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November 1, 1990

Docket No. 50-483

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Mr. Donald F. Schnell
Senior Vice President - Nuclear
Union Electric Company
Post Office Box 149
St. Louis, Missouri 63166

Dear Mr. Schnell:

SUBJECT: AMENDMENT NO. 58 TO FACILITY OPERATING LICENSE NO. NPF-30 (TAC NO. 76432)

The Commission has issued the enclosed Amendment No. 58 to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1. This amendment revises the Technical Specifications in response to your application dated March 6, as supplemented by letters dated June 5, July 5 and July 18, 1990.

The amendment revises Technical Specifications (TSs) and associated Bases, which contain cycle-specific core operating parameters in accordance with NRC Generic Letter 88-16, "REMOVAL OF CYCLE-SPECIFIC PARAMETER LIMITS FROM TECHNICAL SPECIFICATIONS," dated October 4, 1988. The revision relocates the cycle-specific core parameters from the TSs and places them in the Core Operating Limits Report (COLR) under the control of your unit review group (On-site Review Committee (ORC) for the Callaway Plant). The staff acknowledges your commitment to notify the NRC, in writing, of changes to the COLR concurrent with plant implementation.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by
Michael J. Davis for

Anthony T. Gody, Jr., Project Manager
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 58 to License No. NPF-30
2. Safety Evaluation

cc w/enclosures:
See next page

DOCUMENT NAME: 76342 AMD

Office:	LA/PDIII-3	PM/PDIII-3	PD/PDIII-3	SRXB	OGC
Surname:	PKreutzer	TGody/tg	JHannon	RJones	
Date:	10/10/90	10/12/90	10/16/90	10/15/90	10/24/90

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Project Directorate III-3
Division of Reactor Projects - III,
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Surname:	PKreutzer	TGody/tg	JHannon	RJones	
Date:	10/10/90	10/12/90	10/16/90	10/15/90	10/23/90

Mr. D. F. Schnell
Union Electric Company

Callaway Plant
Unit No. 1

cc:

Dr. J. O. Cermack
CFA Inc.
4 Professional Dr., Suite 110
Gaithersburg, MD 20879

Gerald Charnoff, Esq.
Thomas A. Baxter, Esq.
Shaw, Pittman, Potts & Trowbridge
2300 N Street, N. W.
Washington, D. C. 20037

Mr. T. P. Sharkey
Supervising Engineer,
Site Licensing
Union Electric Company
Post Office Box 620
Fulton, Missouri 65251

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
RR#1
Steedman, Missouri 65077

Mr. Alan C. Passwater, Manager
Licensing and Fuels
Union Electric Company
Post Office Box 149
St. Louis, Missouri 63166

Manager - Electric Department
Missouri Public Service Commission
301 W. High
Post Office Box 360
Jefferson City, Missouri 65102

Regional Administrator
U. S. NRC, Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. Ronald A. Kucera, Deputy Director
Department of Natural Resources
P. O. Box 176
Jefferson City, Missouri 65102

Mr. Bart D. Withers
President and Chief
Executive Officer
Wolf Creek Nuclear Operating
Corporation
P. O. Box 411
Burlington, Kansas 66839

Mr. Dan I. Bolef, President
Kay Drey, Representative
Board of Directors Coalition
for the Environment
St. Louis Region
6267 Delmar Boulevard
University City, Missouri 63130

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. 58
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 6, 1990 as supplemented by letters dated June 5, July 5, and July 18, 1990, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-30 is hereby amended to read as follows:

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P PDC

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 58, 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 immediately to be implemented no later than 30 days from the date of its issuance. The licensee will immediately inform the Commission, in writing, of the implementation date.

FOR THE NUCLEAR REGULATORY COMMISSION

John N. Hannon

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: November 1, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 58

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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V	V
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XIV	XIV
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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.

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.

CORE OPERATING LIMITS REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit specific document that provides core operating limits for the current operating reload cycle. The cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

1.11 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 $\bar{E}$ 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 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 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.15 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 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 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 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. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4, and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-annual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.7.

