pl. 1).

The valley train deposits in this area are bounded by glacial drift (which includes till and outwash deposits) on the north and south and by an extensive series of outwash deposits known as the Anoka Sand Plain (Cooper, 1935) on the east.

The alluvium consists of deposits of clay, silt, sand, and gravel that were deposited on the river flood plain. It covers at least part of the valley train deposits but is not differentiated from them on Plate 1 because of the similarity in lithology.

E4.3 Seismology

Earthquakes in various parts of the world have affected recording gauges on observation wells completed in the artesian aquifers in the Twin Cities area. Faults have been mapped at Hastings, Minn., about 45 miles southeast of Elk River, and in the St. Croix River valley about 40 miles east of Elk River; however, the possibility of subsequent earth movements of magnitude in this region is probably very remote.

E4. TOPOGRAPHY

The topography on the valley train deposits is relatively flat, with gently sloping hills and some undrained depressions. In contrast, the topography in the area of undifferentiated drift north of the valley train is of high relief, with steeply sloping hills and many undrained depressions.

Altitudes in the area range from about 852 ft above sea level (mean altitude of the Mississippi River) to about 1,100 ft on a morainic hill about one mile north of the north-central part of the map on Plate 1. An isolated morainic hill in the NE 1/4 sec. 2, T.32 N., R. 26 W. is more than 1,000 ft above sea level.

The R.C.P.A. plant is situated on an elongate hill trending about S. 10 deg E. The main plant structure is built into the west side of the hill; the surface altitude is about 871 ft at the base of the building and about 902 ft in the back yard on top of the hill. The west slope of the hill ranges in altitude from about 902 ft at the plant to about 858 ft at the base of the bluff; the 44-ft drop occurs in a horizontal distance of about 360 ft. The slopes on the west side of the hill immediately north and south of the plant are somewhat steeper. The hill slopes more gently to the northeast, from about 902 ft at the top to an estimated altitude of 875 ft in a creek channel about 1,200 ft from the plant.

E6. HYDROLOGY

E6.1 Surface Water

The Mississippi River and its tributary Elk River are the major streams in the area. The Elk River is dammed at the village-owned hydroelectric plant, forming Lake Orono in the Village of Elk River. The altitude of the flash boards at the dam is 871 ft above m.s.l. which is about 19 ft higher than the mean altitude of the Mississippi River (852 ft above m.s.l.). The following table summarizes the discharge characteristics of the Mississippi River:

Discharge of Mississippi River
(in cubic feet per second)

	Mississippi River at Elk River (a)		Mississippi River near Anoka (b)	
	cfs	date	cfs	date
Max.	49,200	4/12/52	75,900	4/14/52
Aver.	5,324	--	6,997	--
Min.	278	11/15/33	586	9/13/34

- (a) Gauged at Elk River; drainage area 14,500 sq. mi. (approx.); period of record, 1915-56. Flow slightly regulated, except during extreme floods, by six Government reservoirs on head-waters; total usable capacity, 1,640,610 acre-feet; total normal operating capacity, 948,900 acre-feet. Gauge-height during maximum discharge is about 10 ft higher than average.
- (b) Gauged 6-1/2 miles downstream from Anoka; drainage area 19,100 sq. mi. (approx.); period of record, 1931-56. Flow slightly regulated as stated in preceding paragraph. Gauge-height during maximum discharge is about 14 ft higher than average.

The area east of the plant was, under natural conditions, poorly drained. An intermittent creek drainage system apparently flowed easterly to the Rum River, which joins the Mississippi at Anoka. County ditch Number 10 was dug in a segment of this creek channel, reversing the flow from easterly to westerly in the SW 1/4 sec. 2 and the NE 1/4 sec. 11, T. 32 N, R. 26 W. (pl. 1). The upper reach of the creek channel, northeast of the plant, drains into the northern portion of county ditch Number 10; a series of culverts under the railroad tracks and highway lead to the Mississippi River.

The R.C.P.A. plant area has a well-planned system of sewers, which will be discussed more fully under the section, "Course of Wastes."

The City of Minneapolis depends wholly upon Mississippi River water for its municipal supply. The intake is at Fridley, about 25 miles downstream from Elk River. The following information on the Minneapolis supply was obtained from Mr. Clayton Bach, Plant Superintendent, City of Minneapolis Water Department*:

Population supplied -- About 520,000 (1961)
Average daily use -- 53,275,000 gal (1961)
Maximum daily use -- 119,335,000 gal (1961)
Storage -- Raw, none; finished, 117,000,000

Treatment -- Prechlorination, softening with lime and soda ash, coagulation with ferrous sulfate, and Ferrifloc as required; clarification stabilization with alum, carbon dioxide, or Ferrifloc, or a combination of these as required; rapid sand filtration, post-chlorination, and ammoniation.

The principal source of water for the City of St. Paul is the Mississippi River intake at Fridley, about 21 miles downstream from Elk River. The water is not treated directly, as in Minneapolis, but is pumped into a storage reservoir (Vadnais) made up of a system of lakes. The city also has two artesian well fields with a total of 34 wells in reserve. The following information on the St. Paul supply was obtained from Mr. F.W. Gallagher, Superintendent of Water Supply, City of St. Paul Water Department*:

*Figures revised in accordance with data submitted to Allis-Chalmers by RCPA in August and October, 1962.

Population supplied -- About 330,000 (1961)
Average daily use -- About 41.5 mg (1961)
Maximum daily use -- (period ending Dec. 31, 1962) 97 mg (1959)
Storage -- Raw, 10,100 mg**; finished, 92 mg (1962)
Potential storage (Bald Eagle Lake and other lakes) -- 1000 mg

Treatment -- Treatment at the St. Paul plant consists of aeration, coagulation with alum, softening with lime, recarbonation for pH stabilization, rapid sand filtration, and postchlorination.

The storage of the river water in the impounding lakes has some effect on the quality of the water. The lakes afford opportunity for settling of silt and clay, the death of some forms of bacteria, and some reduction in color. However, there is some increase in low forms of animal and vegetable growth. The main value of the lakes is that they tend to equalize the daily fluctuations in the quality of the river water and act as a raw-water storage reservoir to guarantee a constant source of supply.

The Northern States Power Company has a hydro-electric plant at Coon Rapids, about 18 miles downstream from Elk River. No other large users of surface water are known between Elk River and the St. Paul municipal water supply intake.

The Mississippi River is also used for fishing, boating, and swimming; however, no public beaches are known.

E6.2 Ground Water

Aquifers

The valley train deposits and river alluvium are considered as one hydrologic unit in this report. The available well-log data indicate that the unit ranges in thickness from a few feet to 120 ft and is made up of sediments ranging in size from silt to coarse gravel. A sandpoint can be driven almost anywhere within the border of the valley train deposits to obtain an adequate domestic supply of water.

Other sand and gravel bodies occur within the drift beneath the valley train deposits, and they may be considered important aquifers in some parts of this area.

The water table in the area ranges in depth from about 5 to 40 ft. It is about 5 ft below the land surface near the outer edges of the valley train deposits, where the topography rises rather abruptly. On the flood plain west of the river it is about 10 ft deep, whereas on the elongate hill on which the R.C.P.A. plant is located, the water table is about 40 ft below the surface.

**Includes storage in Vadnais system (7500 mg) and Centerville system (2600 mg).

Immediately below the glacial drift are interbedded strata of sandstone, shale, siltstone, and dolomite of the St. Croixan series. Wells in the area encounter these formations at depths ranging from about 70 ft to 180 ft. The thickness of the St. Croixan series in this area is around 500 ft. The sandstones contain abundant water under artesian conditions and a large number of flowing wells are completed in them. Thiel (1947) suggests the high static head is due to the southeast dip of the beds, the static level in the sandstones being almost the same as the static level in the drift in the central part of Sherburne County.

Natural Recharge, Discharge, and Movement

The shallow aquifers in the Elk River area are recharged by local precipitation. The bedrock aquifers may be recharged directly by precipitation where they are at or near the land surface, or they may receive recharge from the overlying drift if the static level in the bedrock is lower than the water level in the drift. In the winter, when the surface is frozen, little or no ground-water recharge occurs from precipitation. Generally, the largest increments of recharge occur at the time of the spring thaw. The amount of recharge from rains is, in part, dependent upon antecedent soil moisture conditions; recharge may occur only after soil moisture and vegetal requirements are satisfied.

Unconfined ground water moves by the most direct route from areas where the water table is high to those where it is low. In general, shallow ground water moves toward streams, discharging as stream-bed seeps and springs and/or moving downstream as underflow. In view of the fact that the shape of the water table is a subdued model of the surface topography, the probable direction of movement of ground water is everywhere toward the nearest body of surface water. Scattered elevations on the water table mostly north and east of the Mississippi (pl. 1) indicate apparent southward slopes of about 20 ft per mile in the north-central part of the area to about 5 ft per mile in the southeastern part of the area. The apparent westward slope is about 10 ft per mile, becoming much steeper adjacent to the river.

