August 28, 1998

Mr. Gary R. Peterson
Site Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745-9635

Distribution
Docket File
PUBLIC
PDII-2 RF
J. Zwolinski
L.Plisco, RII
COgle, RII
ACRS T-2 E26
OGC
G.Hill(4)
e-mail TLH3 SE only
TSB, O-11 F23

SUBJECT: ISSUANCE OF AMENDMENTS - CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (TAC NOS. M99629 AND M99630)

Dear Mr. Peterson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 170 to Facility Operating License NPF-35 and Amendment No. 162 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments are in response to your application dated September 15, 1997, as supplemented by letters dated March 5, April 27, June 15, July 22, and August 10, 1998.

The amendments revise Figures 3.4-2 and 3.4-3 (pressure-temperature limits curves), Table 4.4-5 (reactor vessel surveillance capsule withdrawal schedule), and Sections 3/4.4.9.3 and 3.5.3 (requirements concerning overpressure protection) of the Technical Specifications. The associated Bases are also revised.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:
Peter S. Tam, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 170 to NPF-35
2. Amendment No. 162 to NPF-52
3. Safety Evaluation

cc w/encl: See next page

DOCUMENT NAME: G:\CATAWBA\CAT99629.AMD

OFFICE	PDII-2/PM	PDII-2/LA	OGC	PDII-2/D	
NAME	P.TAM:cn	L.BERRY		H.BERKOW	
DATE	8/18/98	8/18/98	8/28/98	8/24/98	/ /98
COPY	YES NO		YES NO	YES NO	YES NO

OFFICIAL RECORD COPY

9809040125 980828
PDR ADOCK 05000413
P PDR

CP-1

August 28, 1998

Mr. Gary R. Peterson
Site Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745-9635

Distribution
Docket File
PUBLIC
PDII-2 RF
J. Zwolinski
L.Plisco, RII
COgle, RII
ACRS T-2 E26
OGC
G.Hill(4)
e-mail TLH3 SE only
TSB, O-11 F23

SUBJECT: ISSUANCE OF AMENDMENTS - CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (TAC NOS. M99629 AND M99630)

Dear Mr. Peterson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 170 to Facility Operating License NPF-35 and Amendment No. 162 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments are in response to your application dated September 15, 1997, as supplemented by letters dated March 5, April 27, June 15, July 22, and August 10, 1998.

The amendments revise Figures 3.4-2 and 3.4-3 (pressure-temperature limits curves), Table 4.4-5 (reactor vessel surveillance capsule withdrawal schedule), and Sections 3/4.4.9.3 and 3.5.3 (requirements concerning overpressure protection) of the Technical Specifications. The associated Bases are also revised.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:
Peter S. Tam, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 170 to NPF-35
2. Amendment No. 162 to NPF-52
3. Safety Evaluation

cc w/encl: See next page

DOCUMENT NAME: G:\CATAWBA\CAT99629.AMD

OFFICE	PDII-2/PM	PDII-2/LA	OGC	PDII-2/D	
NAME	PTAM:cn	L.BERRY		H.BERKOW	
DATE	8/18/98	8/18/98	8/28/98	8/24/98	/98
COPY	YES NO		YES NO	YES NO	YES NO

OFFICIAL RECORD COPY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 28, 1998

Mr. Gary R. Peterson
Site Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745-9635

SUBJECT: ISSUANCE OF AMENDMENTS - CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (TAC NOS. M99629 AND M99630)

Dear Mr. Peterson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 170 to Facility Operating License NPF-35 and Amendment No. 162 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments are in response to your application dated September 15, 1997, as supplemented by letters dated March 5, April 27, June 15, July 22, and August 10, 1998.

The amendments revise Figures 3.4-2 and 3.4-3 (pressure-temperature limits curves), Table 4.4-5 (reactor vessel surveillance capsule withdrawal schedule), and Sections 3/4.4.9.3 and 3.5.3 (requirements concerning overpressure protection) of the Technical Specifications. The associated Bases are also revised.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Peter S. Tam

Peter S. Tam, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 170 to NPF-35
2. Amendment No. 162 to NPF-52
3. Safety Evaluation

cc w/encl: See next page

Catawba Nuclear Station

cc:

Mr. M. S. Kitlan
Regulatory Compliance Manager
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Mr. Paul R. Newton
Legal Department (PB05E)
Duke Energy Corporation
422 South Church Street
Charlotte, North Carolina 28201-1006

J. Michael McGarry, III, Esquire
Winston and Strawn
1400 L Street, NW
Washington, DC 20005

North Carolina Municipal Power
Agency Number 1
1427 Meadowwood Boulevard
P. O. Box 29513
Raleigh, North Carolina 27626

County Manager of York County
York County Courthouse
York, South Carolina 29745

Piedmont Municipal Power Agency
121 Village Drive
Greer, South Carolina 29651

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of Justice
P. O. Box 629
Raleigh, North Carolina 27602

Elaine Wathen, Lead REP Planner
Division of Emergency Management
116 West Jones Street
Raleigh, North Carolina 27603-1335

North Carolina Electric Membership
Corporation
P. O. Box 27306
Raleigh, North Carolina 27611