OPERABLE - OPERABILITY

1.19 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 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 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 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 The PROCESS CONTROL PROGRAM shall contain the current formulas, 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 Parts 20, 61, and 71, and State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

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

REACTOR TRIP SYSTEM RESPONSE TIME

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

RESTRICTED AFD OPERATION

1.29 RESTRICTED AFD OPERATION (RAFDO) limits the AXIAL FLUX DIFFERENCE (AFD) to a $\pm 3\%$ target band about the target flux difference and restricts power levels to between APL^{ND} and either APL^{RAFDO} or 100% RATED THERMAL POWER, whichever is less. APL^{ND} and APL^{RAFDO} 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

1.30 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

1.31 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

1.32 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.

DELETED

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.

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1% Δk/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1% Δk/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1% Δk/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

   1) Reactor Coolant System boron concentration,

   2) Control rod position,

   3) Reactor Coolant System average temperature,

   4) Fuel burnup based on gross thermal energy generation,

   5) Xenon concentration, and

   6) Samarium concentration.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the Core Operating Limits Report (COLR). The maximum upper limit shall be less positive than +5 pcm/°F for power levels up to 70% RATED THERMAL POWER and a linear ramp from that point to 0 pcm/°F at 100% RATED THERMAL POWER for the all rods withdrawn, beginning of cycle life (BOL) condition.

APPLICABILITY: Beginning of Cycle Life (BOL) Limit - MODES 1 and 2*#
End of Cycle Life (EOL) Limit - MODES 1, 2, and 3#

ACTION:

a. With the MTC more positive than the BOL limit specified in the COLR, operation in MODES 1 and 2 may proceed provided:

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to within the BOL limits specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

   2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

   3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

---

* With K_{eff} greater than or equal to 1.

See Special Test Exception Specification 3.10.3.

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit specified in the COLR, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the COLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified in the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 551°F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 551°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 551°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 561°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

#With K_{eff} greater than or equal to 1.

*See Special Test Exception Specification 3.10.3.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

   a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;

   b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

   c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

   d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

ACTION d - Restore the inoperable rods to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE
FULL-LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal at Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary Coolant System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length shutdown and control rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 551°F, and

b. All Reactor Coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the rod drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be limited in physical insertion as specified in the Core Operating Limits Report (COLR).

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Restore the rod to within the insertion limit specified in the COLR, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be within the insertion limit specified in the COLR:

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the Core Operating Limits Report (COLR).

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

---

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.

FIGURE 3.1-1

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band (flux difference units) about the target flux difference as specified in the Core Operating Limits Report (COLR).

The indicated AFD may deviate outside the applicable required target band at greater than or equal to 50% but less than 0.9 APL^{ND}** or 90% of RATED THERMAL POWER, whichever is less, provided the indicated AFD is within the Acceptable Operation Limits specified in the COLR and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the applicable required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER*,#

ACTION:

a. With the indicated AFD outside of the applicable required target band and with THERMAL POWER greater than or equal to 0.9 APL^{ND}** or 90% of RATED THERMAL POWER, whichever is less, within 15 minutes, either:

   1. Restore the indicated AFD to within the applicable required target band limits, or

---

*See Special Test Exception Specification 3.10.2.

#Surveillance testing of the Power Range Neutron Flux channel may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits specified in the COLR and THERMAL POWER $\leq APL^{NO}$***. A total of 16 hours operation may be accumulated with the AFD outside of the applicable required target band during testing without penalty deviation.

**APL^{ND} is the minimum allowable power level for RESTRICTED AFD OPERATION and is specified in the Core Operating Limits Report per Specification 6.9.1.9.

***APL^{NO} is equal to the

$$\underset{\text{over } Z}{\text{minimum}} \left(\frac{F_Q^{RTP} * K(Z)}{F_Q^M(Z) * W(Z)_{NO}} * 100 \right)$$

and $F_Q^M(Z)$ and $W(Z)_{NO}$ are defined in 4.2.2.2.c.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. Reduce THERMAL POWER to less than 0.9 APL^{ND**} or 90% of RATED THERMAL POWER, whichever is less, and discontinue RESTRICTED AFD OPERATION (if applicable).

b. With the indicated AFD outside of the applicable required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits specified in the COLR and with THERMAL POWER less than 0.9 APL^{ND**} or 90%, whichever is less, but equal to or greater than 50% of RATED THERMAL POWER, reduce:

