11-16-59

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

COMMONWEALTH EDISON COMPANY
DOCKET NO. 50-10

LICENSE

License No. DPR-2, as amended

1. This license applies to the dual-cycle, boiling water type reactor designated by Commonwealth Edison Company (hereinafter referred to as "Commonwealth Edison") as the "Dresden Nuclear Power Station" (hereinafter referred to as "the facility") which is owned by Commonwealth Edison and located in Grundy County, Illinois, and described in Commonwealth Edison's application attested March 31, 1955 and amendments to the application attested June 24, 1955, February 1, 1956, March 9, 1956, March 15, 1956, June 6, 1957, June 12, 1957, July 26, 1957, September 3, 1957, November 5, 1957, December 17, 1957, May 26, 1958, June 5, 1958, August 25, 1958, December 26, 1958, December 30, 1958, January 6, 1959, February 6, 1959, April 3, 1959, and May 15, 1959, (hereinafter collectively referred to as "the application") and for which Construction Permit No. CPPR-2 was issued by the Atomic Energy Commission (hereinafter referred to as "the Commission") on May 4, 1956 and amended on March 31, 1958.

2. Subject to the conditions and requirements incorporated herein, and subject to the order of the Commission in reference to operations of the nuclear facility at the 315 megawatt (thermal) power level the Commission hereby licenses Commonwealth Edison:

   a. Pursuant to Section 104(b) of the Atomic Energy Act of 1954, as amended, (hereinafter referred to as "the Act") and Title 10, CFR, Chapter 1, Part 50, "Licensing of Production and Utilization Facilities", to possess and operate the facility as a utilization facility;

   b. Pursuant to the Act and Title 10, CFR, Chapter 1, Part 70, "Special Nuclear Material", to receive, possess and use 2280 kilograms of contained uranium 235 as fuel for operation of the facility; and

   c. Pursuant to the Act and Title 10, CFR, Chapter 1, Part 30, "Licensing of Byproduct Material", to possess, but not to separate, such byproduct material as may be produced by operation of the facility.

3. This license shall be deemed to contain and be subject to the conditions specified in Section 50.54 of Part 50 and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and rules, regulations and orders of the Commission now or hereafter in effect; and is subject to any additional conditions specified or incorporated below:

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a. Operating Requirements

(1) Commonwealth Edison shall not operate the facility at a steady state power level in excess of 630,000 kilowatts (thermal), and shall not operate the facility in excess of 315 megawatts (thermal) without further Order of the Commission.

(2) Subject to the provisions of this paragraph 3., Commonwealth Edison shall operate the facility only in accordance with the design and performance specifications and operating limits and procedures described in the application and Appendix "A" to this license.

(3) In any case where the procedures or specifications described in the application are not consistent with the requirements of this paragraph 3. and Appendix "A" to this license, the requirements contained in this paragraph 3, and Appendix "A" shall govern.

(4) Commonwealth Edison shall not change or modify the design or performance specifications or operating limits or procedures described in Appendix "A" to this license until after a description and hazards evaluation report of the proposed change has been filed with the Commission by Commonwealth Edison and the Commission shall have authorized such change in writing.

(5) Except with respect to the specifications, limits and procedures contained in Appendix "A", Commonwealth Edison may change or modify the design or performance specifications or operating limits or procedures described in the application only in accordance with the following procedures:

Commonwealth Edison shall provide the Commission with a description and hazards evaluation report of the proposed change. If, within fifteen days after the date of acknowledgment by the Division of Licensing and Regulation of receipt of such report, the Commission does not issue any notice to Commonwealth Edison to the contrary, Commonwealth Edison may make such change without further approval. If, within fifteen days after the date of acknowledgment by the Division of Licensing and Regulation of receipt of such report, the Commission notifies Commonwealth Edison that the hazards involved may be greater than or materially different from those analyzed in the Hazards Summary Report, or that the proposed change involves a material alteration of the facility, the change shall not be made until after such change has been authorized in writing by the Commission. If a license amendment is necessary to authorize the proposed change, the report submitted by Commonwealth Edison shall be deemed to constitute an application for a license amendment. As used in this paragraph 3., a proposed change

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shall be deemed to involve hazards which may be "greater than, or different from, those analyzed in the Hazards Summary Report" if (1) the probability of any type of accident analyzed in the Hazards Summary Report might be increased, or (2) the possible consequences of any type of accident analyzed in the Hazards Summary Report might be increased, or (3) such change might create a credible probability of an accident of a type different from, and the possible consequences of which would not be of a lesser magnitude than each of, the accidents analyzed in the Hazards Summary Report. The "Hazards Summary Report" as used in this paragraph 3. is defined as the "Enclosure Section" attested June 12, 1957, the "Preliminary Hazards Summary Report" attested September 3, 1957 and amendments 1, 2, 3 and 4 thereto respectively attested May 26, 1958, August 25, 1958, December 30, 1958, and February 6, 1959 and the "Operating Procedures and Emergency Plans" attested June 5, 1958 submitted by Commonwealth Edison.

b. Records

In addition to those otherwise required under this license and applicable regulations, Commonwealth Edison shall keep the following records:

(1) Reactor operating records, including power levels and periods of operation at each power level.