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
4830 Concord Road
York, South Carolina 29745

Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
Atlanta Federal Center
61 Forsyth Street, S.W., Suite 23T85
Atlanta, Georgia 30303

Virgil R. Autry, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health and Environmental
Control
2600 Bull Street
Columbia, South Carolina 29201-1708

L. A. Keller
Manager - Nuclear Regulatory
Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

Saluda River Electric
P. O. Box 929
Laurens, South Carolina 29360

Peter R. Harden, IV
Account Manager
Energy Systems Sales
Westinghouse Electric Corporation
P. O. Box 7288
Charlotte, North Carolina 28241-7288

Catawba Nuclear Station

cc:

Mr. T. Richard Puryear
Owners Group (NCEMC)
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Richard M. Fry, Director
Division of Radiation Protection
North Carolina Department of
Environment, Health, and
Natural Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CORPORATION

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

SALUDA RIVER ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 170
License No. NPF-35

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-35 filed by the Duke Energy Corporation, acting for itself, North Carolina Electric Membership Corporation and Saluda River Electric Cooperative, Inc. (licensees), dated September 15, 1997, as supplemented by letters dated March 5, April 27, June 15, July 22, and August 10, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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.

9809040133 980828
PDR ADOCK 05000413
P PDR

2. Accordingly, the license is hereby amended by page 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-35 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 170, which are attached hereto, are hereby incorporated into this license. Duke Energy Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: August 28, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 170

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

Replace the following pages of the Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
VII	VII
VIII	VIII
3/4 4-32 (Figure 3.4-2)	3/4 4-32 (Figure 3.4-2)
3/4 4-33 (Figure 3.4-3)	3/4 4-33 (Figure 3.4-3)
3/4 4-34 (Table 4.4-5)	3/4 4-34 (Table 4.4-5)
3/4 4-36	3/4 4-36
--	3/4 4-36a
3/4 4-37	3/4 4-37
3/4 5-8	3/4 5-8
B 3/4 4-9	B 3/4 4-9
B 3/4 4-14	B 3/4 4-14
B 3/4 4-15	B 3/4 4-15

SECTION PAGE

3/4.4.3 PRESSURIZER 3/4 4-9

3/4.4.4 RELIEF VALVES 3/4 4-10

3/4.4.5 STEAM GENERATORS 3/4 4-12

TABLE 4.4-1 MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION 3/4 4-17

TABLE 4.4-2 STEAM GENERATOR TUBE INSPECTION 3/4 4-18

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

Leakage Detection Systems 3/4 4-19

Operational Leakage 3/4 4-20

TABLE 3.4-1 REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES 3/4 4-22

3/4.4.7 CHEMISTRY 3/4 4-24

TABLE 3.4-2 REACTOR COOLANT SYSTEM CHEMISTRY LIMITS 3/4 4-25

TABLE 4.4-3 REACTOR COOLANT SYSTEM CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS 3/4 4-26

3/4.4.8 SPECIFIC ACTIVITY 3/4 4-27

FIGURE 3.4-1 DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 μCi/gram DOSE EQUIVALENT I-131 . . . . . 3/4 4-28

TABLE 4.4-4 REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM 3/4 4-29

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

Reactor Coolant System 3/4 4-31

FIGURE 3.4-2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE UP TO 15 EFPY 3/4 4-32

FIGURE 3.4-3 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 15 EFPY 3/4 4-33

SECTION PAGE

TABLE 4.4-5 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM WITHDRAWAL SCHEDULE 3/4 4-34

Pressurizer 3/4 4-35

Overpressure Protection Systems 3/4 4-36

TABLE 3.4-3 REACTOR COOLANT PUMP OPERATING RESTRICTIONS FOR LOW TEMPERATURE OVERPRESSURE PROTECTION 3/4 4-36a

3/4.4.10 STRUCTURAL INTEGRITY 3/4 4-38

3/4.4.11 REACTOR COOLANT SYSTEM VENTS 3/4 4-39

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

Cold Leg Injection 3/4 5-1

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$ 3/4 5-4

3/4.5.3 ECCS SUBSYSTEMS - $T_{avg} < 350°F$ 3/4 5-8

3/4.5.4 REFUELING WATER STORAGE TANK 3/4 5-10

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

Containment Integrity 3/4 6-1

Containment Leakage 3/4 6-2

TABLE 3.6-1 SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS 3/4 6-5

Containment Air Locks 3/4 6-8

Internal Pressure 3/4 6-10

Air Temperature 3/4 6-11

Containment Vessel Structural Integrity 3/4 6-12

Reactor Building Structural Integrity 3/4 6-13

Annulus Ventilation System 3/4 6-14

Containment Purge Systems 3/4 6-16

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART AT 15 EFPY: 1/4-T, 43°F
3/4-T, 26°F