1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

2. The Power Range Neutron Flux-High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

c. With the indicated AFD outside of the applicable required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the applicable required target band.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

FIGURE 3.2-1

DELETED

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} * K(Z) \text{ for } P > 0.5, \text{ and}$$

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} \quad K(Z) \text{ for } p \leq 0.5.$$

Where F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the Core Operating Limits Report (COLR),

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}, \text{ and}$$

K(Z) = the normalized $F_Q(Z)$ as a function of core height as specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

FIGURE 3.2-2

DELETED

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 For Normal Operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)_{NO}} \text{ for } P > 0.5$$

$$F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{W(Z)_{NO} \times 0.5} \text{ for } P \leq 0.5$$

where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, $K(Z)$ is the normalized $F_Q(Z)$ as a function of core height, P is the relative THERMAL POWER, and $W(Z)_{NO}$ is the cycle dependent, Normal Operation function that accounts for power distribution transients encountered during Normal Operation. F_Q^{RTP}, $K(Z)$ and $W(Z)_{NO}$ are specified in the Core Operating Limits Report as per Specification 6.9.1.9.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding, by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or

2. At least once per 31 Effective Full Power Days (EFPD), whichever occurs first.

*During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

4.2.2.2 (Continued)

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right)$$

has increased since the previous determination of $F_Q^M(z)$, either of the following actions shall be taken:

1. $F_Q^M(z)$ shall be increased by 2% over that specified in Specification 4.2.2.2c., or

2. $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right) \text{ is not increasing.}$$

f. With the relationships specified in Specification 4.2.2.2c. above not being satisfied:

1. Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

$$\left[\left(\text{max. over } z \text{ of} \left(\frac{F_Q^M(z) \times W(z)\ NO}{F_Q^{RTP} \times K(z)} \right) \right) - 1 \right] \times 100 \quad \text{for} \quad P \geq 0.5$$

$$\left[\left(\text{max. over } z \text{ of} \left(\frac{F_Q^M(z) \times W(z) NO}{F_Q^{RTP} \times K(z)} \right) \right) - 1 \right] \times 100 \quad \text{for} \quad P < 0.5$$

2. Either one of the following actions shall be taken:

(a) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or

(b) Verify that the requirements of Specification 4.2.2.3 for RESTRICTED AFD OPERATION are satisfied and enter RESTRICTED AFD OPERATION.

g. The limits specified in Specifications 4.2.2.2.c., 4.2.2.2.e., and 4.2.2.2.f. above are not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

4.2.2.3 RESTRICTED AFD OPERATION (RAFDO) is permitted at powers above APL^{ND} if the following conditions are satisfied:

a. Prior to entering RAFDO, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain RAFDO surveillance (AFD within the limits specified in the COLR) during this time period. RAFDO is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{RAFDO} or between APL^{ND} and 100% (whichever is more limiting) and F_Q surveillance is maintained pursuant to Specification 4.2.2.4. APL^{RAFDO} is defined as:

$$APL^{RAFDO} = \underset{\text{over } z}{\text{minimum}} \left(\frac{F_Q^{RTP} \times K(Z)}{F_Q^{M}(Z) \times W(Z)_{RAFDO}} \right) \times 100\%$$

where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. The F_Q limit is F_Q^{RTP}. $W(Z)_{RAFDO}$ is the cycle dependent function that accounts for limited power distribution transients encountered during RAFDO. F_Q^{RTP}, $K(Z)$, and $W(Z)_{RAFDO}$ are specified in the Core Operating Limits Report as per Specification 6.9.1.9.

b. During RAFDO, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3.a. shall be satisfied before re-entering RAFDO.

4.2.2.4 During RAFDO, $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND}.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)_{RAFDO}} \quad \text{for } P > APL^{ND}$$

where $F_Q^M(Z)$ is the measured $F_Q(Z)$. The F_Q limit is F_Q^{RTP}. P is the relative THERMAL POWER. $W(Z)_{RAFDO}$ is the cycle dependent function that accounts for limited power distribution transients encountered during RAFDO. F_Q^{RTP}, $K(Z)$, and $W(Z)_{RAFDO}$ are specified in the Core Operating Limits Report as per Specification 6.9.1.9.