(2) Records showing the radioactivity released or discharged into the air or water beyond the effective control of Commonwealth Edison as measured at the point of such release or discharge.

(3) Records of emergency shutdowns, including reasons therefor.

(4) Records containing a description of each change made pursuant to paragraph 3.a.(5) hereof.

c. Reports

(1) Commonwealth Edison shall make an immediate report in writing to the Commission of any significant indication or occurrence of an unsafe condition relating to the operation of the facility.

(2) At least seven days prior to commencing each of the initial operating phases listed below, Commonwealth Edison shall file with the Commission a report describing the testing program intended to be conducted during that respective phase and the general testing methods to be used. Within thirty days of completion

of each operating phase, Commonwealth Edison shall submit a report of the results of the tests and operation conducted pertinent to safety, including a description of changes made in the facility design, performance characteristics and operating procedures. The operating phases referred to above are (a) initial loading and criticality, (b) atmospheric pressure tests, (c) low power (up to 60 MW thermal power) tests, and (d) rates power tests.

(3) Commonwealth Edison shall submit to the Commission a quarterly report for each quarter during the first year after completion of the rated power tests. Such quarterly reports shall be filed within 30 days after the end of the quarter covered by the report. Thereafter Commonwealth Edison shall file an annual report. The first such annual report shall be filed within thirteen months after the filing of the fourth quarterly report referred to above. Each report filed under this paragraph shall include a description of operating experience pertinent to safety and changes in facility design, performance characteristics and operating procedures during the reporting period.

4. Pursuant to Section 50.60 of the regulations in Title 10, Chapter 1, CFR, Part 50, the Commission has allocated to Commonwealth Edison for use in the operation of the facility 9388 kilograms of uranium 235 contained in uranium enriched to approximately 1.5% and 1.7% in the isotope uranium 235. Estimated schedules of special nuclear material transfers to Commonwealth Edison and returns to the Commission are contained in Appendix "B" which is attached hereto. Shipments by the Commission to Commonwealth Edison in accordance with column (2) in Appendix "B" will be conditioned upon Commonwealth Edison's return to the Commission of material substantially in accordance with column (3) of Appendix "B".

. This license is effective as of the date of issuance and shall expire at the date to be determined after consideration of the operations at the 315 megawatt (thermal) power level.

6. This license is subject to the limitations set forth in the "Intermediate Decision and Order for Limited Power Operations" and the "Order Extending Period of Time for Limited Power Operations" issued by the Presiding Officer on September 26, 1959 and November 5, 1959, respectively, which provided for operations to the extent of, but not in excess of, a power level of one megawatt (thermal) for a period of time which will expire on December 10, 1959.

7. In accordance with the Supplemental Intermediate Decision issued on November 12, 1959, In the Matter of Commonwealth Edison Company, when Commonwealth Edison has attained or reached a power level of 315 megawatts or 50% of rated power, Commonwealth Edison shall file with the Commission a report of such 315-megawatt power level, including among other pertinent aspects a description of all existing and prior operating conditions and characteristics of the facility.

FOR THE ATOMIC ENERGY COMMISSION

R. L. Kirk
Acting Director
Division of Licensing and Regulation

Attachments:
Appendix "A"
Appendix "B"

Date of Issuance: November 16, 1959

Appendix "A"

A. INTRODUCTION

The following are the principal design and performance specifications and operating limits and procedures of the Dresden Nuclear Power Station pertaining to safety.

Sections B and C set forth the design and performance specifications and operating limits and principles.

Sections D, E, and F specify the basis for the initial loading and critical testing operations, and the limitations to be observed during start-up, power operation, and refueling and maintenance operations. In these sections, as well as in Section B, where maximum or minimum limits are not given specifically, the values given are "design" values which are subject to normal manufacturing and other tolerances.

B. DESIGN FEATURES

1. Reactor Vessel

The reactor vessel is a vertical cylindrical pressure vessel, with dished top and bottom heads, made of low alloy steel and clad inside with stainless steel. The vessel was designed, built, and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section I. Design parameters for the vessel include:

Inside height, including heads	40 ft 9-5/8 in
Inside diameter	12 ft 2 in
Design pressure	1250 psig
Design temperature	650° F

2. Nuclear Core

Maximum core diameter (circumscribed circle)	129 in
Maximum active cold fuel lengths	107-1/8 in
Maximum number of fuel assemblies	488

3. Fuel

Each fuel assembly consists of 36 vertical fuel rods (except for occasional special assemblies such as instrument bearing assemblies which will have 34 or 35 fuel rods plus the thimbles of approximately the same dimension of the clad fuel rods), each of which is made up of solid cylindrical pellets of uranium dioxide, enriched in uranium-235 to a maximum of 1.5%, and clad in Zircaloy-2. Each fuel rod is composed of four separately clad segments. Pertinent fuel design parameters in the cold condition are:

Fuel pellet diameter	0.493 in
Fuel pellet length	
Regular fuel pellets	0.625 in
Segment end pellets	0.500 in
Outside diameter of cladding	0.56[illegible] in
Cladding wall thickness	0.030 in
Fuel pellet density averaged over a fuel segment	
Minimum	94% of theoretical
Cross-sectional center-to-center distance between fuel rods	0.710 in

4. Control Rods and Drives

The control rods, 80 in number, are made of boron (initially about 2%) stainless steel, have a 6.5-inch cruciform cross section, and travel between fuel channels. The center-to-center distance between control rods is 9.96 inches, and all 80 control rods are located within an 8 foot 3 inch diameter cylindrical space in the central region of the core.