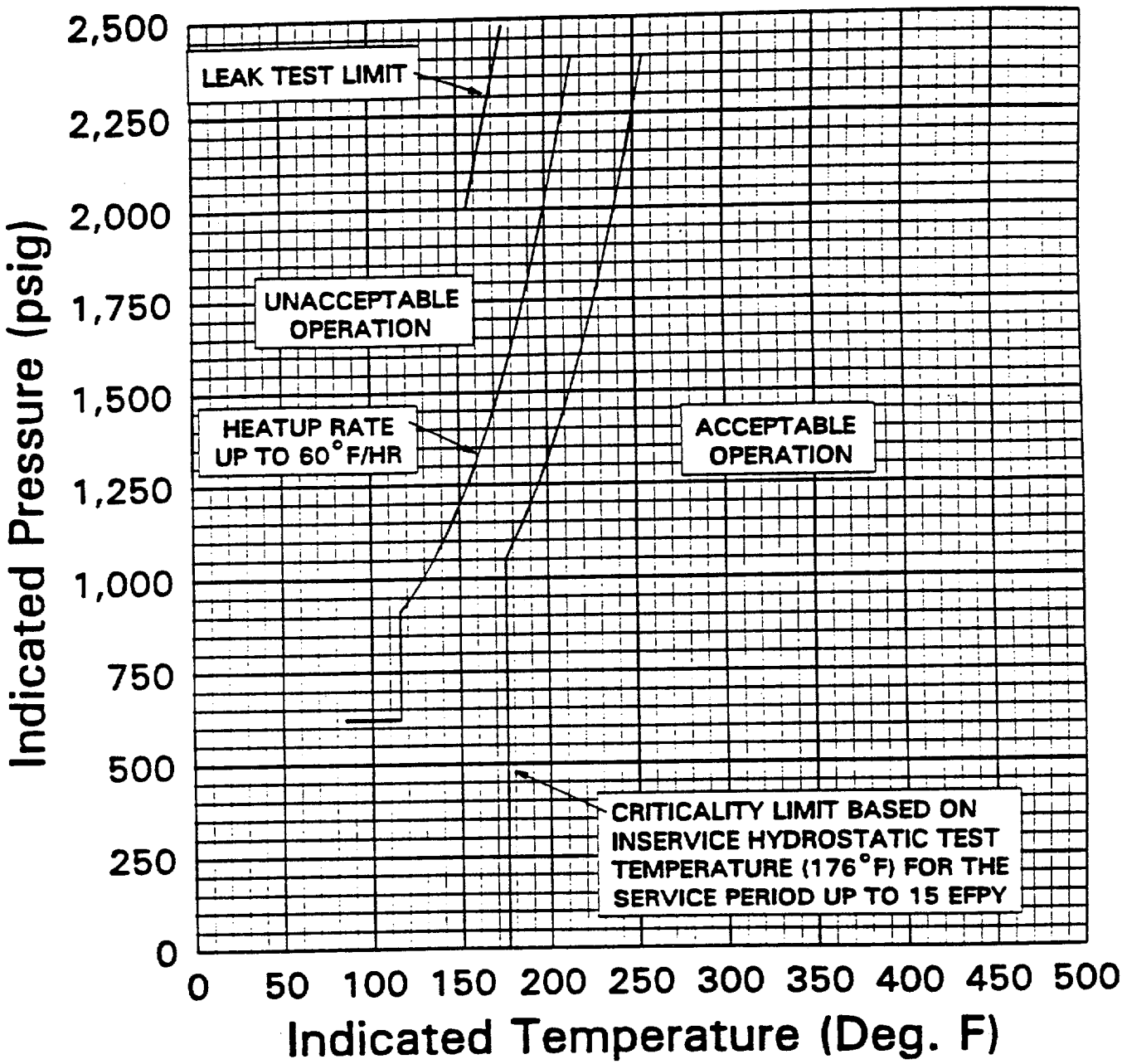

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS (Heatup rates up to 60°F/hr)
APPLICABLE FOR THE FIRST 15 EFPY (Without Margins For Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART AT 15 EFPY: 1/4-T, 43°F
3/4-T, 26°F

