FACILITY OPERATING LICENSE NPF-30

(4) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source of special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

UE is authorized to operate the facility at reactor core power levels not in excess of ~~3411~~ 3565 megawatts thermal (100% power) in accordance with the conditions specified herein and in Attachment 1 to this license. The preoperational tests, startup tests and other items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A** and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. UE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan;

(3) Environmental Qualification (Section 3.11, SSER #3)*

(a) Prior to ~~March 31~~ November 30,*** 1985, UE shall environmentally qualify all electrical equipment accordingly to the provisions of 10 CFR 50.49.

(b) Prior to restart following the first refueling outage, UE shall have qualified the reactor vessel level instrumentation system high volume sensor.

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*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

** Plus approved amendments.

*** See Amendment No. 5.

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DEFINITIONS

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DEFINITIONS

SECTION

DEFINITIONS (Continued)

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3, .

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow from the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

~~DESIGN THERMAL POWER~~

~~1.10 DESIGN THERMAL POWER shall be a design total reactor core heat transfer rate to the reactor coolant of 3565 MWt.~~

DOSE EQUIVALENT I-131

1.~~11~~10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.~~12~~ 11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.~~13~~ 12 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.~~14~~ 13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.~~15~~ 14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.~~16~~ 15 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.~~17~~ 16 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL

1.~~18~~ 17 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

OPERABLE - OPERABILITY

1.~~19~~ 18 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.~~20~~ 19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.~~21~~ 20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, or (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.~~22~~ 21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM

1.~~23~~ 22 The PROCESS CONTROL PROGRAM shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71 and Federal and State regulations, burial ground requirements, and other requirements governing the disposal of the radioactive waste.

PURGE - PURGING

1.~~24~~ 23 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.~~25~~ 24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.~~26~~ 25 RATED THERMAL POWER shall be a total core heat transfer rate to the reactor coolant of ~~3411~~ 3565 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.~~27~~ 26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.~~28~~ 27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.~~29~~ 28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.~~30~~ 29 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SLAVE RELAY TEST

1.~~31~~ 30 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOLIDIFICATION

1.~~32~~ 31 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.~~33~~ 32 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.~~34~~ 33 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.~~35~~ 34 THERMAL POWER shall be the total core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.~~36~~ 35 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.~~37~~ 36 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.~~38~~ 37 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.~~39~~ 38 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features (ESF) Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.~~40~~ 39 VENTING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

WASTE GAS HOLDUP SYSTEM

1.~~41~~ 40 A WASTE GAS HOLDUP SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System off-gases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