The drive mechanism for both normal operation and scram is an all-hydraulic system. Two independent sources of hydraulic pressure are available for scramming the control rods. These are:

a. The accumulator pressure, which will be available and effective under all conditions of reactor operation except "Shutdown" (at which time interlocks prevent any rod withdrawal), and

b. The reactor pressure, if this is greater than about 700 psig and accumulator pressure falls below reactor pressure.

The drives are mounted on the bottom of the reactor vessel, and withdraw the control rods below the core. Upward movement of the control rods, into the core, decreases reactivity. The inserted position of the control rods is determined by a locking device which provides 12 discrete equidistant positions for all rods except those in the outer ring. The last withdrawal step for these is slightly shorter than the others. Only one rod at a time can be withdrawn from the core. Rods may be inserted into the core singly or all may be scrammed together.

Active length of control blades	8 ft 6 in
Velocity for normal insertion or withdrawal	6 in/sec
Maximum time from receipt of scram signal to:	
10% control rod travel	0.6 sec
90% control rod travel	2.5 sec
Maximum number of control rods per accumulator	3 non-adjacent rods

Frequent and thorough periodic checks will be made to assure the proper functioning of the control rod drive system.

5. Liquid Poison System

System actuation	Manual
Maximum time after actuation until poison begins to enter core	20 sec
Minimum weight of boron in system (present as sodium pentaborate in solution)	400 lbs.
Minimum poison worth	0.20 Δk

6. Steam Supply System

Besides the reactor itself, the steam supply system comprises a steam separating drum, four secondary steam generators and recirculating pumps, an emergency condenser, unloading heat exchangers, and the necessary piping and accessories for these components. The plant may on occasion operate with one or two of the secondary steam generator loops bypassed, as long as the operation is stable and meets any other specifications imposed.

The emergency condenser consists of two separate tube bundles of equal capacity in a common shell. Each tube bundle can be started automatically by the appropriate reactor safety system controls (discussed in item B.9 below), and may be also started or stopped manually by remotely controlled valves. Pertinent limits placed on these steam supply system components include the following:

Design pressure of steam drum, secondary steam generators, and primary side of unloading and emergency heat exchangers	1250 psig
Design pressure of primary system piping	1150 psig
Minimum capacity of emergency condenser	6% of rated reactor thermal power
Minimum cooling water stored in emergency condenser	30,000 gal
Minimum emergency condenser cooling period (after scram) without operator attention	8 hrs

7. Main Condenser

The condenser is capable of handling the normal steam flow from the turbine plus the heater drains or bypassed extraction steam. As a heat sink for the reactor, the condenser will handle a flow of 1,900,000 lbs/hr of bypassed primary turbine steam. The condenser can handle this steam flow without desuperheating spray water, in which case the steam temperature entering the condenser will reach a maximum of about 300°F.

The following condenser vacuum trips will be operative except during "Startup" or "Shutdown" conditions:

Reactor scram if vacuum falls to or below	22 in. Hg
Turbine bypass valves close if vacuum falls to or below	7 in.Hg

8. Waste Disposal Systems

a. Solid Wastes

Solid wastes containing radioactive materials include filters, defective equipment, and other miscellaneous trash. Such material will be stored in accordance with AEC regulations (10CFR-Part 20) which may involve storage in an underground concrete storage vault at the site. When feasible such material may be compacted before storage. Spent contaminated resins are sluiced to an underground tank for indefinite storage. Sluice water is decanted after the resins have settled.

b. Liquid Wastes

Equipment is provided to treat radioactive liquid wastes by decay in storage, long-term underground storage, filtration, neutralization, demineralization, or evaporation. The treatment will most often result in water which can be re-used in the plant or released to the river. Batch sampling will be used to determine whether permissible limits can be met for any liquid process wastes to be released. No disposal of these wastes to ground will be made.

The large quantities of river water used for equipment cooling will be monitored before return to the river to detect any process leakage into the cooling flow. This flow may be used to dilute process liquid wastes to below permissible limits for release to the river. The release of liquid wastes shall conform to the provisions of 10 CFR, Part 20.

c. Airborne Wastes

Radioactive airborne wastes are discharged from a 300-foot-high stack. There will be continuous monitoring of the total stack flow and air-ejector flow. A holdup time is provided in both the gland seal exhaust system and the air-ejector exhaust system. The air-ejector monitor automatically initiates closing of the discharge valve on the air-ejector holdup system if the measured rate of discharge of Xe ^{138} exceeds 2×10^6 uc per second, the measurement being made after two minutes decay (travel time in the system from the reactor core to point of measurement). The reactor will be manually shutdown when the stack discharge rate of noble fission gases exceeds 7×10^5 uc per sec.

d. General

Commonwealth Edison shall not release into air any concentration of radioactive material which will result in exposure to concentrations at ground levels in any unrestricted area as that term is defined in 10 CFR Part 20 in excess of the limits specified in Appendix B, Table II of 10 CFR Part 20. For purposes of this limitation concentrations may be averaged over periods not greater than one year.