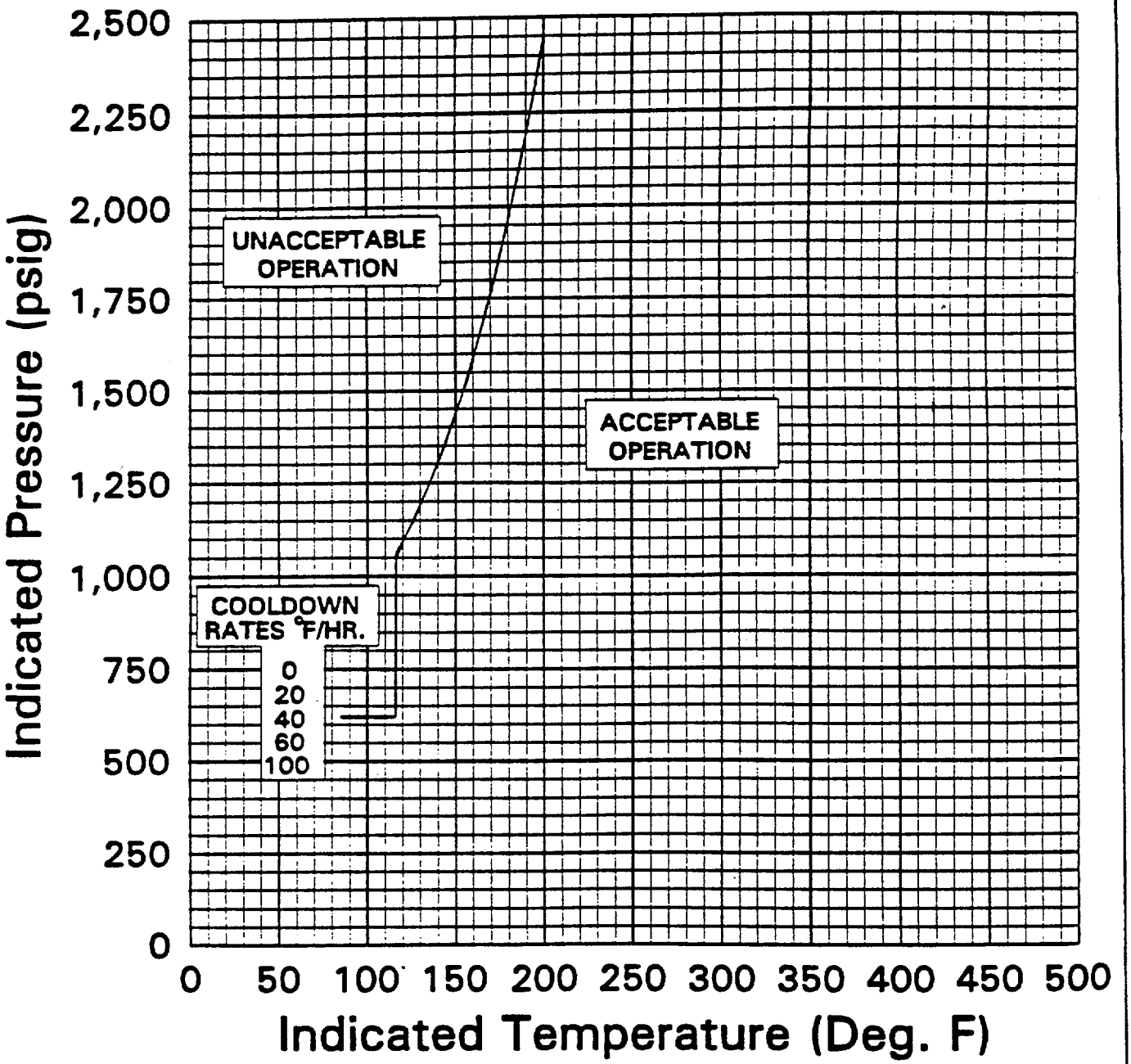

FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS (Cooldown rates up to 100°F/hr)
APPLICABLE FOR THE FIRST 15 EFPY (Without Margins For Instrumentation Errors)

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWN TIME (EFPY)
U	58.5°	3.91	Standby
V	61°	3.66	9
W	121.5°	3.91	13
X	238.5°	3.91	Standby
Y	241°	3.66	4.98
Z	301.5°	4.10	0.79

LIMITING CONDITION FOR OPERATION

3.4.9.3 A Low Temperature Overpressure Protection System shall be OPERABLE with a maximum of one charging pump or one safety injection pump capable of injecting into the Reactor Coolant System, the cold leg accumulators isolated*, Reactor Coolant pump operation limited as specified in Table 3.4-3, and either a or b below:

a. Two power operated relief valves (PORVs)# with a lift setting of less than or equal to 400 psig (as left calibrated), allowable value of less than or equal to 425 psig (as found), or

b. The Reactor Coolant System depressurized with a Reactor Coolant System vent of greater than or equal to 4.5 square inches.

APPLICABILITY: MODE 4 when the temperature of any Reactor Coolant System cold leg is less than or equal to 285°F, MODE 5 and MODE 6 when the head is on the reactor vessel.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or complete depressurization and venting of the Reactor Coolant System through at least a 4.5 square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, restore the inoperable PORV to OPERABLE status within 24 hours or complete depressurization and venting of the Reactor Coolant System through at least a 4.5 square inch vent within the next 8 hours.

c. With both PORVs inoperable, complete depressurization and venting of the Reactor Coolant System through at least a 4.5 square inch vent within 8 hours.

---

* Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum Reactor Coolant System pressure for the existing Reactor Coolant System cold leg temperature allowed by the pressure/temperature limit curves provided in Specification 3/4.4.9.

When using the PORVs to meet the requirements of this Specification, each Reactor Coolant System cold leg temperature shall be greater than or equal to 65°F. When the Reactor Coolant pumps are secured, this temperature shall be measured at the residual heat removal heat exchanger outlet.

