



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. STN 50-483

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 35
License No. NPF-30

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment filed by Union Electric Company (UE, the licensee) dated March 31, 1987, as supplemented by letters dated April 21, September 18, October 2, October 23 and November 13, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, Facility Operating License No. NPF-30 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraphs 2.C.(1) and 2.C.(2) to read as follows:

(1) Maximum Power Level

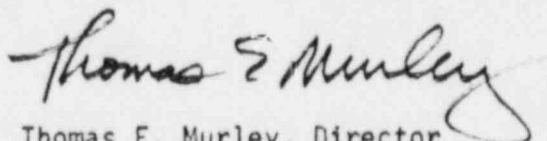
UE is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100% power) in accordance with the conditions specified herein and in Attachment 1 to this license. The preoperational tests, startup tests and other items identified in Attachment 1 to this 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, as revised through Amendment No. 35, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. UE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 30, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 35

OPERATING LICENSE NO. NPF-30

DOCKET NO. 50-483

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Corresponding overleaf pages are provided to maintain document completeness.

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1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Technical Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

DEFINITIONS

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.

DOSE EQUIVALENT I-131

1.10 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."

DEFINITIONS

\bar{E} - AVERAGE DISINTEGRATION ENERGY

1.11 \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

1.12 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

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.14 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

1.15 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

1.16 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.

DEFINITIONS

OFFSITE DOSE CALCULATION MANUAL

1.17 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

1.18 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

1.19 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

1.20 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 50.55, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.21 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

1.22 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

1.23 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.

DEFINITIONS

QUADRANT POWER TILT RATIO

1.24 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

1.25 RATED THERMAL POWER shall be a total core heat transfer rate to the reactor coolant of 3565 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 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

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

RESTRICTED AFD OPERATION

1.28 RESTRICTED AFD OPERATION (RAFDO) limits the AXIAL FLUX DIFFERENCE (AFD) to a +3% target band about the target flux difference and restricts power levels to between APLND and either APLRAFDO or 100% RATED THERMAL POWER, whichever is less. APLND and APLRAFDO are defined in Specifications 3.2.1 and 4.2.2.3, respectively. RAFDO may be entered at the discretion of the licensee.

SHUTDOWN MARGIN

1.29 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 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 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.

DEFINITIONS

SOLIDIFICATION

1.32 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.33 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 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 THERMAL POWER shall be the total core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.36 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 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.38 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.

DEFINITIONS

VENTILATION EXHAUST TREATMENT SYSTEM

1.39 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 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 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.

TABLE 1.1
FREQUENCY NOTATION

<u>NOTATION</u>	<u>FREQUENCY</u>
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for four loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