9. Safety System

The reactor safety system will include monitoring devices external and internal to the reactor. The out-of-core system utilizes two paralleled safety channels, each channel with its separate power supply and sensing elements. Both are of fail-safe design throughout (that is, de-energizing will cause a scram), and both must de-energize to cause a scram. Table I, below, lists the external safety circuit sensors, their maximum or minimum trip settings, the number of sensors of each type in each channel, the coincidence reading feature, and the automatic functions performed in addition to a scram. In addition to the trips listed in Table I, an automatic scram is provided in the event of power supply failure, by virtue of the fail-safe design used. A manual scram control is available to the operator also, and an additional manual control scrams the reactor and closes the sphere isolation valves.

The in-core monitoring system is to provide in required locations indication of local power, automatic scram at not more than 125% of rated local power, and alarm at a selected level below the scram setting. The automatic scram may be actuated by coincidence of signals from two or more monitors, provided that the coincidence arrangement does not have the effect of leaving unmonitored a core region exceeding in size the limits specified below.

Whenever the reactor is operating at a high power level (as used in this paragraph 9."high power level" shall mean a thermal power level exceeding 315 MW or 50% of rated local power), there shall be a sufficient number of operating in-core local power monitors to meet the following conditions:

a. There will be at least three monitored horizontal layers reasonably evenly spaced in the region of the active core bounded by planes 1 ft below its top and 1 ft above its bottom.

b. Within the central 8.5 foot diameter vertical cylindrical core volume, no two adjacent horizontal layers may be without an operating local power monitor in any vertical cylindrical core volume that exceeds 4 ft. in diameter.

c. The in-core local power monitors will be so located that when the neutron flux within the core is purposely distorted by withdrawal of adjacent control rods from any region of the core, this distortion shall be detected by at least two operating in-core local power monitors. Commonwealth Edison shall conduct experiments to demonstrate compliance with this requirement.

d. There will be at least 32 operating in-core local power monitors present in the core.

An operating in-core local power monitor is defined as one which has a response time of less than one second, indicates approximately linear response to changes in local power and does not display erratic changes in calibration. Periodic tests will be conducted to demonstrate the operating condition of the in-core local power monitors.

For operations at a power level less than high power level, as defined above, the in-core local power monitoring system is not required provided that at least five of the six external power range neutron flux monitors are in operation and are so connected that indication of reactor thermal power exceeding 62.5% of the authorized power level by any one monitor will scram the reactor.

Rated local power is defined as the condition corresponding to a peak heat flux of 280,000 Btu/(hr)(sq.ft.) in the vicinity of the point monitored.

A four-position safety system selector switch is provided to bypass those scram trips which are not required or desirable under some conditions. The bypasses for the external system are indicated in the "Remarks" column of Table I. The in-core local power monitoring system may be bypassed in the "shutdown" position and the "refuel" position. The four positions for the selector switch are:

a. "Start" position, to allow startup before full condenser vacuum is established;

b. "Run" position, for normal plant operation with period trip bypassed;

c. "Refuel" position, to allow some control rods to be withdrawn for safety during refueling if the high neutron flux sensors have been set to trip at a low value;

d. "Shutdown" position, to allow ventilation of the reactor enclosure while testing or maintaining the safety system with the reactor shutdown. Control rods cannot be withdrawn in this position.

10. Radiation-Type Reactor Instrumentation

In addition to the external and internal core neutron flux channels and the period channels that are a part of the reactor safety system, there are provided two indicating startup channels and a battery-operated neutron flux channel.

11. Reactor Enclosure

The enclosure housing the reactor and the steam supply system is a spherical steel shell, 190 feet in diameter, with the equator approximately 56 feet above ground level. From the enclosure, the minimum off-site distance is one-half mile to:

a. Skinner Island in the Kankakee River;

b. The navigation channel in the Illinois River; and

c. The land boundaries of the site.

The enclosure was designed, built, and tested in accordance with the ASME Boiler and Pressure Vessel Code. Pertinent design parameters for this vessel are:

Design Pressure	29.5 psig
Design Temperature (coincident with design pressure)	325°F
Maximum Wind Velocity	110 mph
Horizontal Acceleration	3.3% gravity
Maximum Leakage Rate at 37 psig	0.5%/day

All normally open lines penetrating the enclosure shell through which leakage could credibly occur in the event of the "maximum credible accident" are provided with check valves or isolation valves which close without operator attention. Table I indicates the scram signals which initiate closure of the automatic isolation valves. The isolation valves on the primary steam lines, which are not closed from any scram signal, will close automatically from low reactor pressure -- a condition that would be encountered in any system rupture approaching the severity of the postulated "maximum credible accident." All the primary isolation valves are backed up by additional valves which can be operated from positions that are tenable after the accident. Normally closed lines penetrating the enclosure are protected against being opened during operation, or in potentially hazardous non-operating situations, by interlocks and/or operating rules.