Silts and fine sands transmit water rather slowly, perhaps on the order of tens of feet per year. The rates of ground-water movement in coarser materials may be on the order of hundreds or perhaps thousands of feet per year, depending upon the permeability and the prevailing hydraulic gradient.

The static levels in the artesian sandstone aquifers are higher than the water table in the glacial drift; consequently, under natural conditions, water from the sandstones would tend to move upward into the drift.

Artificial Discharge

Between Elk River and Columbia Heights (pl. 2) three municipalities, Elk River, Anoka, and Coon Rapids, obtain water from the bedrock formations. The Village of Elk River (population 1,399) has one well completed in sandstone at 312 ft, pumping an average of 210,000 gpd and a maximum of 280,000 gpd. Anoka (population 7,396) has three wells in sandstone that range in depth from 430 to 450 ft, pumping an average of 1 million gpd and a maximum of 3-1/4 million gpd. Coon Rapids has one well, which is cased to 217 ft and open in the bedrock to 472 ft. In August 1958 another well was being drilled to an anticipated depth of about 900 ft. No data were available on the pumpage, because the water system was new.

The C. M. Babcock Roadside Memorial Park, on the bank of the Mississippi River, is maintained by the Minnesota Highway Department and is used by a great number of tourists during the summer months. There is only one shallow, hand-pumped well for the use of the public, but it is mentioned here because it is directly across the highway to the west of the R.C.P.A. plant.

Private water supplies downstream include:

(1) the Cargill Research Farm well about two miles southeast of the reactor site, completed in white sandstone at a depth of 315 ft; also five sandpoint wells scattered throughout the farm area, completed at depths of from 25 to 30 ft;

(2) the Northern States Power Company well at the Coon Rapids dam, completed in bedrock at a depth of 397 ft; used for cooling bearings and for the sanitary and domestic needs of the plant and eight dwellings on the company property;

(3) an undetermined number of small commercial establishments, including motels and trailer courts; and

(4) housing developments, individual residences, and farms, most of which are situated on the river terraces; source for most of these wells are the shallow outwash deposits.

Quality of Water

The water in the outwash deposits is reported to be generally hard, with a moderate quantity of iron. Some well owners report excessive amounts of iron. The temperature of the water is about 49 F. Water in the bedrock aquifers ranges in hardness from about 170 to 260 ppm and contains a rather high amount of iron; as a result, some wells are equipped with iron-removal units. The temperature is about 48 F to 49 F. Table 2 contains four chemical analyses for wells completed in bedrock; one analysis is included for untreated Mississippi River water collected at the St. Paul municipal water supply intake at Fridley.

E7. COURSE OF WASTES

A large volume (several thousand gallons) of liquid spilled on the surface at the reactor site could follow one or both of two courses. It could flow over the surface, through a system of drainage lines and ditches into the Mississippi River, or it could seep into the ground, percolate downward to the water table, and move laterally to the Mississippi.

E7.1 Surface

Plate 3 is a plan view of the R.C.P.A. plant, showing the approximate location of the existing plant drainage system and the diameters of the lines. The back yard of the plant, around the reactor containment, is a concrete slab about five inches thick. A fluid spilled upon it would probably run into the drain at the southeast corner of the building, flow directly into the highway ditch in front of the plant, and then through a 24-in. culvert under the double-lane highway to the river. Assuming that little or none of the liquid ran into the artificial drainage system but flowed off the slab to the west, it would move down a 44-ft embankment into the highway ditch, providing that soil conditions were such as to allow little percolation into the ground. If the fluid should flow to the south-east and remain west of the railroad tracks (pl. 1), it would probably be impounded in a discontinuous ditch between the tracks and a bituminous-surfaced road behind the plant. If the ditch could not contain the fluid, it would probably flow into a small depression about one quarter of a mile from the plant and remain there, or continue southward into county ditch Number 10, through two 36-in. culverts under the double-lane highway, and into the river. In the event that a spill passes over the railroad tracks and into the creek drainage system east of the plant, it would probably flow into one of the sloughs (shallow, closed depression with a swampy bottom) a short distance east of the tracks and might remain there indefinitely. If the spill were of sufficient volume to flow through the sloughs, it would enter county ditch Number 10 and flow westward into the river. It is conceivable that a mass of liquid could flow to the north along a bituminous-surfaced road between the railroad tracks and the plant. In this event it might possibly flow along the road or in a discontinuous shallow ditch between the road and the railroad fill and make its way into the Village of Elk River. Along this route a large portion of the spill would probably flow off the road to the west, down the slope to the ditch system on the east side of the double-lane highway at the base of the bluff, thence into the Mississippi River.

The above-mentioned possibilities of surface drainage are based on the assumption that little or none of the liquid would seep into the ground. This condition would prevail if there were an abundance of soil moisture or if soil frost prevented or impeded infiltration. The first condition would exist in the spring of the year at the time of thaw or immediately after or during intense rainstorms. The second condition is possible during the winter months (November through March); however, the frost layer may not be continuous throughout the plant area because the system of sewers would tend to keep the ground from freezing completely.

A postulated fluid spill reaching the river via the plant drainage system would probably present the greatest hazard so far as contamination of the river is concerned. By this route several thousand gallons of fluid might reach the river in perhaps 5 to 10 minutes. The greatest hazard involved here is possible contamination of the water supplies of St. Paul and Minneapolis, whose intakes are 21 and 25 miles downstream, respectively. In assessing this hazard, the time of travel and dilution of the waste must be considered. Figure E1 is a plot of discharge versus average velocity as measured on the Mississippi River at Elk River, and Fig. E2 is a similar plot for the gauging station near Anoka.

From Fig. E2 and the discharge records and channel properties at Anoka, the dispersal patterns can be computed for the following flow conditions.

- (1) Median maximum daily discharge, about 29,000 cfs.
- (2) Median minimum daily discharge, about 2,000 cfs.
- (3) Average discharge, 7,000 cfs.

The discharge measurements at Anoka yield the following information for these selected discharges:

Discharge cfs	Average Velocity fps	Mean Depth ft
2,000	0.90	4.5
7,000	1.80	6.7
29,000	3.75	12.4

Three questions regarding any contaminant entering the stream at the reactor site are of interest to the Twin City water users:

- (1) For a given amount of contaminant, what concentrations may be expected at the St. Paul water intake at Fridley?
- (2) What distribution will this contaminant have with respect to time or distance along the reach?
- (3) How soon after the contaminant enters the Mississippi River at the reactor site will it reach Fridley?

The answers to these questions can be gained by examining the dispersal pattern in the Elk River-Fridley reach of the Mississippi River for three selected discharges. The dispersal pattern in the Elk River-Fridley reach can be computed as follows:

$$S = \frac{M}{BD \sqrt{\pi 4Kt}} e^{-\frac{(x-\bar{V}t)^2}{4Kt}}$$

where,

- S = the concentration in millicuries per cubic foot
- M = the amount of contaminant introduced in millicuries
- B = the stream width in feet
- D = the stream depth in feet
- t = the time in seconds
- x = the distance downstream from the point of introduction of the contaminant
- $\bar{V}$ = the average velocity
- K = the diffusion coefficient in ft²/sec

If the parameter K can be computed for this reach, then the dispersal pattern can be determined. Previous experimental work indicates that a conservative estimate of the diffusion coefficient is:

$$K = 20.2D (gDS)^{1/2}$$

where,

- S = the slope of the water surface
- g = gravity (32.2 ft/sec²)

The reach from Elk River to Anoka is 19.8 miles in length, and the difference in water surface elevation is about 43.5 ft for the range in discharge under consideration; thus, the average water surface slope is 0.000416.

The diffusion coefficients for the three conditions are tabulated below:

Discharge cfs	Diffusion coefficient ft ² /sec
2,000	23
7,000	41
29,000	103

Using the basic diffusion equation the dispersal patterns for Fridley can be computed, and are shown in Fig. E3.

In the computations given below the following assumptions have been made:

- (1) Cross-sectional properties at Anoka are representative of the entire reach.
- (2) Uniform flow throughout the reach.
- (3) Spot injection of a soluble contaminant.
- (4) No loss of contaminant through uptake by sediment or aquatic life.

It must be pointed out that these assumptions will not be completely satisfied and that the dispersion will be greater than shown herein.

The maximum concentrations shown on Fig. E3 are tabulated below for a release of one curie:

Anoka discharge cfs	Maximum concentration at Fridley per curie release	
	$\mu\text{c}/\text{ft}^3$	$\mu\text{c}/\text{ml}$
2,000	0.0093	$3.3(10)^{-7}$
7,000	0.0058	$2.1(10)^{-7}$
29,000	0.0026	$0.9(10)^{-7}$

It is coincidental that the three curves are spread over the same linear distance. The depth and velocity increase in such a manner to offset each other.

If amounts of contaminant other than one curie were introduced, the resulting concentrations at Fridley would be proportional to the spill.

The time of travel of the center of the slug of contaminant is directly related to the mean velocity. The following times of travel were computed for the indicated discharges:

Anoka discharge cfs	Mean Velocity fps	Time of travel to Fridley hrs
2,000	0.90	34
7,000	1.75	17
29,000	3.80	8

As shown in Fig. E-3, the initial arrival of the contaminant will precede the center of the slug.