APPLICABLE FOR LICENSED CORE THERMAL POWER = 3565 MWE

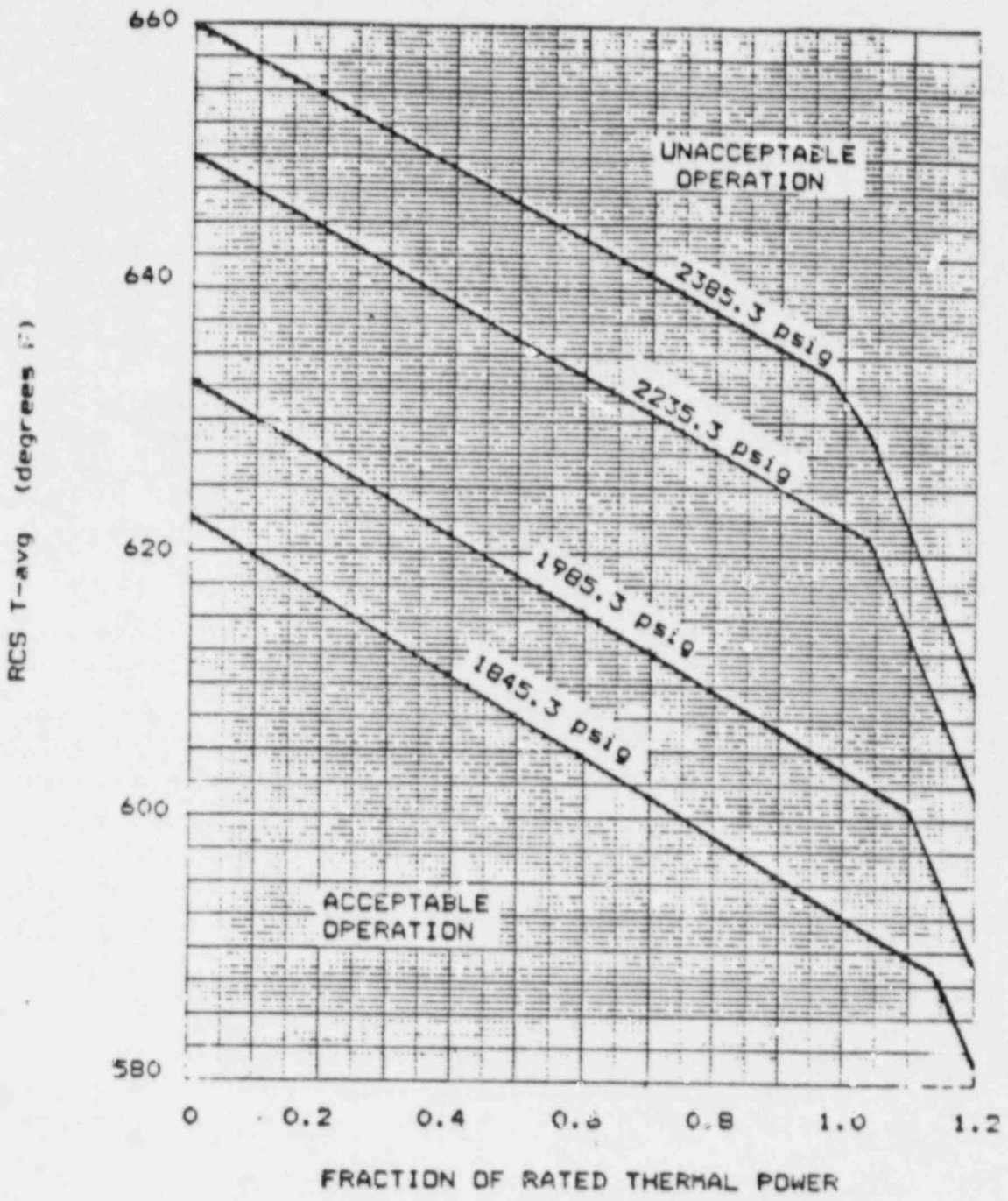


FIGURE 2.1-1
REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

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 , $\tau_1 = 8$ s,
 $\tau_2 = 3$ s; $\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT ; τ_3 = Time constant utilized in the lag compensator for ΔT , $\tau_3 = 0$ s; ΔT_0 = Indicated ΔT at RATED THERMAL POWER; K_1 = 1.15; K_2 = 0.0251/ $^{\circ}$ 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} , $\tau_4 = 28$ s,
 $\tau_5 = 4$ s; T = Average temperature, $^{\circ}$ 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, $\tau_6 = 0$ s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	\leq	588.4°F (Referenced T_{avg} at 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 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 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 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 RATED THERMAL POWER.

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

TABLE 2.2-1 (Continued)
 TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \frac{1}{(1 + \tau_3 S)} \leq \Delta T_0 \{K_4 - K_5 \frac{(\tau_7 S)}{1 + \tau_7 S} \frac{1}{1 + \tau_6 S} T - K_6 [T \frac{1}{1 + \tau_6 S} - T^*] - f_2(\Delta I)\}$$

- 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 ,
 $\tau_1 = 8$ s., $\tau_2 = 3$ s.;
 - $\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT ;
 - τ_3 = Time constant utilized in the lag compensator for ΔT , $\tau_3 = 0$ s.;
 - ΔT_0 = Indicated ΔT at 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} , $\tau_7 = 10$ 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, $\tau_6 = 0$ s.;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

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

T = Average Temperature, °F;

T'' = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq 583.4^\circ\text{F}$);

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 3.3% of ΔT span.