The enclosure is provided with a post-incident cooling system, for use in the event of a serious primary system rupture, to aid in the reduction of enclosure pressure and leakage. This system is designed for a heat removal capacity of at least 30×10^6 Btu/hr at an enclosure internal temperature of 256°F.

Operation of this cooling system will be automatically initiated by the scram signal from high sphere pressure after a time delay not to exceed ten minutes. The reactor operator may also manually initiate operation of this system or override the automatic initiation signal. The system will be maintained in operable condition at all times the primary system is pressurized. Periodic tests will be conducted to demonstrate the proper functioning of the post-incident cooling system.

About one year after initial power operation, a leakage test of the enclosure will be conducted at a low pressure (less than 10 psi) to determine the reliability of penetrations. (It is expected that this test will probably be made during the time the plant is down for the first major inspection of the turbine which is also expected to be done after about a year's operation.)

After a study of the results of the above leakage testing and discussion of these results with the AEC, the future program of testing will be agreed to.

Control Room

The control room is shielded from all power plant equipment and from the reactor enclosure so as to be tenable even in the event of the "maximum credible accident." The minimum thickness of concrete shielding betwee the control room and enclosure for direct-line radiation is the equivalent of five feet.

C. OPERATING PRINCIPLES

The basic operating principles that will be adhered to in the operation of the Dresden plant are as follows:

a. Before being placed into regular service, the plant will be proved out in a comprehensive controlled test and initial operation program, as indicated in Sections D and E.1, below.

b. Operation and control of most of the power plant equipment will be centralized in the control room.

c. The control room will be manned at all times by at least two operators, one of which shall be licensed.

d. While most operating and control functions are initiated in the control room, operators may perform some functions at remote operating panels and valve racks -- at the direction of the control room staff or with their prior knowledge.

e. Startup, normal shutdown, and all other repetitive operations will be performed in accordance with specific check lists.

f. Maintenance of much of the equipment outside the reactor shielding may be undertaken by contact methods and without overall plant shutdown. Plant shutdown and semi-remote methods will be employed as necessary.

g. All tests and routine maintenance of protective devices and power plant equipment will be done in accordance with prescribed schedules.

h. Radiation monitoring by fixed or portable instrumentation will be provided for entry to all radiation zones.

i. All personnel leaving a contaminated radiation zone, and equipment being removed from such zones, will be appropriately surveyed to assure control of contamination.

j. Irradiated fuel is to be moved from the reactor to storage, under water, by semi-remote methods.

k. Enclosure isolation provisions (i.e., air locks and equipment hatches closed, and instrumentation operating to close the automatic isolation valves if it should become necessary) will be in effect during all periods of reactor operation, including startup and shutdown operation, and during any operation involving insertion or removal of fuel assemblies in the core or withdrawal of control rods when the reactor head is off.

l. Operation of the radioactive waste-handling system will be done in such a manner that it will be unlikely that the disposal of radioactive materials will result in the exposure of any persons on or off the plant to radiation in excess of the permissible limits.

m. The plant is so protected at all times by automatic safety devices that no single operator error or reasonably conceivable combination of operator errors could cause a severe accident.

n. All significant unexpected incidents, unsafe acts, or incidents of excessive exposure to radiation will be investigated to effect procedures to prevent recurrence.

o. In the event of any situation which may compromise the safety of continued operation, it will be the required procedure to shut the plant down as quickly as the situation calls for, and to take other planned emergency actions to protect persons and property.

D. INITIAL LOADING AND CRITICAL TESTING

1. General

On successful completion of all pre-operational checks necessary for safe loading, the reactor will be loaded to initial critical and then to the proper loading for power operation. Between these two loading steps a number of tests will be run and measurements made to check the calculated nuclear parameters. This phase will be conducted at atmospheric pressure with the reactor vessel lid removed, and at approximately ambient temperature. Items which will receive particular attention during this phase of operation, or that will be carried out in a manner different from normal refueling, are discussed below.

2. Instrumentation

A minimum of one neutron-sensitive chamber per grid quadrant will be installed in the reactor before the initial loading starts. A neutron source of sufficient strength to provide appreciable readings at all times on the neutron-sensitive chambers will be installed, and the chamber-source geometric relationships will be maintained as the loading progresses so that the chamber sensitivity will always be sufficient to measure the neutron multiplication. A minimum of four neutron-sensitive chambers will be tied into the reactor safety system in such a way that tripping of any channel will cause a scram. During this period with the head removed the reactor safety system in-core monitors will not be in place.

The initial loading and critical testing is to be done with the safety system selector switch in the "refuel" position (see item B.9). This indicates that the following scram sensors will be in operation:

a. High sphere pressure
b. Low water level in reactor (This sensor may be bypassed in the "refuel" position until initial power operation.)
c. High reactor pressure (This sensor is in service in the "refuel" position because there is no reason to bypass it. With the reactor head off, of course, it cannot be effective.)

d. High level in scram dump tank
e. High neutron flux (For the initial loading operation the normal out-of-core chambers and dual-channel system are replaced by the special in-core chamber system described above in this item 2. As with the normal system, however, with the selector switch in the "refuel" position, the neutron flux trip setting must be reduced to a maximum setting of 10^{-3} rated power.)

f. Short period.