ACTION (Continued)

d. In the event either the PORVs or the Reactor Coolant System vent(s) are used to mitigate a Reactor Coolant System pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or Reactor Coolant System vent(s) on the transient, and any corrective action necessary to prevent recurrence.

e. The provisions of Specification 3.0.4 are not applicable.

f. With more than the specified number of charging pumps and/or safety injection pumps capable of injecting into the Reactor Coolant System*, immediately initiate action to verify a maximum of one charging pump or one safety injection pump is capable of injecting into the Reactor Coolant System.

g. With a cold leg accumulator not isolated as specified, within one hour, isolate the affected accumulator. If the affected accumulator is not isolated within one hour, within the next 12 hours, either increase Reactor Coolant System cold leg temperature to greater than 285°F, or depressurize the affected accumulator to less than the maximum Reactor Coolant System pressure for the existing cold leg temperature allowed by Specification 3/4.4.9.

h. With Reactor Coolant pump operation not in accordance with the requirements of Table 3.4-3, immediately initiate action to limit pump operation as specified in Table 3.4-3.

TABLE 3.4-3

REACTOR COOLANT PUMP OPERATING RESTRICTIONS FOR LOW TEMPERATURE OVERPRESSURE PROTECTION

Reactor Coolant System Cold Leg Temperature	Maximum Number of Pumps Allowed in Operation
≥ 67°F	1
≥ 68°F	2
≥ 73°F	4

*Two charging pumps may be capable of injecting into the Reactor Coolant System during pump swap operation for less than or equal to 15 minutes.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The Reactor Coolant System vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

4.4.9.3.3 A maximum of one charging pump or safety injection pump shall be verified to be capable of injecting into the Reactor Coolant System at least once per 12 hours.

4.4.9.3.4 Each cold leg accumulator shall be verified to be isolated at least once per 12 hours when accumulator isolation is required.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} < 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

#A maximum of one charging pump or one Safety Injection pump shall be OPERABLE whenever the temperature of one or more of the Reactor Coolant System cold legs is less than or equal to 285°F.

PRESSURE/TEMPERATURE LIMITS (Continued)

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of the effective full power years (EFPY) of service life as indicated on the applicable heatup or cooldown curves. The service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron ($\bar{E}$ greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT}.

The adjusted reference temperature has been computed using the guidance of Regulatory Guide 1.99, Revision 2. The heatup and cooldown limit curves in Figures 3.4-2 and 3.4-3 include predicted adjustments for the shift in RT_{NDT} at the end of the identified service life.

Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the pressure vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the pressure vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in the following paragraphs.

PRESSURE/TEMPERATURE LIMITS (Continued)

defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

PRESSURE/TEMPERATURE LIMITS (Continued)

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs or a Reactor Coolant System vent opening of at least 4.5 square inches ensures that the Reactor Coolant System will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the cold legs are less than or equal to 285°F. Either PORV has adequate relieving capability to protect the Reactor Coolant System from overpressurization when the transient is limited to either: (1) the start of an idle reactor coolant pump with the secondary water temperature of the steam generator less than or equal to 50°F above the cold leg temperatures, or (2) the start of a Safety Injection pump and its injection into a water solid Reactor Coolant System.

The Maximum Allowed PORV Setpoint for the Low Temperature Overpressure Protection System (LTOPS) is derived by analysis which models the performance of the LTOPS assuming various mass input and heat input transients and incorporates instrument uncertainties as well as corrections for Reactor Coolant Pump operation and the static pressure difference between the Reactor Vessel Beltline Region and the location of the pressure transmitters used for LTOP. Operation with a PORV Setpoint less than or equal to the maximum allowable value of 425 psig (as found) ensures that Appendix G criteria will not be violated with consideration for a maximum pressure overshoot beyond the PORV Setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all but one Safety Injection pump and all but one centrifugal charging pump while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of a RCP if secondary temperature is more than 50°F above primary temperature.

The Maximum Allowed PORV setpoint for the LTOPS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 4.4-5.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CORPORATION

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 162
License No. NPF-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-52 filed by the Duke Energy Corporation, acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees), dated September 15, 1997, as supplemented by letters dated March 5, April 27, June 15, July 22, and August 10, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page 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-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 162 , which are attached hereto, are hereby incorporated into this license. Duke Energy Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: August 28, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 162

FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Technical Specifications with the enclosed pages. The revised pages are dentified by Amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
VII	VII
VIII	VIII
3/4 4-33 (Figure 3.4-2)	3/4 4-33 (Figure 3.4-2)
3/4 4-34 (Figure 3.4-3)	3/4 4-34 (Figure 3.4-3)
3/4 4-35 (Table 4.4-5)	3/4 4-35 (Table 4.4-5)
3/4 4-37	3/4 4-37
--	3/4 4-37a
3/4 4-38	3/4 4-38
3/4 5-8	3/4 5-8
B 3/4 4-9	B 3/4 4-9
B 3/4 4-14	B 3/4 4-14
B 3/4 4-15	B 3/4 4-15

SECTION PAGE

3/4.4.3 PRESSURIZER 3/4 4-9

3/4.4.4 RELIEF VALVES 3/4 4-10

3/4.4.5 STEAM GENERATORS 3/4 4-12

TABLE 4.4-1 MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION 3/4 4-18

TABLE 4.4-2 STEAM GENERATOR TUBE INSPECTION 3/4 4-19

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

Leakage Detection Systems 3/4 4-20

Operational Leakage 3/4 4-21

TABLE 3.4-1 REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES 3/4 4-23

3/4.4.7 CHEMISTRY 3/4 4-25

TABLE 3.4-2 REACTOR COOLANT SYSTEM CHEMISTRY LIMITS 3/4 4-26

TABLE 4.4-3 REACTOR COOLANT SYSTEM CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS 3/4 4-27