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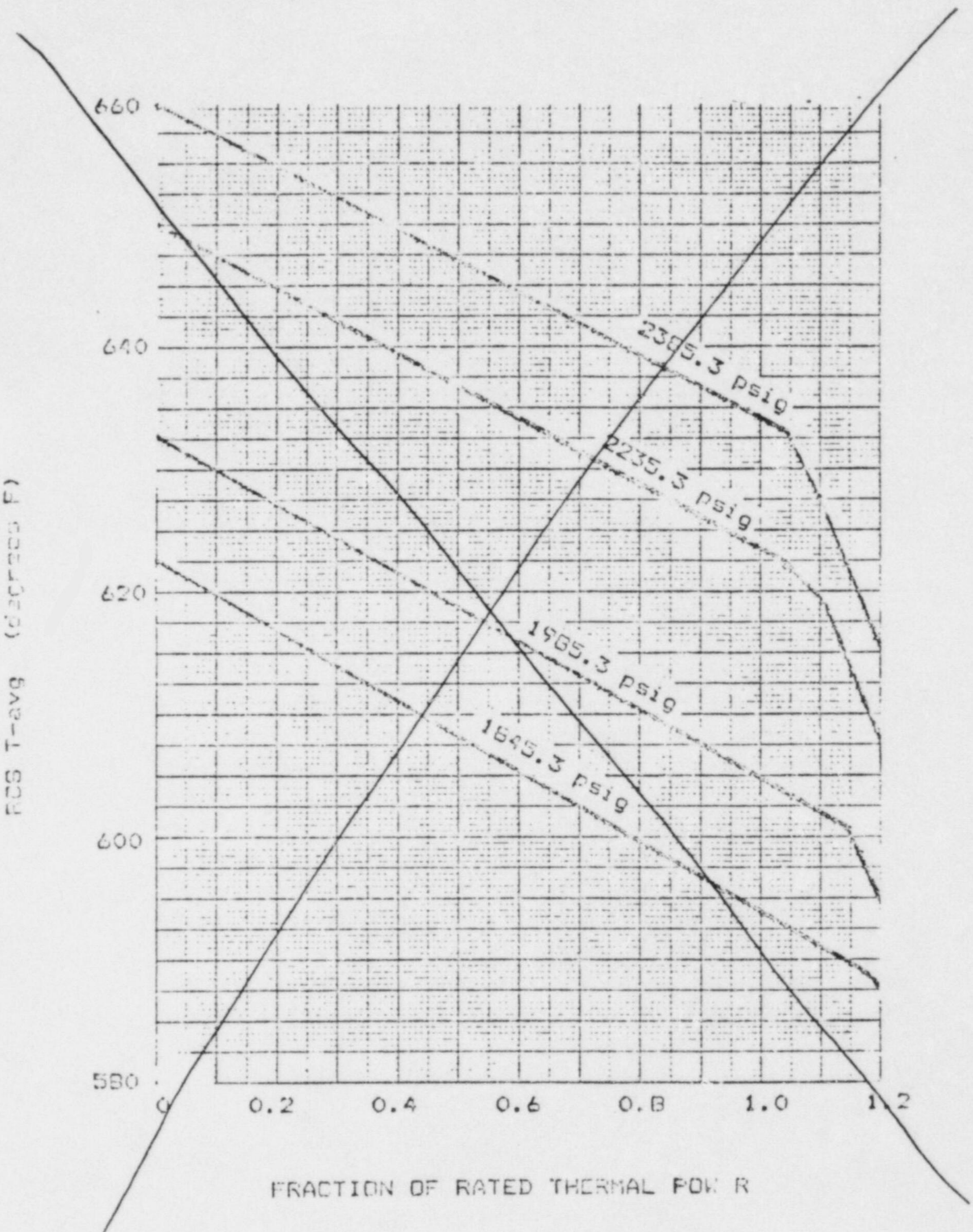