E7.2 Subsurface

Assuming the spilled liquid does not flow overland but percolates into the ground, it would filter through about 40 ft of dry to partially saturated fine sand to coarse gravel before reaching the water table (pl. 4). About 7 ft below the surface at the reactor site is a carbonate-cemented gravel layer ranging in thickness from 1/2 to 1-1/2 ft and apparently dipping about 3 deg to the south. Because this layer is, in places, poorly cemented and permeable, it is conceivable that part of the layer could impede downward percolation of a fluid for some time, and other parts could facilitate its movement. Drillers report a similar "hardpan" layer a few feet thick, at depths of 16 ft or more in the valley train deposits south of the plant. They report that water is found immediately under this layer.

At an altitude of about 861 ft above mean sea level, water-table springs emanate as seeps along the base of the hill on which the R.C.P.A. plant is situated (see pls. 1 and 4). It is possible that at least part of a fluid contaminant that might join the water table near the plant would eventually issue from these springs.

Because of the variable permeability of the outwash it is difficult to estimate the time that it would take for a fluid to percolate through the outwash at the plant site. In all probability part of the fluid would be held by capillary attraction until subsequent recharge from rainfall would either displace or wash it down. Downward percolation to the water table and lateral movement at or below the water table would be quite slow. The rate of lateral ground-water movement in outwash deposits of the type found in this area might be on the order of hundreds or perhaps thousands of feet per year. The reactor site is about 750 ft from the river.

E8. CONCLUSIONS

A liquid spill at the reactor site would ultimately reach the Mississippi River. There are essentially two courses for the liquid to take:

(1) it may flow directly over the surface or through a system of drainage ditches and sewers into the Mississippi River. The time required to reach the river might vary over extremely wide limits, depending on the route travelled and the amount of liquid spilled. By direct overland flow or in the sewer system a contaminant might reach the river in a matter of minutes. Through the sloughs and drainage ditches each of the plant, a spill might take a relatively long period of time to reach the river or it might remain in the sloughs indefinitely;

(2) part of the liquid may percolate into the ground, in which case it must percolate downward through about 40 ft of dry to partially saturated permeable material before reaching the water table, and then move laterally at or below the water table, toward the Mississippi River. This could take from a few months to a number of years, depending, in part, on the permeability and the prevailing hydraulic gradient. Another possibility is that part of the underground movement will be toward the springs south of the plant, from which point the liquid would reach the river via the highway drainage system.

Under natural conditions it is unlikely for shallow ground water to percolate into the bedrock formations, since the static head in the bedrock is higher than the static head in the glacial drift. Also, in most cases, there is relatively impervious till, or shale, or both, between the drift aquifers and the bedrock aquifers.

Based on streamflow and velocity data obtained at the Anoka gauging station, the time of travel of the maximum concentration of a contaminant from Elk River, Minn., to the St. Paul water-supply intake (21 miles) was computed to be about 8 hr for the median maximum daily discharge (29,000 cfs). For this discharge the maximum concentration at the intake would be 0.0026 microcuries per cubic foot for the release of 1 curie of activity at the reactor site. The initial arrival of the contaminant would precede the maximum concentration by about 1 hr.

Further investigative work would be needed to answer the several questions that have been raised. Specifically, field determinations should be made of the permeability of the outwash deposits at the plant, and detailed water-table maps should be constructed to determine the direction of ground-water movement. Consideration should also be given to the experimental determination of the course and time of travel of a spilled liquid containing a suitable tracer. If it is found that ground water from the plant area issues from the springs to the south, they should be monitored and sampled periodically to establish background activity.

E9. LOGS OF WELLS AND SOIL BORINGS

E9.1 Well Logs

Material	Thickness (ft)	Depth (ft)
Coon Rapids municipal well, SW 1/4 NW 1/4 sec. 13, T.31 N., R.24 W.*		
Sand	45	45
Clay, blue	15	60
Clay, sandy	41	101
Hardpan	23	124
Sand	21	145
Sand and clay	3	148
Sand and gravel	20	168
Hardpan	9	177
Sandstone, soft	30	207
Sandstone and shale	158	365
Clay, red	5	370
Shale and sandstone	26	396
Sandstone with some shale	12	408
Sandstone (417-437 very soft)	36	444
Sandstone with some shale	24	468
Shale	4	472
NE 1/4 NE 1/4 NE 1/4 sec. 2, T.32 N., R.26 W.**		
Sand, reddish	25	25
Hardpan	7	32
Clay	5	37
Sand, dirty	5	42
Sand and gravel	18	60
NE 1/4 NE 1/4 NW 1/4 sec. 2, T.32 N., R.26 W.**		
Clay	38	38
Sand, fine	7	45
Till	23	68
Sand and gravel	40	108

E9.1 Well Logs (cont'd)

Material	Thickness (ft)	Depth (ft)
NE 1/4 SE 1/4 SE 1/4 sec. 3, T.32 N., R.26 W.**		
Sand and gravel	65	65
Clay, yellow to blue	35	100
Sand and gravel	65	165
Shale, varied colors	75	240
Sandstone, white	170	410
R.C.P.A. plant well, SE 1/4 NW 1/4 NE 1/4 NE 1/4 sec. 3, T.32 N., R.26 W.*		
Pit	14	14
Sand and gravel	16	30
Clay, sandy	10	40
Clay, blue	44	84
Sand, gravel, rocks, and hardpan	52	136
Sandstone	22	158
Shale	47	205
Sandstone, some shale	65	270
Sandstone, soft	20	290
Sandstone, caving	1	291
NW 1/4 NW 1/4 NW 1/4 sec. 11, T.32 N., T.26 W.**		
Sand and gravel	40	40
Clay, blue	85	125
Sandstone, soft	70	195
Sandstone, hard	60	255
SW 1/4 SE 1/4 NE 1/4 sec. 32, T.33 N., R.26 W.**		
Sand, gravel, and boulders	20	20
Sand and gravel	15	35
SE 1/4 SE 1/4 NW 1/4 sec.32, T.33 N., R.26 W.**		
Sand	25	25
Clay	45	70
Gravel, coarse, and sand	50	120
Shale	6	126
Sandstone	21	147

E9.1 Well Logs (cont'd)

Material	Thickness (ft)	Depth (ft)
SW 1/4 NE 1/4 SW 1/4 sec. 32, T.33 N., R.26 W.**		
Sand and gravel	30	30
Sand, gravel, and clay	30	60
Sand and gravel	6	66
NW 1/4 SW 1/4 SW 1/4 sec. 33, T.33 N., R.26 W.**		
Sand	25	25
Sand and gravel, dirty	25	50
Gumbo, hard, black	50	100
Shale, green	75	175
Sandstone	35	210
Elk River municipal well, NW 1/4 NE 1/4 SW 1/4 sec. 34, T.33 N., R.26 W.***		
Sand and gravel	120	120
Sandstone, yellowish, and shale	50	170
Sandstone, white	130	300
Sandstone, dark red	8	308
SW 1/4 SW 1/4 NW 1/4 sec. 35, T.33 N., R.26 W.**		
Sand	15	15
Clay	25	40
Sand and gravel	10	50
NW 1/4 NW 1/4 SW 1/4 sec. 35, T.33 N., R.26 W.**		
Sand	40	40
Clay, gray	20	60
Sand and gravel, reddish to gray	25	85

E9.1 Well Logs (cont'd)

Material	Thickness (ft)	Depth (ft)
SE 1/4 SE 1/4 SW 1/4 sec. 35, T.33 N., R.26 W.**		
Clay, yellow	30	30
Sand and gravel, dirty	50	80
Sand, green	5	85
NW 1/4 NE 1/4 NW 1/4 sec. 26, T.121 N., R.23 W.**		
Sand and gravel, fine	40	40
Clay, gray	60	100
Sand and gravel, dirty	20	120
Sandstone, soft	55	175
Sandstone, hard	80	255

* From drillers' records

** Logs reported from memory

*** From files of Minnesota Geological Survey

E9.2 Soil Borings (1)

Material	Thickness (ft)	Depth (ft)
A-C1 (2)		
Fine brown sand	2	2
Fine sand	2	4
Medium to coarse sand	1	5
Medium to coarse sand and gravel with veins of fine sand	1	6
Medium to fine sand with small stones - one inch in diameter . .	1	7
Fine sand with layers of coarse sand	1	8
Medium to fine sand and gravel	7	15
Fine sand	1	16
Medium to fine sand and gravel	3	19
Medium sand and gravel	1	20
Fine sand	3	23
Medium to fine sand and gravel	4	27
Medium to fine sand with trace of gravel	1	28
Medium to fine sand and gravel	3	31
Fine sand and gravel	3	34
Medium to fine sand and gravel	6	40

(1) See plate 3 for location of soil borings.

(2) Borings designated "A-C" made by Caswell Engineering Co., Minneapolis, Minn., July 1958.