4.2.2.4 (Continued)

d. Measuring $F_Q^M(z)$ in conjunction with target flux difference determination according to the following schedule:

1. Prior to entering RAFDO after satisfying Section 4.2.2.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to mapping, and

2. At least once per 31 Effective Full Power Days.

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z)}{K(z)} \right]$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1. $F_Q^M(z)$ shall be increased by 2 percent over that specified in 4.2.2.4.c, or

2. $F_Q^M(z)$ shall be measured at least once per 7 EFPD until two successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z)}{K(z)} \right] \text{ is not increasing.}$$

f. With the relationship specified in 4.2.2.4.c above not being satisfied, comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated with the following expression:

$$\left[\left(\text{max. over } z \text{ of} \left(\frac{F_Q^M(z) \times W(z)_{RAFDO}}{F_Q^{RTP} \times K(z)} \right) \right) - 1 \right] \times 100 \text{ for } P \geq APL^{ND}$$

$$P$$

g. The limits specified in 4.2.2.4.c, 4.2.2.4.e, and 4.2.2.4.f above are not applicable in the following core plane regions:

1. Lower core region from 0 to 15 percent, inclusive.

2. Upper core region from 85 to 100 percent, inclusive.

4.2.2.5 When $F_Q(z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 or 4.2.2.4, an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq F^{RTP}_{\Delta H} [1 + PF_{\Delta H} (1-P)]$$

where: $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER (RTP) specified in the Core Operating Limits Report (COLR). $PF_{\Delta H}$ = the Power Factor Multiplier for $F^N_{\Delta H}$ specified in the COLR.

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

$F^N_{\Delta H}$ Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used since an uncertainty of 4% for incore measurement of $F^N_{\Delta H}$ has been included in the above limit.

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Within 2 hours either:

   1. Restore the $F^N_{\Delta H}$ to within the above limits, or

   2. Reduce THERMAL POWER TO LESS THAN 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux-High Trip Setpoint to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours.

b. Demonstrate through in-core flux mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core flux mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting End of Life (EOL) MTC value. The 300 ppm surveillance limit MTC value represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting EOL MTC value.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boration Systems ensure that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include: (1) borated water sources, (2) centrifugal charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature equal to or greater than 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The Boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 17,658 gallons of 7000 ppm borated water from the boric acid storage tanks or 83,745 gallons of 2350 ppm borated water from the RWST. With the RCS average temperature less than 350°F, only one boron injection flow path is required.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core at or above the safety analysis DNBR limits during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definition of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods; and

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelopes of the F_Q limit specified in the Core Operating Limits Report (COLR) times the normalized axial peaking factor are not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

The limits on AXIAL FLUX DIFFERENCE (AFD) are given in the COLR. Two modes of operation are permissible. One mode is Normal Operation and the AFD limit is specified in the COLR. After extended load following maneuvers, the AFD limits may result in restrictions in the maximum allowed power to guarantee operation with $F_Q(Z)$ less than its limiting value. To prevent this occurrence, another operating mode which restricts the afd to a relatively small target

3/4.2.1 AXIAL FLUX DIFFERENCE (Continued)

band and does not allow significant changes in power level has been defined. This mode is called RESTRICTED AFD OPERATION, which restricts the AFD to the limits specified in the COLR and restricts power levels to between APL^{ND} and either APL^{RAFDO} or 100% of RATED THERMAL POWER, whichever is less. Prior to entering RESTRICTED AFD OPERATION, a 24-hour waiting period at a power level (±2%) above APL^{ND} and below that allowed by Normal Operation is necessary. During this time period load changes and control rod motion are restricted to that allowed by the RESTRICTED AFD OPERATION procedure. After the waiting period, RESTRICTED AFD OPERATION is permitted.

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits specified in the COLR while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are lss significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4.2-1 shows a typical monthly target band.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded, and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

(12½% for VANTAGE 5 fuel) and the appropriate fuel rod bow DNBR penalty (less than 1.5% per WCAP-8691, Rev. 1). The margin between design and safety analysis DNBR limits of 6.3% for Optimized fuel and 17.4% for VANTAGE 5 fuel includes greater than 3% margin for both Optimized fuel and VANTAGE 5 fuel for plant design flexibility.