3. Reactor Water Level

Prior to the initial loading, the reactor vessel will be filled with water to a minimum of three feet above the active height of the core. As indicated in D. 2, the low water level sensor may be bypassed during the initial loading and test period when the water is required primarily as a moderator.

4. Shutdown Margin

a. "Stuck Rod" Criterion: At every stage during the initial loading, and in the fully loaded configuration, the control rods must provide a shutdown control margin of at least 0.01 k with any rod wholly out of the core and completely unavailable.

b. "Cocked Rod" Criterion: During core alterations after the first fuel cell is loaded, the reactor must be subcritical by at least 0.01 k with at least one control rod fully withdrawn in the region of the alteration and available for rapid scram insertion.

5. Minimum Critical Core

The following criteria will be used in loading to the minimum critical size core:

a. The selected core area will be within the control rod pattern.

b. Except for the final fuel increment, the size of each fuel addition will never exceed one-half the estimated amount to reach criticality. This estimate will be made using neutron multiplication measurements determined between fuel increment additions. The final fuel increment will not exceed a reactivity worth of $0.005\Delta k$.

c. The shut down margin criteria given in item D.4, will be applied.

The following tests will be conducted, as a minimum, with the minimum critical core:

d. Measurements of the strength of some of the control rods encompassed by or adjacent to the fuel loading;

e. Temperature coefficient ($\Delta k/^{o}F$ change in moderator temperature);

f. Void coefficient ($\Delta k/$ $^{o}/_{o}$ change in voids). The void coefficient averaged over the interior of a fuel channel will always be negative when the core is critical. Sufficient measurements will be made to conclude that the void coefficient, measured as indicated in the sentence above, is negative in all fuel channels.

6. Twice Minimum Critical Core

On successful completion of item D.5, the core will be loaded to approximately twice the minimum critical size, but not beyond the control rod outer periphery. The following measurements will be made, as a minimum, with this core:

a. Selected control rod calibrations;

b. Rod configurations for critical;

c. Temperature coefficient;

d. Void coefficient. This must meet the limitations indicated in item D.5.f.

7. Core Loaded to Periphery of Control Rod Pattern

On successful completion of item D.6, the core will be loaded to encompass all control rod cells (i.e., one row of fuel assemblies outside of control rod periphery). Criticality checks will be made periodically during this loading. Calculations and extrapolations of measurements made in accordance with items D.5 and D.6 will be used, prior to the actual fuel loading, to indicate whether the shutdown margin and void coefficient criteria can be met. Measurements will be made with this core, as a minimum, of the critical rod configurations with non-uniform distribution of withdrawn control rods.

8. Completed Core Loading

Calculations and test data from item D.7 will be used to determine the maximum number of fuel assemblies permitted in the area outside the control rod periphery in the fully loaded core. Following this determination, approximately half the number of fuel elements will be added uniformly around the core. The determination will then be repeated and followed by another "half loading." This sequence will be repeated until the last fuel assembly is inserted. The completed core will meet, as a minimum, the shutdown margin requirements indicated in item D.4. The test program for the completed core will include, as a minimum, the following items:

a. Critical rod configurations with non-uniform distribution of withdrawn control rods;

b. Temperature coefficient;

c. Void coefficient. This must meet the limitations indicated in item D.5.f.

E. POWER OPERATION

1. Power Test Program

Following successful completion of the initial loading and critical testing program, the power test program will be carried out. This program will include the following:

a. A series of tests at low power, and low to rated pressure, with studies of stability, reactivity, and power distribution as they are affected by variations in reactor pressure, temperature, voids, control rod positions, and transient poisons;

b. A series of tests at various power levels up to rated power, at low to rated pressure, with studies similar to those indicated in item E.1.a;

c. Studies of power transient effects, simulated equipment failures, and process mishaps;

d. Radiation surveys.

This testing program will be conducted at stepwise increasing power levels. At each power level, experiments will be conducted to predict the power level or other operating condition of the reactor at which maximum heat flux, reactor instability, or any other reactor design or operational limitation would be reached. The reactor power level or other limiting condition will not be increased in the next step more than halfway from the test conditions to the predicted limitation, provided that for reactor power levels in excess of 50% of rated power no step increase in power shall exceed 60 MW.

2. <u>Safety System Scram Settings</u>

Operative scram sensors and their settings with the safety system selector switch in the "Start" and "Run" positions are given in item B.9. The control rods cannot be withdrawn in the "Start" position until at least 15 inches of Hg condenser vacuum have been obtained, and the switch to the "Run" position cannot be made without scramming until at least 23 inches of Hg condenser vacuum have been obtained. The reactor will also scram if a reactor pressure of 200 psig is exceeded prior to obtaining at least 23 inches of Hg condenser vacuum.

3. <u>Determination of Maximum Reactor Power</u>

The "rated" or operational thermal power of the reactor is not, at least during initial power operation, a fixed value. Lower than equilibrium fuel cycle power production may be necessary initially, due to the possibilities of:

a. Less than full load of 488 fuel assemblies in order to meet shutdown margin criteria; and

b. Higher peaking factors initially than at equilibrium fuel cycle.