E9.2 Soil Borings (cont'd)

Material	Thickness (ft)	Depth (ft)
A-C 2		
Fine sand with trace of gravel	1	1
Fine brown loamy sand	2	3
Fine sand	1	4
Fine brown loamy sand	2	6
Medium to fine sand	1	7
Fine brown loamy sand	5	12
Medium to fine sand and trace of gravel	2	14
Medium to fine sand	6	20
Fine sand with trace of gravel	9	29
Medium to fine sand and gravel	4	33
Medium to coarse sand and trace of gravel	3	36
Medium to coarse sand and gravel	2	38
Medium to fine sand and gravel	2	40
A-C 3		
Fine sand with trace of gravel	3	3
Fine loamy sand	4	7
Fine sand with trace of gravel	1	8
Fine loamy sand	16	24
Fine loamy sand with veins of fine sandy loam	2	26
Medium to fine sand	2	28
Fine sand with trace of gravel	12	40

E9.2 Soil Borings (cont'd)

Material	Thickness (ft)	Depth (ft)
AMF K-14 (3)		
Concrete	0.5	0.5
Tan and brown coarse loamy sand (moist) with a little gravel . . .	2	2.5
Light brown sand (moist) with a little gravel and lense of brown sand at 3-1/2 ft and a lense of sandy loam at 11 ft . . .	14.5	17
Light brown coarse sand (moist to 41 ft then water bearing) with some gravel	27	44
Grayish brown sandy loam (moist) with a little gravel and lenses of clay loam	2	46
Gray clay loam (rather stiff) with a little gravel	2	48
Brown sandy loam (moist) with a little gravel	11	59
Gray clay loam (very stiff) with a little gravel	6	65
Gray fine sandy loam (moist)	1	66
Gray clay loam (hard).	3	69
Gray sandy loam (moist)	8	77
Gray clay (hard) with a little gravel	7	84
Gray sand with a little gravel	1	85
Gray clay (hard) with a little gravel	10	95
Dark gray clay (hard)	2	97
Gray clay (hard) with a little gravel	3	100
AMF J-13		
Concrete	0.2	0.2
Tan & brown sandy loam (moist)	1.8	2
Tan & brown coarse loamy sand (moist)	0.5	2.5
Tan & light brown sand (moist)	2	4.5
Light brown sand (moist) with a little gravel & a few lenses of sandy loam from 10 to 12 ft.	9	13.5
Fan fine loamy sand (moist)	1	14.5
Tan sand (moist)	4.5	19
Light grayish-brown coarse sand (moist) with some gravel, a few lenses of sandy loam	23	42
Brown sand loam (moist) with a little gravel and boulders	5	47
Grayish-brown sandy loam (moist) with a little gravel	12.5	59.5
Gray clay loam (very stiff to hard) with a little gravel	10	69.5

(3) Borings designated "AMF" made by Twin City Testing and Engineering Laboratory, Inc., St. Paul, Minn., March 1957.

E9.2 Soil Borings (cont'd)

Material	Thickness (ft)	Depth (ft)
AMF J-15		
Concrete	0.5	0.5
Light brown coarse loamy sand (moist)	2	2.5
Light brown sand (moist)	2	4.5
Brown coarse sand (moist) with a little gravel	1.5	6
Light brown sand (moist) with a little gravel to about 9 ft, then with some gravel	8	14
Grayish-brown coarse sand (moist) with some gravel	4	18
Light brown coarse sand (moist) with gravel to about 22 ft, then with some gravel	11	29
Dark brown coarse sand (moist) with some gravel and a lense of clay loam at 31 ft	3	32
Grayish-brown sandy loam (moist) with a little gravel and a few boulders below 57 ft	27.5	59.5
T-1 (4)		
Reddish brown loamy sand and gravel	1.5	1.5
Reddish brown sandy loam	24.5	26
Gray clay loam - very stiff	2	28
T-2		
Brown sand and gravel	15	15
Dark brown sandy loam	1	16
Gray gravel - dense	2	18
Gray gravel - very coarse	2	20
Brown sandy loam	11	31
Gray clay loam - stiff	3	34

(4) Borings designated "T" made for R.D. Thomas and Associates, Minneapolis, Minn.

E9.2 Soil Borings (cont'd)

Material	Thickness (ft)	Depth (ft)
T 3		
Brown sand and gravel	6	6
Dark brown sandy loam - moist	20.5	26.5
Blue clay loam - very stiff	1.5	28
T 4		
Brown sand, some gravel	12	12
Gray coarse sandy loam	3	15
Coarse sand, some gravel	3	18
Gray sand	1.5	19.5
Gray sand and gravel	5.5	25
Brown sand and gravel	10	35
Gray clay loam - stiff	15	50

E10. ANALYSES OF SAND AND GRAVEL SAMPLES

Radioactive Waste Disposal

Five samples of sand and gravel from the east bluff of the Mississippi River about 1/4 mile SE of Elk River, Minnesota, were submitted for analyses.

<u>Serial No.</u>	<u>Field No.</u>	<u>Description and Location</u>
268691	1	Sand, 0-6', Elk River Reactor Site, Minn.
268692	2	Sand, 6-7.5' "
268693	2'	Cemented, 6-7.5' "
268694	3	Sand, 7.5-21' "
268695	4	Sand, 21-22' "

Total ion exchange capacity of each sample was determined using the Bower-Truog method. Due to the large number of gravel sized particles in samples 268692 and 268693, the exchange capacity of the medium sand (0.25-0.5 mm) was determined in lieu of the total sample. The results are as follows:

<u>Field No.</u>	<u>Exchange Capacity</u>
1	1.6 m.e./100 gpm
2	1.2 "
2'	1.8 "
3	1.9 "
4	1.7 "

Mechanical Analyses

Grain size determinations were made of each sample utilizing the standard pipette method for silt and clay fractions. The sand sizes were fractionated according to Wentworth scale.

<u>Fraction</u>	<u>Description</u>	<u>Percentages for sample number</u>				
		<u>1</u>	<u>2</u>	<u>2'</u>	<u>3</u>	<u>4</u>
>2 mm	Gravel	1.4	33.8	49.6	0.0	10.7
1-2 mm	Very coarse sand	3.3	15.8	15.0	0.1	7.7
0.5-1 mm	Coarse sand	16.4	30.8	10.0	2.4	33.2
0.25-0.5 mm	Medium sand	45.7	13.9	6.4	39.5	36.1
0.125-0.25 mm	Fine sand	24.8	2.1	10.4	50.8	8.9
0.06-0.125 mm	Very fine sand	3.5	0.5	1.4	4.4	0.3
0.002-0.06 mm	Silt	1.6	1.4	2.6	2.0	1.9
<0.002	Clay	1.7	0.5	3.5	2.7	1.2

The gross mineralogy of sample 1 was determined. The principal minerals in each size grade are as follows:

A = Abundant C = Common P = Present in small quantities

2 mm		1-2 mm		0.05-1 mm	
Quartz	A	Quartz	A	Quartz	A
Feldspar	A	Plagioclase Feldspar	A	Plagioclase Feldspar	A
Magnetite	P	Potassium Feldspar	A	Potassium Feldspar	C
Mica	P	Magnetite	P	Magnetite	P
Amphibole	P	Mica	P	Mica	P
Chlorite	P	Amphibole	P	Amphibole	P
		Chlorite	P	Chlorite	P

0.25-0.5 mm		0.125-5 mm	
Quartz	A	Quartz	A)
Plagioclase Feldspar	C	Plagioclase Feldspar	C)
Potassium Feldspar	C	Potassium Feldspar	C)
Magnetite	P	Magnetite	P)
Mica	P	Amphibole	A)
Amphibole	P	Iron Oxides	P)
		Garnet	P)
		Epidote	P)
		Mica	P)
		Others	P)

Light Minerals

Heavy residue = 3% of fraction

0.06-125 mm		Silt		Clay	
Quartz	A)	Quartz	A	Quartz	C
Plagioclase Feldspar	C)	Plagioclase		Feldspar	C
Potassium Feldspar	C)	Feldspar	C	Jarosite	C
Magnetite	C)	Potassium		Montmoril-	
Amphibole	A)	Feldspar	C	lomite	C
Iron Oxides	P)	Magnetite	P	Mica	P
Garnet	P)	Mica	P	Chlorite	P
Epidote	P)	Iron Oxides	P		
Mica	P)				
Others	P)				

Heavy residue = 7% of fraction

REFERENCES FOR APPENDIX E

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Table 1

Principal formations penetrated by wells in the vicinity
of Elk River, Minnesota

System	Series	Formation		Approximate Thickness (feet)	Physical Characteristics	Water-Bearing Properties
Quaternary	Recent	Alluvium and Lake Deposits		(?)	Clay, Silt, Sand, Gravel, and Marl	Undetermined
	Pleistocene	Glacial Drift	Till	70 - 180	Heterogeneous, unsorted deposits ranging in size from clay to boulders with some isolated sand and gravel lenses.	Yields small supplies to a few domestic wells.
			Outwash		Poorly to well stratified and sorted deposits ranging in size from clay to gravel.	May yield small to large supplies.
Cambrian	St. Croixan	Undifferentiated		500±	Friable sandstones predominate with some dolomite, shale, and siltstone.	Sandstones yield large supplies with high heads. A large number of flowing wells are completed in the sandstones.