3/4.4.8 SPECIFIC ACTIVITY 3/4 4-28

FIGURE 3.4-1 DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 μCi/gram DOSE EQUIVALENT I-131 . . . . . 3/4 4-29

TABLE 4.4-4 REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM 3/4 4-30

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

Reactor Coolant System 3/4 4-32

FIGURE 3.4-2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE UP TO 15 EFPY 3/4 4-33

FIGURE 3.4-3 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 15 EFPY 3/4 4-34

SECTION PAGE

TABLE 4.4-5 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM WITHDRAWAL SCHEDULE 3/4 4-35

Pressurizer 3/4 4-36

Overpressure Protection Systems 3/4 4-37

TABLE 3.4-3 REACTOR COOLANT PUMP OPERATING RESTRICTIONS FOR LOW TEMPERATURE OVERPRESSURE PROTECTION 3/4 4-37a

3/4.4.10 STRUCTURAL INTEGRITY 3/4 4-39

3/4.4.11 REACTOR COOLANT SYSTEM VENTS 3/4 4-40

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

Cold Leg Injection 3/4 5-1

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$ 3/4 5-4

3/4.5.3 ECCS SUBSYSTEMS - $T_{avg} < 350°F$ 3/4 5-8

3/4.5.4 REFUELING WATER STORAGE TANK 3/4 5-10