The maximum reactor power is, consequently, defined as that thermal power at which the maximum heat flux for any fuel rod is reached. This maximum heat flux will never exceed 350,000 Btu/ft^2-hr, based on calculations and experimental data. The rated heat flux and resulting rated reactor power are then set to 80% of their maximum values and, as indicated in items B.9, the high neutron flux scram setting will be no higher than an indicated 120% of the rated reactor power. However, in no case will the high neutron flux setting be allowed to exceed an indicated reactor thermal power of 782 MW (125% of the planned operational power of the fully loaded core).

Within the limitations on reactor power and heat flux set forth in the paragraph above, the reactor will be operated in such fashion as to:

a. Maintain at all times a burnout heat flux margin of at least 2 at the point closest to burnout in the hottest channel in the core based on a uniform steam quality over the cross-section of the channel; and
b. Be always well within the bounds of stability, as evidenced by the operation itself and any experimental data produced during the "Power Test Program" phase of operation.

4. Pressure Limits

a.	Maximum normal reactor operating pressure	1000 psig
b.	Maximum pressure setting for automatic reactor shutdown	1050 psig
c.	Maximum pressure setting for opening of electromatic relief valves	1085 psig
d.	Maximum pressure setting for opening of first main safety valve	1205 psig
e.	Maximum safety valve pressure setting	1250 psig
f.	Combined capacity of safety valves	At least 150% rated primary steam flow

5. Reactivity Limits

a. The average reactivity addition rate from withdrawal of the control rod with the maximum reactivity worth in the most adverse withdrawal pattern will not exceed $0.0029 \Delta k$/sec.

b. With the reactor in any condition, the shutdown margin criterion in item D.4.a must be met.

c. With the reactor in the hot operating condition, operation will not be continued when the reactivity worth of the control rods known to be stuck out of the core, or otherwise unavailable for control, exceeds half the value at which hot shutdown could not be accomplished. (It is expected that this would require shutdown for repair of the inoperable rods if three adjacent, or the equivalent in reactivity worth of nonadjacent, rods are known to be unavailable.) If, in such case, it is calculated that following shutdown and cooling of the reactor the shutdown margin will not be at least $0.01 \Delta k$, the liquid poison will be introduced prior to reaching the "cold" condition where this criterion could not be satisfied.

d. The void coefficient of reactivity will meet the limitations indicated in item D.5.f.

e. The moderator temperature coefficient of reactivity will change from positive to negative at a temperature below the maximum reactor temperature that can be attained under atmospheric conditions. Measurements will be made to confirm this.

6. Waste Disposal

The disposal of wastes resulting from power operations is discussed in item B.8. Disposal of all waste off site will be in a manner such that it is unlikely any person will receive radiation exposures in excess of the approximate permissible limits.

F. REFUELING AND MAINTENANCE

1. Operating Principles

All refueling and maintenance operations will be carried out in accordance with all the applicable operating principles given in Section C. Items in Section C which are particularly pertinent to these operations are those lettered c, e, f, g, h, i, j, k, and m.

2. Safety System Scram Sensors

Operative scram sensors and their setting with the selector switch in the "Refuel" position are given in item B.9.

Since some maintenance can be carried out under any of the possible reactor conditions, the safety sensors in operation will depend upon the particular job to be done. Even with the reactor in the shutdown condition, however, any maintenance work involving the removal of control rods from the core will be done with the safety system selector switch in the "Refuel" position.

3. Shutdown Margin

At every stage of refueling or maintenance, the minimum shutdown margin will satisfy the "stuck rod" criteria discussed in items E.5.b and E.5.c.

During movement of fuel in the core, or control rod maintenance, the minimum shutdown margin will satisfy the "cocked rod" criterion given in item D.4.b.

4. Liquid Poison System

The liquid poison system will be operative during refueling and maintenance operations in the reactor as well as during normal power operations.

TABLE I

SAFETY SYSTEM

External Sensors			Scram Trip Setting	Scram Trip Point (a)	Other Automatic Functions Performed	Remarks
Type	Number in each Channel	Coincidence in each Channel				
High Sphere Pressure	2	1 out of 2	Maximum Pressure of 2.0 psig	Setting $\leq$ 0.2 psig	Closes isolation valves and ventilation ducts	No bypasses
Low Water Level in Reactor Vessel	2	1 out of 2	At level which is a minimum of 43" above the top of the active fuel	Setting $\leq$ 1"	Closes isolation valves and ventilation ducts.	Bypassed in "Shutdown" position prior to initial power operation only. Bypass is to be removed.
Closure of Turbine Stop & Bypass Valves	2	1 out of 2	When oil pressure controlling these valves drops to a minimum of 50 psig	Setting $\leq$ 5 psig	Closes ventilation ducts	Bypassed in "Shutdown" or "Refuel" positions.
Closure of Primary Steam Sphere Isolation Valves	2(b)	2 out of 2 (One from each valve)	Closure of both valves beyond a maximum of 25% of stroke	Setting $\leq$ 5% stroke	Closes ventilation ducts and starts emergency cooling	Bypassed in "Shutdown" or "Refuel" positions.
Low water Level in Primary Steam Drum	2	1 out of 2	At level which is a maximum of 5" below the drum center line	Setting $\leq$ 1"	Closes ventilation ducts	Bypassed in "Shutdown" or "Refuel" positions.
Low Condenser Vacuum	2(c)	1	At minimum condenser vacuum of 22" Hg	Setting $\leq$ 0.25"	Closes ventilation ducts	Bypassed in "Shutdown" or "Refuel" positions. Bypassed in "Start" position if reactor pressure below 200 psig & condenser vacuum greater than 15"Hg
		1	At minimum condenser vacuum of 23" Hg	Setting $\leq$ 0.25"		
High Reactor Pressure	2	1 out of 2	At maximum reactor pressure of 1050 psig	Setting $\leq$ 10 psig	Closes ventilation ducts & starts emergency cooling.	Bypassed in "Shutdown" position.