TABLE 2 -- CHEMICAL ANALYSES OF GROUND WATERS AT ELK RIVER AND COON RAPIDS, MINN. AND OF MISSISSIPPI RIVER WATER AT FRIDLEY, MINN.

(Analytical results in parts per million except as indicated)

Location	Depth of Well (feet)	Analyst	Date of Collection	Temperature (°F)	Silica (SiO ₂)	Total iron (Fe)	Calcium (Ca)	Magnesium (Mg)	Sodium (Na)	Manganese (Mn)	Bicarbonate (HCO ₃)	Carbonate (CO ₃)	Sulfate (SO ₄)	Chloride (Cl)	Fluoride (F)	Nitrate (NO ₃)	Boron (B)	Dissolved solids		Hardness as CaCO ₃			Nitrate Nitrogen	Turbidity	Specific conductance (micromhos)	pH	Color	Remarks	
																		Sum	Residue	Total	Noncarbonate hardness as CaCO ₃	Percent sodium							
Coon Rapids Municipal Well SW/4NW/4 sec. 13, T. 31 N., R. 24 W.	472	Minn. Dept. of Health	4/22/58	--	--	0.42	--	0.28	--	--	--	--	2.0	--	0.12	--	--	--	--	170.0	--	--	1.0	--	--	7.6	--	--	
Northern States Power Co. Hydroelectric Plant Well SE/4NE/4 sec. 27, T. 31 N., R. 24 W.	398	No. States Power Co.	12/22/53	--	--	(Al+Fe) 0.3	54.6	16.4	15.1	--	261.9	--	19.2	1.8	--	--	--	236.1	--	204.0	0.0	--	--	--	335.0	7.5	--	H ₂ S=0.3	
Rural Cooperative Power Association Plant Well SE/4NW/4 sec. 3, T. 32 N., R. 26 W.	275	H. M. Mueller Co., Mpls., Minn.	--	--	18.2	1.10	140.0	94.00	24.0	--	--	--	10.0	3.0	--	--	--	256.5	--	232.6	--	--	--	--	--	7.4	--	--	
Elk River Municipal Well, NW/4NE/4SW/4 sec. 34, T. 33 N., R. 26 W.	308	Minn. Dept. of Health	12/19/57	--	--	0.52	54.0	--	--	<.01	--	--	--	--	1.60	--	--	--	--	260.0	--	--	--	--	--	7.7	--	--	
Untreated Mississippi River Water (average)	---	City of St. Paul	1940-49	52	9.0	0.30	43.53	15.70	(Na and K as Na) 3.04	0.10	--	--	15.87	2.99	--	0.42	--	--	241.0	179.0	--	--	--	9.1	--	8.1	62.5	Dissolved O ₂ =9.6 Al=0.72

PLATE 2.-MAP OF MISSISSIPPI RIVER AND TRIBUTARIES FROM ELK RIVER
TO COLUMBIA HEIGHTS, MINNESOTA.

SCALE
0 1 2 3 4 MILES

PLATE 4.-GEOLOGIC CROSS SECTION A-A', RURAL COOPERATIVE POWER ASSOCIATION PLANT, ELK RIVER, MINNESOTA.

UNITED STATES DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WATER RESOURCES DIVISION

FIGURE E-1

File { Washington
Number { District

MISSISSIPPI RIVER AT ELK RIVER, MINN.

16-12603-2 GPO

9-213-A
(October 1949)

UNITED STATES DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WATER RESOURCES DIVISION

FIGURE E-2

File { Washington
Number { District

MISSISSIPPI RIVER NEAR ANOKA, MINNESOTA

16 13468 2 GPO

FIGURE 2: GRAPH SHOWING AVERAGE VELOCITY VS. DISCHARGE
OF MISSISSIPPI RIVER NEAR ANOKA, MINNESOTA

Figure 3.— Distance - concentration curves at Fridley, Minnesota for release of one curie of activity into the Mississippi River at the Elk River nuclear reactor site.

APPENDIX F

METEOROLOGICAL ESTIMATES FOR ELK RIVER, MINNESOTA

Prepared by

Special Projects Section
Office of Meteorological Research

U.S. Department of Commerce
Weather Bureau

F1. SUMMARY OF SALIENT WEATHER FACTORS

Weather data are available only from adjacent locations rather than from the proposed site. This limits the analysis to a general estimation of the climate. However, meteorological data at hand indicate the following salient features of the climatology of the Elk River area, based, in part, on subjective analysis:

(a) Relatively high frequency of nighttime inversions in summer, which are rapidly destroyed by daytime solar heating. Such inversions will occur on nights when skies are relatively clear and winds are light.

(b) Higher probability of longer duration periods of inversions in fall and winter. However, the frequent passage of storm systems influencing the area during these seasons and the subsequent high wind speeds suggest that the occurrence of a stagnation period in excess of three days would be infrequent.

(c) The prevalence of relatively high wind speeds, at least during daytime hours, even though concurrent with a persistent subsidence inversion associated with an Arctic air mass in winter, would permit good ventilation during the day. Minimum transport of airborne material would be realized during the nocturnal period when light winds and relatively clear skies would permit the establishment of a surface-based inversion.

(d) The occurrence of precipitation concurrent with inversion existence is rare; the greatest probability of such conditions are in the winter and spring when there is warm moist air advection from the south and existing snow cover. Precipitation winds are primarily from the northwest-north and east-southeast sectors.

(e) Visibility data represent a poor parameter of stability index, since a large percentage of visibility restrictions are due to dust, snow, and blowing snow in this region.

Additional analysis of meteorological features in the immediate vicinity of the Elk River site would require on-site observations. Specific features such as the possible effect of the adjacent river upon the vertical temperature lapse rate and air movement, channeling and drainage winds, etc. would require special investigations.

F2. LOCATION AND TOPOGRAPHY

Elk River is located on the eastern bank of the Mississippi River, about 35 miles northwest of Minneapolis, Minnesota. The surrounding country consists mostly of gently rolling land with numerous small lakes.

F3. METEOROLOGICAL DATA

U.S. Weather Bureau offices are located at the Minneapolis-St. Paul Airport and St. Cloud, Minnesota. Most of the conventional meteorological data referred to in this report will be from a five-year summary of hourly weather observations taken at the Minneapolis-St. Paul Airport (1). Upper air data are obtained at the St. Cloud and St. Paul weather stations. The inversion data referred to in this report have been condensed from an inversion study conducted by the Massachusetts Institute of Technology(2).

F4. CLIMATOLOGY

F4.1 General

The Elk River region has a predominantly continental-type climate, characterized by wide variations in temperature, ample summer rainfall, and scanty winter precipitation. The mean temperature for winter is approximately 18F; for summer, 71F. Temperature extremes range from about -36F to 108F.

From May through September, rainfall averages about 16 in., accounting for approximately 65 percent of the annual precipitation. Winter precipitation consists mostly of snow, averaging about 42 in. a season. Thunderstorms are the principal source of precipitation in spring and summer, and they may be accompanied by destructive hail. Severe storms such as tornadoes, glaze, and blizzards do occur in the State of Minnesota, and their occurrence is frequent enough to warrant consideration.

The movement of high and low pressure systems across the continent, influencing the region, causes frequent and often rapid changes in temperature, particularly during the winter and spring months.

F4.2 Diffusion Climatology

F4.2.1 Winds

No wind records are available from Elk River; however, tabulation of wind data from Minneapolis in this Appendix may be considered representative of the general features for Elk River. These data show that winds from the southwest and northwest sectors are

(1) "Climatology of the United States No. 30-21 (Minneapolis, Minnesota)," U.S. Weather Bureau, Washington, D.C.

(2) Austin, J. M.; "Low-Level Inversions," Massachusetts Institute of Technology, September 1957.

most frequent during the winter months, with an increase of wind frequency from the southeast sector in other seasons, reaching a maximum for that sector in summer. Northwest winds will occur as a result of outbreaks of polar or Arctic air from Canada that move southward into the Great Plains area. Southeast winds will result from a return flow on the rear of a high pressure cell centered over the eastern Great Lakes-New England area; or, from a flow ahead of a storm system that is centered to the south of Minnesota (usually in the Texas Panhandle-Great Plains region).

Monthly wind roses are shown in Figs. F-1-F-4. A condensation of these wind roses into seasonal categories is shown below.

Seasonal Wind Direction Frequencies (%)

<u>Season</u>	<u>N-ENE</u>	<u>E-SSE</u>	<u>S-WSW</u>	<u>W-NNW</u>	<u>CALM</u>
Winter	15	27	31	37	1
Spring	22	27	17	33	1
Summer	17	34	27	21	1
Fall	14	28	26	32	1

F4.2.2 Precipitation

The synoptic regimes influencing this region of Minnesota are such that precipitation winds will be primarily from the northwest-north or east-southeast sectors. Precipitation occurrence concurrent with northwest-north winds in the colder months will be primarily in the form of snow. High winds speeds, occasionally reaching blizzard proportions, will occur during such cold advection regimes. During the late spring and summer months, the predominance of shower and thunderstorm activity affecting the area will be associated with fronts or squall lines moving from the northwest. However, winds during a brief period of thunderstorm passage may blow from any direction at a given locality.