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

Containment Integrity 3/4 6-1

Containment Leakage 3/4 6-2

TABLE 3.6-1 SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS 3/4 6-5

Containment Air Locks 3/4 6-8

Internal Pressure 3/4 6-10

Air Temperature 3/4 6-11

Containment Vessel Structural Integrity 3/4 6-12

Reactor Building Structural Integrity 3/4 6-13

Annulus Ventilation System 3/4 6-14

Containment Purge Systems 3/4 6-16

MATERIAL PROPERTY BASIS

LIMITING MATERIALS: INTERMEDIATE SHELL, B8605-2
LIMITING ART AT 15 EFPY: 1/4-t, 112.6 °F
3/4-t, 96.0 °F

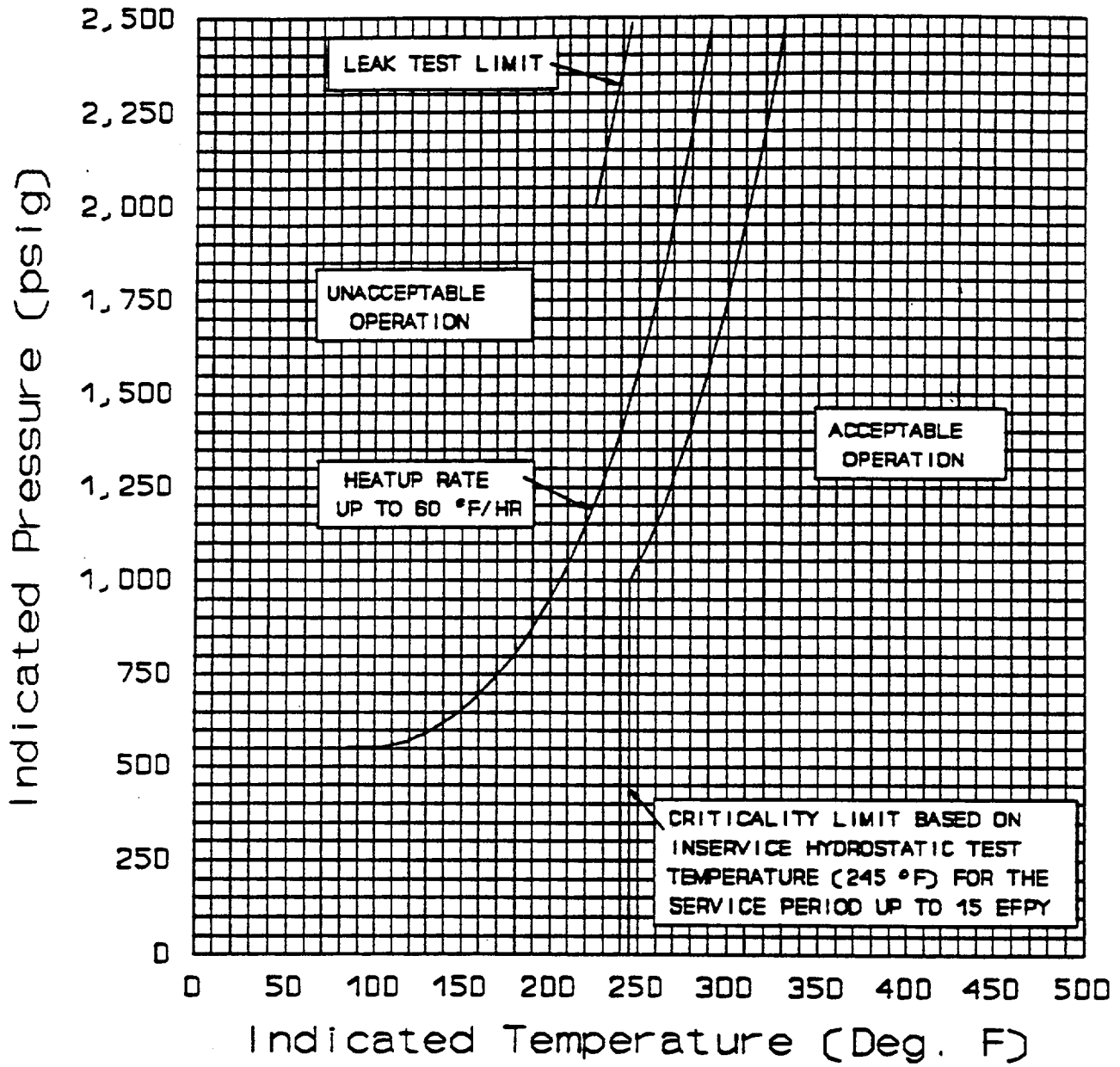

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS (Heatup rates up to 60°F/hr)
APPLICABLE FOR THE FIRST 15 EFPY (Without Margins For Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIALS: INTERMEDIATE SHELL, B8605-2
LIMITING ART AT 15 EFPY: 1/4-t, 112.6 °F
3/4-t, 96.0 °F

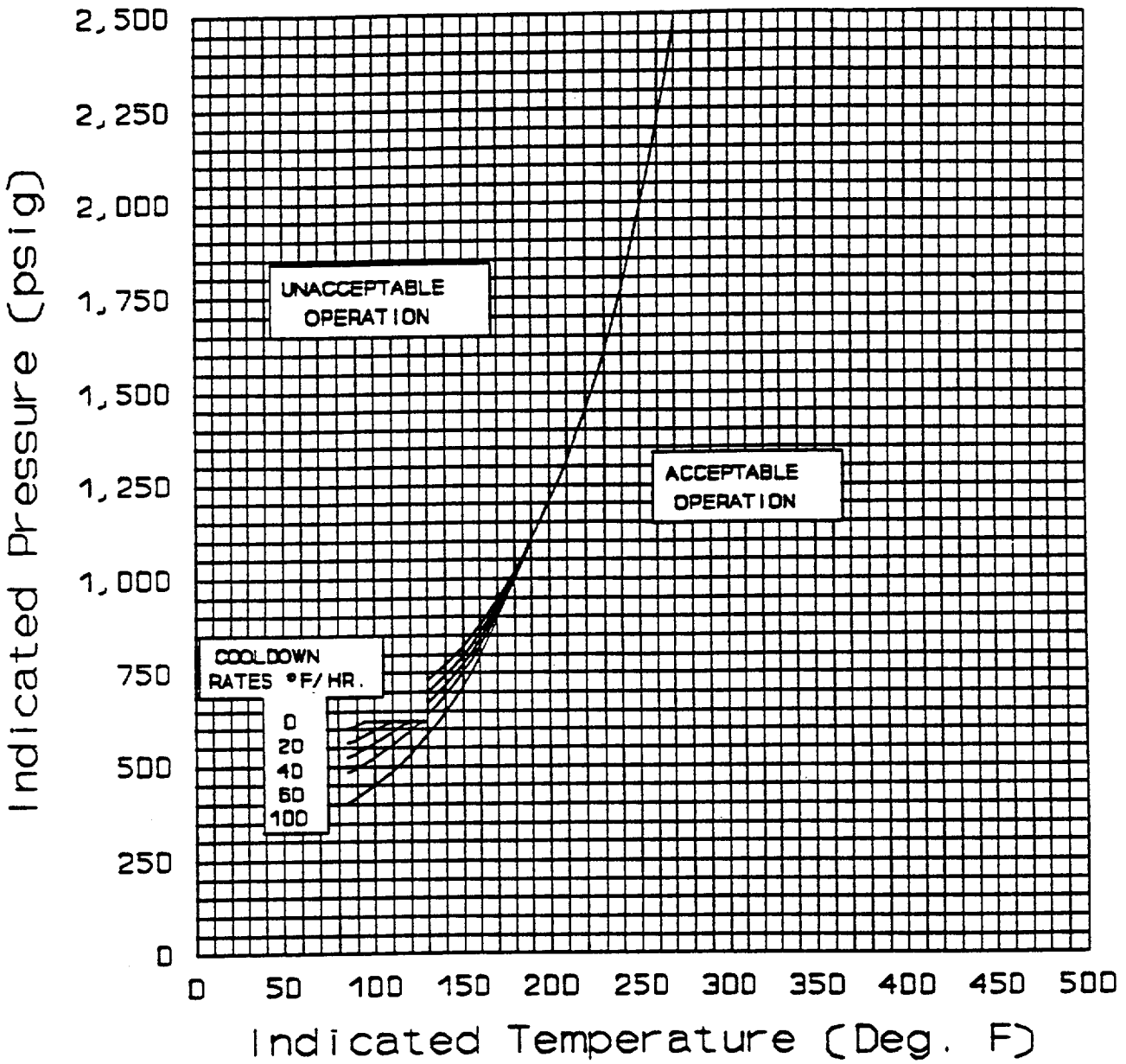

FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS (Cooldown rates up to 100°F/hr) APPLICABLE FOR THE FIRST 15 EFPY (Without Margins For Instrumentation Errors)