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either Normal Operation or RESTRICTED AFD OPERATION, $W(z)_{NO}$ or $W(z)_{RAFDO}$, to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. $W(z)_{NO}$ accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(z)_{RAFDO}$ accounts for the more restrictive operating limits required by RESTRICTED AFD OPERATION which result in less severe transient values. The W(z) functions are specified in the Core Operating Limits Report per Specification 6.9.1.9.

Provisions to account for the possibility of decreases in margin to the $F_Q(z)$ limit during intervals between surveillances are provided. Any decrease in the minimum margin to the $F_Q(z)$ limit compared to the minimum margin determined from the previous flux map is determined by comparing the ratio of:

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right)$$

taken from the current map to the same ratio from the previous map. The ratios to be compared from the two flux maps do not need to be calculated at identical z locations. Increases in this ratio indicate that the minimum margin to the $F_Q(z)$ limit has decreased and that additional penalties must be applied to the measured $F_Q(z)$ to account for further decreases in margin that could occur before the next surveillance. More frequent surveillances may also be substituted for the additional penalty.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A

3/4.2.4 QUADRANT POWER TILT RATIO (Continued)

limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters is maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain the safety analysis DNBR limit throughout each analyzed transient. The indicated T_{avg} value of 592.6°F and the indicated pressurizer pressure value of 2220 psig correspond to analytical limits of 595.2°F and 2202 psig respectively, with allowance for measurement uncertainty.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

When RCS flow rate is measured, no additional allowances are necessary prior to comparison with the limits of Section 3.2.5. A measurement uncertainty of 2.2% (including 0.1% for feedwater venturi fouling) for RCS total flow rate has been allowed for in determination of the design DNBR value. The measurement uncertainty for the RCS total flow rate is based upon performing a precision heat balance and using the result to normalize the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a nonconservative manner. Therefore, an inspection is performed on the feedwater venturi each refueling outage.

MONTHLY OPERATING REPORT

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer PORVs or RCS safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

b. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,

c. Control Bank Insertion Limits for Specification 3/4.1.3.6,

d. Axial Flux Difference Limits, target band, and APL^{ND} for Specification 3/4.2.1,

e. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(Z), $W(Z)_{NO}$, APL^{ND} and $W(Z)_{RAFDO}$ for Specification 3/4.2.2,

f. Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}$, and Power Factor Factor Multiplier, $PF_{\Delta H}$, limits for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient 3.1.3.5 - Shutdown Bank Insertion Limit; 3.1.3.6 - Control Bank Insertion Limit; 3.2.1 - Axial Flux Difference; 3.2.2 - Heat Flux Hot Channel Factor; and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

b1. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT", September 1974 (W Proprietary).

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

CORE OPERATING LIMITS REPORT (Continued)

b2. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

b3. NUREG-0800, Standard Review Plan, U. S. Nuclear Regulatory Commission, Section 4.3 Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

c. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor (FQ Methodology for W(Z) surveillance requirements).

d. WCAP-10266-P-A, REV. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermalhydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special Reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In a-dition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

RECORD RETENTION (Continued)

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

d. Records of gaseous and liquid radioactive material released to the environs;

e. Records of transient or operational cycles for those unit components identified in Table 4.7-1;

f. Records of reactor tests and experiments;

g. Records of training and qualification for current members of the unit staff;

h. Records of in-service inspections performed pursuant to these Technical Specifications;

i. Records of quality assurance activities required by the QA Program;

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 58 TO FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY
CALLAWAY PLANT, UNIT 1
DOCKET NO. STN 50-483

1.0 INTRODUCTION

By letter dated March 6, 1990 (Ref. 1), Union Electric Company (UE), the licensee, proposed changes to the Technical Specifications (TS) for the Callaway Plant. The proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to a Core Operating Limits Report (COLR) for the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988 (Ref. 5).