Precipitation occurrence with winds from the east-southeast sectors is due to warm air advection, occurring during the initial stages of storm development in the Great Plains-Texas Panhandle area. During such a situation warm moist air overrides the colder air that exists near the surface, resulting in a steady-type precipitation that is typical of an incipient warm-front regime. Generally, wind speeds are relatively light, and low-level stability may exist, during the initial stages of such warm front patterns, characterized by low stratus and low visibilities.

F4.2.3 Inversions

Inversion data for St. Paul and St. Cloud, tabulated in Tables F-2 (ab) - F-5, indicate that both nocturnal and daytime inversion winds are frequently from the northwest and southeast sectors for summer and winter seasons, respectively; a relatively high frequency of inversion winds from the southwest is also indicated for both seasons.

Deeper and more intense surface-based inversions will occur in winter, attributed to the combined effect of strong radiational cooling and stability intensification from above by subsidence with the Arctic air masses common to this area. From the existing data it can be inferred that longer periods of consecutive hours of inversion durations are more probably during the fall and winter seasons when the area is dominated by subsidence inversions associated with slowly moving Arctic air masses. Conversely, the summer season should indicate a relatively high frequency of nocturnal surface-based inversions that are rapidly dissipated by daytime solar heating. The frequent passage of storm systems affecting the area during the spring months, with a resultant higher percentage of cloudiness and higher wind speeds (Tables F6 and F7), suggests that long durations of inversions in spring would be rare; however, warm air advection from the south and southeast prior to warm front passage may present situations of low-level stability and subsequent reduced visibilities, low stratus or fog, and light steady-type precipitation in spring.

F5. SEVERE WEATHER (TORNADOES)

The region of highest tornado frequency in the United States is south of Minnesota; however, the occurrence of tornadoes in south and central Minnesota is frequent enough to warrant consideration. A study of "Tornado Occurrence in the United States" (3) for the period 1916-1950 ... the following statistics:

Tornado Summary for Minnesota (1916-1950)

<u>Period 1916-1950</u>	<u>Number</u>	<u>Tornado Days</u>
Totals for period	117	92
Yearly average for period	3.34	2.63

(3) "Tornado Occurrence in the United States," Technical Paper No. 20, U.S. Weather Bureau, Washington, D.C., September 1952.

Closer analysis of Weather Bureau tornado data shows that approximately 80 percent of the tornado occurrences in Minnesota for the 35-yr period were reported in the southern half of the state during the months of May through August. No tornadoes were reported for the 35-yr period in the months of December, January, and February; the remaining months show the following distribution of occurrence:

Total Number of Tornado Days and Tornadoes for Minnesota
(1916-1950)

	<u>MAR</u>	<u>APR</u>	<u>MAY</u>	<u>JUN</u>	<u>JUL</u>	<u>AUG</u>	<u>SEP</u>	<u>OCT</u>	<u>NOV</u>
Days	1	4	17	31	21	12	3	1	2
Tornadoes	1	6	19	38	28	19	3	1	2

TABLE F-1

PERCENTAGE FREQUENCIES OF SKY COVER AND WIND SPEED
MINNEAPOLIS, MINNESOTA (1950-1954)
BASED ON HOURLY OBSERVATIONS

	<u>0-3</u>	<u>4-7</u>	<u>8-10</u>	<u>Clam</u>	<u>1-3</u>	<u>4-7</u>	<u>8-12</u>	<u>13-18</u>	<u>>18</u>
Jan	35	8	57	<1	7	27	34	23	9
Feb	40	8	52	1	7	25	32	24	11
Mar	33	8	59	<1	5	19	33	27	15
Apr	36	8	56	1	6	16	27	29	20
May	38	13	49	<1	6	17	23	28	16
Jun	43	14	44	<1	5	19	36	27	13
Jul	51	14	34	1	8	27	37	21	5
Aug	48	14	39	1	9	26	39	21	3
Sep	48	12	40	1	6	25	37	22	8
Oct	52	10	37	1	6	23	35	26	10
Nov	43	9	48	1	5	21	32	26	16
Dec	30	7	63	1	7	21	34	29	9

TABLE F-2

FREQUENCY (Number of Days) OF PRECIPITATION AMOUNTS
MINNEAPOLIS, MINNESOTA (1950-1954)

Precipitation Amounts in Inches

<u>Month</u>	<u>Trace</u>	<u>0.01</u>	<u>0.02-0.09</u>	<u>0.10-0.24</u>	<u>0.25-0.49</u>	<u>0.50-0.99</u>	<u>1.00-1.99</u>	<u>>2</u>	<u>Total number of Days</u>	<u>Days per Month</u>
Jan	53	11	20	7	4	1			96	19
Feb	50	1	12	7	5	4			79	16
Mar	28	8	29	15	10	6	1		97	19
Apr	35	8	13	12	12	3	1		84	17
May	33	6	14	13	11	7	2		86	17
Jun	19	9	15	10	15	4	6	2	80	16
Jul	20	7	12	17	9	4	9		78	16
Aug	28	5	7	15	10	5	3		73	15
Sep	28	11	9	13	7	4	2		74	15
Oct	20	6	11	6	6	1			50	10
Nov	34	4	13	8	4	4			67	13
Dec	53	14	23	16	6		1		113	23

TABLE F-3

PERCENTAGE FREQUENCY OF VISIBILITY, MINNEAPOLIS, MINNESOTA (1950-1954)BASED ON HOURLY OBSERVATIONS

<u>Month</u>	<u>Visibility in Miles</u>		
	<u>0-3/4</u>	<u>1-6</u>	<u>>7</u>
Jan	1	21	77
Feb	3	22	76
Mar	4	19	77
Apr	1	12	88
May	1	5	95
Jun	1	4	95
Jul	1	4	96
Aug	1	6	94
Sep	1	7	92
Oct	1	7	93
Nov	1	10	89
Dec	2	23	75

INVERSION CATEGORIES

<u>Category</u>	<u>Definition</u>
a.	Surface-based inversions; no other inversion below 2 Km.
b.	Main inversion; above surface but below 2 Km; no other inversion existent.
c.	Surface-based inversion concurrent with main inversion.
d.	Inversion based above 2 Km and below 500 mb ($\sim$ 18,300 ft.).
e.	No inversion below 500 mb.

All summer inversion data are obtained from records at St. Paul for the months of July and August in the period 1940-1944.

All winter inversion data are obtained from records at St. Cloud for the months of December and January in the period 1948-1952.

TABLE F-4

PERCENTAGE FREQUENCY OF INVERSION CATEGORIES WITH
WIND DIRECTION GROUPS

SUMMER

Wind Groups (Clockwise)	Time	Cases	Inversion Category (%)				
			a	b	c	d	e
NNE-ENE	0300Z	13	23	15	38	15	11
	1500	14	0	0	36	0	64
E-SSE	0300Z	122	45	22	21	2	9
	1500	57	2	0	58	5	35
S-W	0300Z	43	23	26	16	14	21
	1500	58	2	2	60	3	33
WNW-N	0300Z	53	25	30	25	8	13
	1500	49	2	0	41	24	33
CALM	0300Z	10	80	20	0	0	0
	1500	1	0	0	100	0	0

WINTER

NNE-ENE	0300Z	19	16	5	79	0	0
	1500	18	6	0	94	0	0
E-SSE	0300Z	82	30	15	52	1	1
	1500	90	30	9	60	0	1
S-W	0300Z	65	42	28	30	0	0
	1500	54	32	15	50	2	2
WNW-N	0300Z	129	19	16	62	1	2
	1500	129	16	12	72	0	1
CALM	0300Z	9	56	22	22	0	0
	1500	19	53	21	26	0	0

TABLE F-5

PERCENTAGE FREQUENCY OF INVERSION CATEGORIES (%)

<u>Season</u>	<u>Time</u>	<u>Cases</u>	<u>Inversion Category (%)</u>				
			<u>a</u>	<u>b</u>	<u>c</u>	<u>d</u>	<u>e</u>
Winter	0300Z	306	28	18	52	1	1
	1500	310	24	11	63	0.3	1
Summer	0300Z	243	37	23	21	6	12
	1500	182	2	1	53	10	35

TABLE F-6

AVERAGE HEIGHT (Meters) OF GROUND INVERSION TOPS AND
MAIN INVERSION BASES (Figures in parentheses, cases)

<u>Season</u>	<u>Ground Inv. Top</u>	<u>Ground Inv. with Main Inv.</u>		<u>Main Inv. Base</u>
		<u>Ground Inv. Top</u>	<u>Main Inv. Base</u>	
Winter 03Z	866 (84)	350 (54)	895 (54)	524 (161)
Summer 15Z	936 (75)	418 (35)	1056 (35)	405 (196)
Summer 03Z	329 (90)	232 (56)	1352 (56)	1081 (53)
	15Z 229 (3)	115 (1)	595 (1)	946 (96)

TABLE F-7

PERCENT FREQUENCY OF WEATHER CLASSES WITH EACH INVERSION CLASS

<u>Category</u>	<u>Time</u>	Winter			Summer			
		<u>Precip</u>	<u>Partly Cloudy</u>	<u>Clear</u>	<u>Precip</u>	<u>Cloudy</u>	<u>Partly Cloudy</u>	<u>Clear</u>
<u>a</u>	0300Z	13	42	45	18	16	21	46
	1500*	15	45	40	33	33	0	33
<u>b</u>	0300Z	20	34	46	11	13	13	64
	1500	14	46	40	0	0	0	100
<u>c</u>	0300Z	53	22	25	26	15	32	26
	1500*	36	42	22	19	31	17	33
<u>d</u>	0300Z	100	0	0	29	13	0	57
	1500	100	0	0	11	17	44	28
<u>e</u>	0300Z	50	25	25	36	16	22	26
	1500	67	0	33	25	25	29	22

* Occurred in less than 3% of cases.