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

APPLICABLE FOR LICENSED CORE THERMAL POWER = 3565 MWt

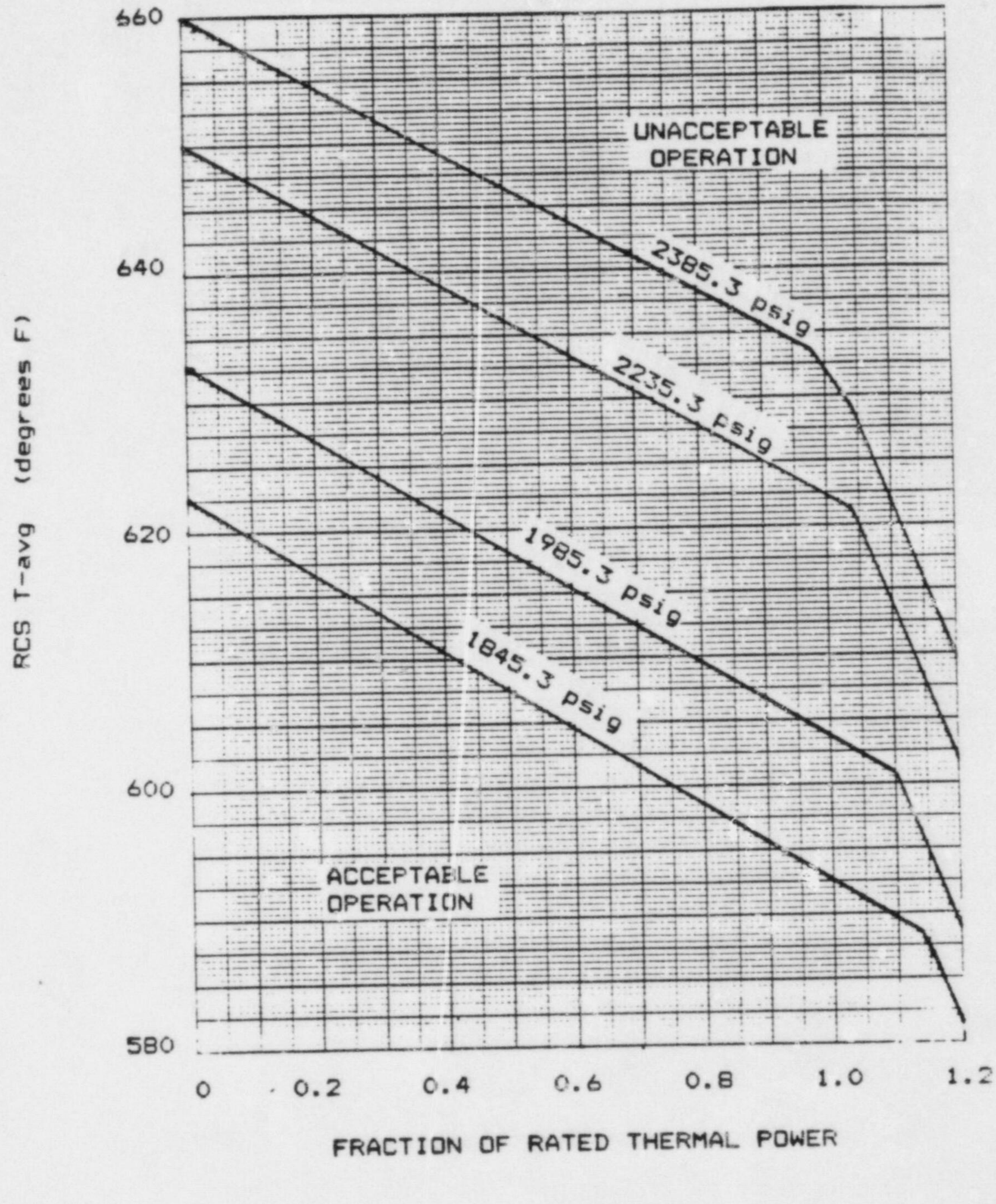

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3 s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, τ_3 = 0 s;
ΔT_0	=	~~61.8~~ 63.2°F (Referenced ΔT at ~~DESIGN~~ RATED THERMAL POWER);
K_1	=	1.15;
K_2	=	0.0251/°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 28 s, τ_5 = 4 s;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	≤	588.4°F (Referenced T_{avg} at ~~DESIGN~~ RATED THERMAL POWER);
K_3	=	0.00116;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) For $q_t - q_b$ between -35% and + 6%, $f_1(\Delta I) = 0$, where q_t and q_b are percent ~~DESIGN~~ RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of ~~DESIGN~~ RATED THERMAL POWER;

(ii) For each percent that the magnitude of $q_t - q_b$ exceeds -35%, the ΔT Trip Setpoint shall be automatically reduced by 1.91% of its value at ~~DESIGN~~ RATED THERMAL POWER; and

(iii) For each percent that the magnitude of $q_t - q_b$ exceeds +6%, the ΔT Trip Setpoint shall be automatically reduced by 1.89% of its value at ~~DESIGN~~ RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~2.0%~~ 2.3 of ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:

ΔT = Measured ΔT by RTD Manifold Instrumentation;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, τ_1, = 8 s., τ_2 = 3 s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constant utilized in the lag compensator for ΔT, τ_3 = 0 s;

ΔT_o = ~~61.[illegible]~~ 632 °F (Referenced ΔT at ~~DESIGN~~ RATED THERMAL POWER);

K_4 = 1.080;

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature;

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation;

τ_7 = Time constant utilized in the rate-lag compensator for T_{avg}, τ_7 = 10 s;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.0065/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$;

T = Average Temperature, °F;

T'' = Indicated T_{avg} at RATED ~~DESIGN~~ THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.4°F);

S = Laplace transform operator, s^{-1}; and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~4.1%~~ 3.3 of ΔT span.

CALLAWAY UPRATING SUBMITTAL
ATTACHMENT 3
ENVIRONMENTAL EVALUATION

Union Electric has performed an environmental evaluation to support uprating the Callaway Plant from the currently licensed power level to the core thermal power of 3565 MWt which corresponds to a NSSS power level of 3579 MWt. This change does not provide a significant adverse environmental impact, nor a significant increase in effluents, nor an adverse land or cultural resource altering activity. The original licensing evaluations for the plant, including the NRC Environmental Evaluations (Ref. NUREG-75/011, 3/75, Section 1.1), were based on a NSSS thermal power level of 3579 MWt. Therefore, the proposed uprating remains within the bounds of the original environmental analyses.