Union Electric Company's original submittal was supplemented by letters dated June 5 (Ref. 2), July 5 (Ref. 3), and July 18, 1990 (Ref. 4). The June 5 submittal forwarded several administrative changes and one change to TS 3.1.1.3 as requested by the staff; all the changes forwarded by this submittal provided clarification to the licensee's original amendment request. The July 5 submittal provided a current set of proposed TS pages. The subject pages, II, XX, 1-4, 1-6, 3/4 1-15 and 6-21 were affected by two unrelated NRC-approved amendments subsequent to the licensee's original submittal. The replaced pages do not change the intent of the licensee's original submittal. The licensee's July 18 submittal corrects an error in the referenced analytical methods in TS section 6.9.1.9. The licensee's original submittal incorrectly referenced WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," as the analytical method used for TS 3.2.1 (AXIAL FLUX DIFFERENCE). The analytical method utilized for TS 3.2.1 is correctly referenced as WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," in the July 18 submittal. The corrected reference provided merely an administrative change to the licensee's original submittal. The three subsequent submittals described above provided clarification and corrections only; therefore, they do not impact the original proposed determination of no significant hazards consideration published in the Federal Register.

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2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(1) The Definitions section of the TS was modified to include a definition of the Core Operating Limits Report that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with NRC-approved methodologies that maintain the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

(a) Specification 3.1.1.3 and Surveillance Requirement 4.1.1.3

The moderator temperature coefficient (MTC) limits for this specification and for this surveillance requirement are specified in the COLR.

(b) Specification 3.1.3.5 and Surveillance Requirement 4.1.3.5

The shutdown bank insertion limit for this specification and for this surveillance requirement is specified in the COLR.

(c) Specification 3.1.3.6

The control bank insertion limits for this specification are specified in the COLR.

(d) Specification 3.2.1

The axial flux difference limits, target band, and the minimum allowable power level for (APL^{ND}) this specification are specified in the COLR.

(e) Specification 3.2.2 and Surveillance Requirement 4.2.2

The total peaking factor (F_Q) limit at rated thermal power, the normalized F_Q limit as a function of core height K(Z), the transient xenon effect on F_Q as a function of core height [$W(Z)_{NO}$ and $W(Z)_{RAEDO}$], and the allowed power level APL^{ND} for this specification and for this surveillance requirement are specified in the COLR.

(f) Specification 3.2.3 and Surveillance Requirement 4.2.3

The nuclear enthalpy rise hot channel factor (F-delta-H) limit at rated thermal power and the power factor multiplier (PF-delta-H) for this specification and surveillance requirement are specified in the COLR.

The bases of affected specifications have been modified by the licensee to include appropriate reference to the COLR. Based on its review, the staff concludes that the changes to these bases are acceptable.

(3) Specification 6.9.1.9 is revised to delete a previous reporting requirement on Peaking Factor Limit Report and to add the Core Operating Limits Report to the reporting requirements of the Administrative Controls section of the TS. The specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, these specifications require that the values of these limits be established using NRC-approved methodologies and be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:

(a) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (W Proprietary).

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limit, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

(b1) WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974 (W Proprietary).

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

(b2) T. M. Anderson to K. Kniel (NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

(b3) NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control)).

(c) WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification," June 1983 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor [F_Q Methodology for W(Z) surveillance requirements]).

(d) WCAP-10266-P-A, Rev. 2, "The 1981 Version of Westinghouse Evaluation Model Using BASH Code," March 1987 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TSs. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC-approved methodologies, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable.

A change was also made to the Action Statement of Specification 3.1.3.1. Action c2 was changed to reference Specification 3.1.3.6 instead of Figure 3.1.1. This change was necessary because the figure has been relocated to the COLR. Consequently, this change is administrative and, therefore, acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or a change to a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). This amendment also involves changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Letter (ULNRC-2168) from D. F. Schnell (UE) to NRC, dated March 6, 1990.

2. Letter (ULNRC-2223) from D. F. Schnell (UE) to NRC, dated June 5, 1990.

3. Letter (ULNRC-2241) from D. F. Schnell (UE) to NRC, dated July 5, 1990.

4. Letter (ULNRC-2253) from D. F. Schnell (UE) to NRC, dated July 18, 1990.

5. Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988.

Principal Contributor: Daniel Fieno

Dated: November 1, 1990