Figure F-1

WINTER WIND ROSES
MINNEAPOLIS, MINN
1950-1954

AVERAGE WIND SPEEDS (MPH) FOR SIXTEEN POINT COMPASS

Month	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
Dec	10	10	10	10	11	11	12	13	10	9	8	9	12	13	13	12
Jan	11	10	10	10	10	10	10	11	9	8	7	7	12	13	12	13
Feb	10	12	13	15	10	10	11	12	6	8	8	8	10	13	13	11

Figure F-2

SPRING WIND ROSES
MINNEAPOLIS, MINN.
1950-1954

AVERAGE WIND SPEEDS (MPH) FOR SIXTEEN POINT COMPASS

Month	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
Mar	14	13	14	13	13	12	12	10	8	8	8	10	13	14	13	14
Apr	13	14	13	13	14	13	11	13	11	9	9	9	13	16	14	13
May	11	11	11	11	11	13	13	14	13	11	11	12	14	16	14	12

Figure F-3

SUMMER WIND ROSES
MINNEAPOLIS, MINN.
1950-1954

AVERAGE WIND SPEEDS (MPH) FOR SIXTEEN POINT COMPASS

Month	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
Jun	10	11	10	10	10	11	12	13	14	12	11	12	14	14	13	11
Jul	9	10	9	10	9	9	10	11	11	11	10	8	10	12	11	9
Aug	9	9	10	9	8	8	10	10	10	10	9	9	9	11	11	11

Figure F-4

FALL WIND ROSES
MINNEAPOLIS, MINN.
1950-1954

AVERAGE WIND SPEEDS (MPH) FOR SIXTEEN POINT COMPASS

Month	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
Sep	9	10	8	9	9	10	10	11	11	10	9	10	13	14	12	11
Oct	11	11	10	9	9	9	11	12	12	11	10	10	12	14	13	12
Nov	12	12	10	10	9	10	11	11	11	10	9	8	13	17	15	12

APPENDIX G

RCPA ENVIRONMENTAL MONITORING PROGRAM

G1. GENERAL

From July 15, 1959, to July 15, 1962, an extensive environmental survey program has been performed by the University of Minnesota under a subcontract and by members of the RCPA Radiological Physics Department. The environmental survey program was designed to establish as well as possible the radioactivity levels in the environment immediately around the plant site and within a 15-mile radius. Information and data from the program have been compared with findings of the Minnesota Department of Health, which has been conducting a statewide survey program.

Although the scope of the program has changed from year to year, the sampling points and analyses which have been done at one time or another are listed. The number behind each type of analysis is representative of the total samples analyzed for that particular type of activity. Some samples were not analyzed and were stored for future reference, while other samples were collected even though not scheduled in the program. This helped coordinate the activities of the organizations involved.

G2. MISSISSIPPI RIVER

Surface Water - Samples have been collected from three locations at the cities of Elk River, Anoka, and Minneapolis. A fourth sample was collected about 1000 yards downstream of the plant site. Analyses: Beta, 884; Alpha, 468.

Bottom Sediments - Samples were collected near the surface-water sampling points. Analyses: Beta, 48.

Clams - Samples were collected near the surface-water sampling points. Analyses: Beta, 36; Strontium-90, 36.

Fish - Samples were collected near the surface-water sampling points. Analyses: Beta, 30; Alpha, 26; Strontium-90, 30.

Bottom Fauna - Samples were collected near the surface-water sampling points. Analyses: Beta, 36; Alpha, 36.

Algae - Samples were collected in the river near the plant site. Analyses: Beta, 6; Alpha, 6.

Plankton - Samples were collected in the river near the plant site. Analyses: Beta, 6; Alpha, 6.

G3. VEGETATION

Samples of alfalfa were collected from farms in the Elk River area. For the most part, these came from the same locations as milk samples and were limited to a 15-mile radius around the plant. Analyses: Beta, 40; Strontium-90, 40.

G4. SOIL

Samples from the upper layers of the soil were collected within a 10-mile radius of the plant. Analyses: Strontium-90, 30.

G5. ANIMALS

Rabbits were collected from the immediate area of the plant. Analyses: Cesium-137, 12; Iodine-131, 12; Potassium-40, 2; Beta, 2.

G6. MILK

Samples were collected from the Elk River area and two reference stations located approximately 80 miles south and 60 miles west of the site. Analyses: Strontium-90, 104; Cesium, 32.

G7. PRECIPITATION

Samples were collected at Elk River and the two milk samples reference stations. Analyses: Alpha, 68; Beta, 68.

G8. GROUND WATER

Samples of ground water were obtained from private and public wells in the Elk River area up to a distance of 15 miles from the site. Both deep and shallow wells were selected. Analyses: Beta, 167; Alpha, 101.

G9. RIVERS

Three rivers other than the Mississippi were analyzed. These included the Rum, Crow, and Elk. Analyses: Beta, 48; Alpha, 22.

G10. LAKES

Lakes located northeast, northwest, southeast, and southwest of the reactor site were sampled. These lakes are approximately 10 miles from Elk River. Analyses: Beta, 9; Alpha, 8.

G11. DRAINAGE DITCH

Samples were collected from the drainage ditch fronting the RCPA headquarters building. This ditch was fed by a natural spring and county ditch No. 10. Analyses: Beta, 82; Alpha, 8.

G12. AIR

Samples of air particulates were collected northeast, northwest, southeast, and southwest of the site at distances of approximately 10 miles, at the reactor site, and at the University of Minnesota campus, 50 miles from the site. Analyses: Beta, 1512; Alpha, 1427.

G13. EXTERNAL BETA-GAMMA RADIATION

G-M tubes located on the roof of the RCPA headquarters building have measured and recorded background radiation levels on the plant site. Continuous recorded data in counts per minute have been accumulated over the past years.

During the operation of the Elk River Reactor, portions of the past environmental survey program will be continued. Air samples will be taken at the same sampling points; Mississippi River water samples will be collected; milk sampling will be done in the Elk River area; precipitation collected; and animals will be checked.

The scope of the program will change periodically, based on data obtained and the influence of outside activities such as nuclear bomb testing.

Future environmental programs for the reactor will be coordinated with the statewide radiological survey performed by the State of Minnesota Department of Health.

APPENDIX H

BURNOUT HEAT FLUX IN ERR

H1. SUMMARY

The consequences of exceeding the burnout heat flux in a nuclear reactor are so severe that it is imperative to maintain the maximum heat flux safely below the burnout point at all times. No satisfactory way to predict the burnout point from theoretical considerations exists. Therefore it becomes necessary to rely on experimental results. These results are often correlated by empirical formulas, and these formulas are seldom valid beyond the range of the data used in each correlation. A critical evaluation of all the available data and the correlations is thus necessary in order to arrive at a good estimate of the burnout heat flux for ERR conditions. This evaluation is made to determine the heat flux safety factor of ERR at 58.2 Mwt.

The burnout heat flux at the conditions of the ERR core is estimated to be 1.0×10^6 Btu/(hr)(ft²). The maximum heat flux in the core at 58.2 Mwt is 0.332×10^6 Btu/(hr)(ft²). Hence the safety factor on burnout is 3.0. This value is believed to be more than adequate to assure safe operation.

H2. DISCUSSION

The application of all available correlations to conditions estimated for the maximum flux point in the core gives the burnout heat flux values in Table H-1. The spread of these values is large, and it is not readily apparent which values are reliable. Accordingly, a search was made for the experimental values of burnout heat flux reported in the literature in order to find experiments most closely resembling the expected reactor conditions. None of the experiments satisfactorily represents these conditions. The data closest to the desired parameters are given in Table H-2.

Steam quality increases with the height in each flow channel, and as a result the burnout heat flux decreases with the height. Because of this effect it is conceivable that the point of closest approach to burnout may not coincide with the maximum heat flux point but may be displaced upward. In order to establish the minimum ratio of burnout heat flux to reactor heat flux, the burnout flux was calculated by means of the Bettis correlation, as a function of steam quality, and was related to core height. These results, together with reactor heat flux and the burnout safety factor (the ratio of burnout to actual heat flux), are shown in Fig. H.1. For the calculated conditions, the displacement of the minimum safety factor point from the maximum heat flux point is too small to be noticeable.