Attachment 4
ULNRC-1471
March 31, 1987

CALLAWAY UPRATING SUBMITTAL
ATTACHMENT 4
SIGNIFICANT HAZARDS EVALUATION

SIGNIFICANT HAZARDS EVALUATION FOR CALLAWAY PLANT UPRATING TO 3579 MWt

This evaluation supports Union Electric Company's license amendment request to increase the thermal output of the Callaway Plant. At present, Callaway is licensed to operate at a reactor core thermal power level of 3411 MWt, which corresponds to a Nuclear Steam Supply System (NSSS) thermal power level of 3425 MWt. This amendment requests the necessary License and Technical Specification changes to operate the Callaway Plant at a reactor core thermal power level of 3565 MWt, which corresponds to an NSSS thermal power level of 3579 MWt. This represents an increase of 4.5% and is equivalent to the Engineered Safety Features Design Rating of the Callaway Plant. Union Electric has submitted a license amendment request (ULNRC-1470, dated March 31, 1987) to utilize Westinghouse 17x17 VANTAGE 5 fuel beginning with Callaway Cycle 3 operation. The requested uprating is scheduled to begin concurrently with Cycle 3 operation.

In order to implement the uprating, Callaway Plant, Facility Operating License NPF-30, Section 2.C.(1) requires revision to indicate operation is authorized at reactor core power levels not in excess of 3565 megawatts thermal. Technical Specification changes include revision of the definitions of Design Thermal Power (Section 1.10) and Rated Thermal Power (Section 1.26), rescaling of Figure 2.1-1 (Core Safety Limits), and revision of the notes associated with OT Delta T and OP Delta T (Table 2.2-1).

In support of the requested uprating, the NSSS system and component designs were analyzed to verify continued compliance with licensing criteria and standards currently required by the Callaway operating license and the functional requirements specified in the FSAR for operation at 3579 MWt. The review was conducted in accordance with the methodology documented in Westinghouse topical report WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant". This methodology has been referenced in connection with the approval of similar license amendments for other facilities. WCAP-10263 methodologies were also utilized in the review of the interfaces between the NSSS and the Balance of Plant (BOP) systems and components. Reviews of the BOP systems and components were conducted and verify their capability to meet the requirements for operation at 3579 MWt. Review of the turbine generator is based on the Valves Wide Open heat balance which represents an NSSS power of 3562 MWt. The ability of the turbine generator to operate up to 3579 MWt will be determined by careful monitoring of plant performance parameters. These reviews demonstrate that the Callaway Plant is capable, in the present design configuration, of operating at the uprated conditions without violating any of the design or safety limits specified in the Callaway FSAR and Facility Operating License NPF-30.

In conjunction with the use of Optimized Fuel Assemblies (OFA) during Callaway Cycle 2 operation, accident analyses were performed at the uprated power conditions. These were documented in ULNRC-1207, dated 11/15/85 and ULNRC-1247, dated 1/28/86. In conjunction with the change to VANTAGE 5 fuel for Cycle 3, certain of the accident analyses were redone. These analyses were also based on the uprated power condition and are documented in ULNRC-1470, dated March 31, 1987. The analyses that were not redone are those that are still bounded by the OFA analyses performed for Cycle 2. The results of the accident analyses demonstrate that all safety limits are met at the uprated power conditions.

Radiological dose consequences pertaining to the uprated power conditions have been completed and documented as part of the license amendment in support of Callaway Cycle 3 for the utilization of VANTAGE 5 fuel, which are documented in ULNRC-1470 dated March 31, 1987. The evaluation is also applicable to this amendment request. The environmental impact of operating Callaway Plant at the uprated condition was previously evaluated in the Callaway Environmental Report as a bounding assumption. This has been approved by NRC in NUREG-75/011, Section 1.1 dated 3/75.

a) This change does not involve a significant increase in the probability or consequences of an accident previously evaluated. This is based on the environmental evaluation being previously approved at the uprated conditions and the radiological evaluation which is presented in the Callaway Cycle 3 amendment request for the utilization of VANTAGE 5 fuel.

b) This change does not create the possibility of a new or different kind of accident from any accident previously evaluated. This is based on system and component reviews which verified their capability to operate at the uprated conditions.

c) This change does not involve a significant reduction in a margin of safety. Accident analyses were performed at the uprated conditions and demonstrate the DNB design basis remains unchanged, the RCS pressure limit of 2700 psig is not exceeded, and the LOCA results remain well below the regulatory limits given in 10CFR50.46.

Based on the above discussions, the amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated; does not create the possibility of a new or different kind of accident from any accident previously evaluated; does not involve a reduction in the required margin of safety. Based on the foregoing, the requested amendment does not present a significant hazard.

CALLAWAY UPRATING SUBMITTAL

ATTACHMENT 5: APPENDICES

APPENDIX A

NSSS UPRATING - LICENSING REPORT