TABLE I

SAFETY SYSTEM (Continued)

External Sensors			Scram Trip Setting	Scram Trip Point[a]	Other Automatic Functions Performed	Remarks
Type	Number in each Channel	Coincidence in each Channel				
High Level in Scram Dump Tank	2	1 out of 2	At tank level which is $4'4\frac{1}{2}''$ above the base line of the lower tangent of the tank	Setting $\leq$ 1"	Closes ventilation ducts	Bypassed in "Shutdown position. Manual bypass also prevents control rod withdrawal. At scram point, there is sufficient free volume remaining in the scram dump tank to accommodate the water from 2.7 scrams.
High Neutron Flux	3	1 out of 3 or 1 out of 2 if 1 is bypassed	When leakage flux indicates a maximum of 120% rated power	Setting $\leq$ 3% of rated power	Closes ventilation ducts	Bypassed in "Shutdown" position. Interlocked in "Refuel" position so as to require reduction in trip setting to a maximum setting of 10-3 rated power. At any time other than in the "shutdown" position only one of the six flux trips provided can be bypassed.
Short Period	3[d]	2 out of 3	At minimum period of 4 seconds	Setting $\leq$ 0.5 seconds	Closes ventilation ducts	Bypassed in "Shutdown" and "Run" positions.

(a) The point at which a scram is actually initiated may be different from the "scram trip setting" by the amount of the tolerance for instrument inaccuracies. The amount of this tolerance is indicated by the values given in this column.

(b) Each valve-position switch (of which there is one per valve) has two contacts in each safety channel.

(c) There are two vacuum sensors. Each sensor has one contact in each of the two safety channel.

(d) There are three period sensoring chambers. Each chamber has two contacts in each of the two safety channels.

APPENDIX "B"

TO

COMMONWEALTH EDISON COMPANY

PROPOSED LICENSE

Estimated Schedule of Transfers of Special Nuclear Material from the Commission to Commonwealth Edison and to the Commission from Commonwealth Edison

(1) Date of Transfer (Fiscal Year)	(2) Transfers from AEC to Commonwealth Edison Kilograms U-235	(3) Returns by Commonwealth Edison to AEC Kilograms U-235	(4) Net Yearly Distribution Kilograms U-235	(5) Cumulative Distribution Kilograms U-235
1958	243.3	-	243.3	243.3
1959	974.3	198.0	776.3	1,019.6
1960	376.7	84.0	292.7	1,312.3
1961	376.7	185.0	191.7	1,504.0
1962	376.7	185.0	191.7	1,695.7
1963	376.7	185.0	191.7	1,887.4
1964	376.7	185.0	191.7	2,079.1
1965	426.5	185.0	241.5	2,320.6
1966	386.7	185.0	201.7	2,522.3
1967	346.8	-*	346.8	2,869.1
1968	312.9	-	312.9	3,182.0
1969	272.1	-	272.1	3,454.1
1970	228.2	-	228.2	3,682.3
1971	228.2	-	228.2	3,910.5
1972	228.2	-	228.2	4,138.7
1973	228.2	-	228.2	4,366.9
1974	228.2	-	228.2	4,595.1
1975	228.2	-	228.2	4,823.3
1976	228.2	-	228.2	5,051.5
1977	228.2	-	228.2	5,279.7
1978	228.2	-	228.2	5,507.9
1979	228.2	-	228.2	5,736.1
1980	228.2	-	228.2	5,964.3
1981	228.2	-	228.2	6,192.5
1982	228.2	-	228.2	6,420.7
1983	228.2	-	228.2	6,648.9
1984	228.2	-	228.2	6,877.1
1985	228.2	-	228.2	7,105.3
1986	228.2	-	228.2	7,333.5
1987	228.2	-	228.2	7,561.7
1988	228.2	-	228.2	7,789.9
1989	228.2	-	228.2	8,018.1
1990	228.2	-	228.2	8,246.3
1991	228.2	-	228.2	8,474.5
1992	228.2	-	228.2	8,702.7
1993	228.2	-	228.2	8,930.9
1994	228.2	-	228.2	9,159.1
1995	228.2	-	228.2	9,387.3
	10,779.3	1,392.0*	9,387.3	

*The fuel being discharged during fiscal years 1967 through 1995 will have been depleted to such an extent in U-235 that it no longer is special nuclear material. Therefore, no credit for returns has been included for these years.