TABLE H-1

BURNOUT HEAT FLUX VALUES PREDICTED FROM CORRELATIONS

<u>Correlation</u>	<u>Predicted Burnout Heat Flux (in millions of Btu/(hr)(ft²))</u>
Bernath (Ref. 1)	0.88
Griffith (Ref. 3)	2.63
KAPL	
design equation* (Ref. 10)	0.33
Reynolds (Ref. 6)	8.29
Sonnemann (Ref. 7)	0.24
Bettis: (Refs. 2 and 10)	
Equation V-2	
Old design equation*	0.90
Equation VI-1	
Round tube correlation, 2000 psi	1.57
Equation VI-3	
Rectangular channel correlation, 2000 psi	2.25
Equation VI-12	
Rectangular channel correlation, lower pressures	2.13
Equation VI-13	
Rectangular channel correlation, lower pressures, design equation*	1.57

* "Design equation" means a correlation above which substantially all of the experimental data is intended to fall. The other correlations are intended to give the best representation of the data.

TABLE H-2

COMPARISON OF REACTOR CONDITIONS AND EXPERIMENTAL BURNOUT RESULTS

Source of Data	Burnout Heat Flux $\times 10^{-6}$ (Btu/(hr)(ft ²))	Pressure (psia)	Mass Velocity $\times 10^{-6}$ lb/(hr)(ft ²)	Channel Shape	Channel Equivalent diameter (ft)	Ratio of Length to Diameter	Sub-cooling at inlet (°F)	Enthalpy at Burnout Point Btu/lb	Steam Quality at Burnout Point	Distribution of Heat Generation
Elk River	1.0 (Estimated)	936	1.04	*	0.0944	53	3	541	0.013	(see Fig. H.1)
Ref. 4	0.88	1000	0.93	Round	0.0188	109	434	555	0.019	Uniform
Ref. 4	1.02	1200	1.10	Round	0.0119	21	111	599	0.045	Uniform
Ref. 9	0.81	600	1.03	Round	0.0150	65	7	611	0.191	Uniform
Ref. 6	1.82	1053	1.91	Round	0.0150	50	130	578	0.050	Uniform
Ref. 6	1.93	1000	2.88	Round	0.0150	65	137	554	0.018	Uniform
Ref. 8	0.67	830	1.10	Rectangular	0.0162	153	60	763	0.365	Uniform
Ref. 5	0.42	830	0.82	Rectangular	0.0162	153	60	716	0.294	Cosine

* Space between round tubes placed on a square pitch.

H2.1 BURNOUT CORRELATIONS

The predictions of burnout heat flux by the many correlations that have been proposed differ widely, as may be seen from Table H-1. One of the principal reasons for this spread of values is that each correlation was derived on the basis of a relatively narrow range of data and, when applied to any specific set of conditions, usually includes a varying degree of extrapolation. In addition, the correlations are not all on the same basis. There is disagreement among the various authors as to the definition of burnout heat flux. Some authors give correlations intended to provide the best fit of the data, while others present more conservative "design equations" in which substantially all experimental points are intended to fall above the correlating curve. The most common definition of burnout heat flux is that of the maximum heat flux in nucleate boiling, i.e., the heat flux at the transition from nucleate to film boiling. The Bettis workers (Ref. 10), however, define burnout heat flux as the heat flux in nucleate boiling at which deviation from the linear relationship between heat flux and wall-fluid temperature difference begins. This definition is reported to result in experimental values no more than five percent lower than those for the more usual definition of burnout flux.

All of the reported experimental burnout data have been taken for flow either on the inside of round tubes or for flow within channels of rectangular cross-section. In the ERR core, the coolant flows on the outside of round tubes; therefore, the use of any of the available data is gross extrapolation, even when other parameters such as pressure, equivalent diameter, mass velocity, etc. are all at their desired values. Another difference between experimental and reactor conditions is heat flux distribution. Almost all of the data were taken on uniformly heated test sections; only a few were taken with a cosine heat flux distribution in the axial direction. The flux distribution in the Elk River reactor is closer to a cosine shape than to a uniform shape, but actually fits neither of these patterns. The difference in heat generation distribution is believed not to result in as serious an extrapolation as the other factors, since burnout behavior is believed to be primarily a function of conditions at the burnout point, and only to a much lesser degree a function of upstream and downstream factors.

The following is a discussion of each correlation considered:

Bernath (Ref. 1)

This correlation is based on round tube results in the pressure range of 500-3000 psi and for operation with subcooling at the burnout point. The form of the correlation permits extrapolation to the desired quality range. The predicted value appears reasonable.

Griffith (Ref. 3)

This correlation was developed from pool boiling data for a variety of fluids. The data on which it is based cover sufficiently wide pressure, steam quality, and velocity ranges. No explanation for the high value predicted for reactor conditions is apparent.

KAPL (Ref. 10)

This design equation is intended to be conservative. It represents a line which is two standard deviations below the mean of the data. In the quality range, the correlation states that burnout heat flux is a function of mass velocity only; clearly, many parameters of importance are overlooked. The low values predicted for ERR conditions from this equipment should therefore be given little weight.

Jens and Lottes (Ref. 4)

This correlation applies to subcooled boiling only. Its mathematical form does not permit extrapolation to the quality range. Hence no estimate of burnout heat flux for ERR conditions can be presented using the Jens-Lottes equation.

Reynolds (Ref. 6)

This correlation is based on data for 0.18 and 0.31-in. ID tubes. Other parameters are in the desired ranges. The effect of tube diameter was introduced into the correlation by means of dimensional analysis but is largely untested. The equivalent diameter for the ERR flow channels is much larger than the test section diameters in Reynolds' work. The unreasonably high value of burnout heat flux predicted from Reynolds' correlation probably is due, at least in part, to erroneous representation of the effect of diameter in the correlation.

Sonnemann (Ref. 7)

This correlation was developed while the author was working at Bettis and is based primarily on Bettis data. It is therefore surprising that the order of magnitude of the predicted value does not agree with that predicted by the various Bettis equations. In the Bettis equations the effect of tube diameter is relatively weak, but in Sonnemann's correlation it is much stronger. It is probable that the poor prediction from this correlation is, as for Reynolds' correlation, due to a spurious effect of equivalent diameter.

Bettis (Refs. 2 and 10)

The work at Bettis constitutes the most thorough analysis of burnout available at the present time. The Bettis correlations are based on a large body of data, including practically all of the data on which the various other correlations are based, and on many experiments performed at Bettis. Separate correlations are presented for round tubes and for rectangular channels. Both these correlations were applied to ERR conditions by use of the customary equivalent diameter for the non-circular geometries. The principal correlations are based on a pressure of 2000 psi, and other equations are recommended for rectangular channels at pressures between 600 and 1850 psi. The low pressure equations are similar in form to the principal correlations on which the most effort has been expended. No equation is presented for round tubes at the lower pressures because the authors feel insufficient data are available for this purpose. In most cases the authors recommend multiplying the correlation equations by a factor of 0.65 to obtain "design equations." If this is done, there must be no allowance in the burnout safety factor for data spread. It is believed that the values of burnout heat flux predicted by the various Bettis equations should be given considerably more weight than the values predicted by the various other equations. Accordingly, the Bettis correlation, Equation VI-1, was used for the calculation of burnout heat flux as a function of distance from core inlet (bottom) in (Fig.H.1).

In arriving at the estimate of burnout heat flux for ERR conditions, principal weight was given to the experimental results in Table H-2, and lesser weight was given to the correlations. The spread in the results of Table H-2, although large, is still smaller than the spread of the correlation values in Table H-1. This spread occurs although all the correlation values are intended to apply to the experimental values for a series of different reactor conditions. It is concluded that the Bernath and Bettis correlations give the best order-of-magnitude predictions for ERR conditions and that the remaining correlations are wholly inapplicable.

H2.2 BURNOUT SAFETY FACTOR

The estimated burnout heat flux for ERR conditions is $1.0 \times 10^6 \text{ Btu}/(\text{hr})(\text{ft}^2)$. This value represents the best estimate that can be made at this time. Safety factors must be applied between the burnout heat flux and the peak reactor heat flux to account for uncertainties in both these fluxes. In the Bettis work, multiplication of correlation heat flux values by 0.65 is recommended to account for a -35 percent spread in the data. Following this practice, the lower limit of burnout heat flux for ERR conditions is $0.65 \times 10^6 \text{ Btu}/(\text{hr})(\text{ft}^2)$. This value still permits a safety factor of 1.96 to account for uncertainties in reactor heat flux only. Uncertainties in reactor heat flux arise from uncertainty in the ratio of peak

to average reactor flux, reactor power drift or error from the nominal power, and reactor transient effects. The safety factor of 1.96 for all these factors is believed to be more than adequate. Consequently, future moderate increases of reactor power are not necessarily precluded by burnout considerations.

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