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWN TIME (EFPY)
U	58.5°	3.79	Standby
V	61°	3.62	9
W	121.5°	3.79	13.5
Y	238.5°	3.79	Standby
X	241°	3.62	4.52
Z	301.5°	4.09	0.86

LIMITING CONDITION FOR OPERATION

3.4.9.3 A Low Temperature Overpressure Protection System shall be OPERABLE with a maximum of one charging pump or one safety injection pump capable of injecting into the Reactor Coolant System, the cold leg accumulators isolated*, Reactor Coolant pump operation limited as specified in Table 3.4-3, and either a or b below:

a. Two power operated relief valves (PORVs)# with a lift setting of less than or equal to 400 psig (as left calibrated), allowable value of less than or equal to 425 psig (as found), or

b. The Reactor Coolant System depressurized with a Reactor Coolant System vent of greater than or equal to 4.5 square inches.

APPLICABILITY: MODE 4 when the temperature of any Reactor Coolant System cold leg is less than or equal to 285°F, MODE 5 and MODE 6 when the head is on the reactor vessel.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or complete depressurization and venting of the Reactor Coolant System through at least a 4.5 square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, restore the inoperable PORV to OPERABLE status within 24 hours or complete depressurization and venting of the Reactor Coolant System through at least a 4.5 square inch vent within the next 8 hours.

c. With both PORVs inoperable, complete depressurization and venting of the Reactor Coolant System through at least a 4.5 square inch vent within 8 hours.

---

* Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum Reactor Coolant System pressure for the existing Reactor Coolant System cold leg temperature allowed by the pressure/temperature limit curves provided in Specification 3/4.4.9.

When using the PORVs to meet the requirements of this Specification, each Reactor Coolant System cold leg temperature shall be greater than or equal to 65°F. When the Reactor Coolant pumps are secured, this temperature shall be measured at the residual heat removal heat exchanger outlet.

ACTION (Continued)

d. In the event either the PORVs or the Reactor Coolant System vent(s) are used to mitigate a Reactor Coolant System pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or Reactor Coolant System vent(s) on the transient, and any corrective action necessary to prevent recurrence.

e. The provisions of Specification 3.0.4 are not applicable.

f. With more than the specified number of charging pumps and/or safety injection pumps capable of injecting into the Reactor Coolant System*, immediately initiate action to verify a maximum of one charging pump or one safety injection pump is capable of injecting into the Reactor Coolant System.

g. With a cold leg accumulator not isolated as specified, within one hour, isolate the affected accumulator. If the affected accumulator is not isolated within one hour, within the next 12 hours, either increase Reactor Coolant System cold leg temperature to greater than 285°F, or depressurize the affected accumulator to less than the maximum Reactor Coolant System pressure for the existing cold leg temperature allowed by Specification 3/4.4.9.

h. With Reactor Coolant pump operation not in accordance with the requirements of Table 3.4-3, immediately initiate action to limit pump operation as specified in Table 3.4-3.

TABLE 3.4-3

REACTOR COOLANT PUMP OPERATING RESTRICTIONS FOR LOW TEMPERATURE OVERPRESSURE PROTECTION

Reactor Coolant System Cold Leg Temperature	Maximum Number of Pumps Allowed in Operation
≥ 67°F	1
≥ 73°F	2
≥ 95°F	4

*Two charging pumps may be capable of injecting into the Reactor Coolant System during pump swap operation for less than or equal to 15 minutes.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The Reactor Coolant System vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

4.4.9.3.3 A maximum of one charging pump or safety injection pump shall be verified to be capable of injecting into the Reactor Coolant System at least once per 12 hours.

4.4.9.3.4 Each cold leg accumulator shall be verified to be isolated at least once per 12 hours when accumulator isolation is required.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} < 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

#A maximum of one charging pump or one Safety Injection pump shall be OPERABLE whenever the temperature of one or more of the Reactor Coolant System cold legs is less than or equal to 285°F.

PRESSURE/TEMPERATURE LIMITS (Continued)

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of the effective full power years (EFPY) of service life as indicated on the applicable heatup or cooldown curves. The service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT}.

The adjusted reference temperature has been computed using the guidance of Regulatory Guide 1.99, Revision 2. The heatup and cooldown limit curves in Figures 3.4-2 and 3.4-3 include predicted adjustments for the shift in RT_{NDT} at the end of the identified service life.

Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the pressure vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the pressure vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in the following paragraphs.

PRESSURE/TEMPERATURE LIMITS (Continued)

defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

PRESSURE/TEMPERATURE LIMITS (Continued)

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs or a Reactor Coolant System vent opening of at least 4.5 square inches ensures that the Reactor Coolant System will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the cold legs are less than or equal to 285°F. Either PORV has adequate relieving capability to protect the Reactor Coolant System from overpressurization when the transient is limited to either: (1) the start of an idle reactor coolant pump with the secondary water temperature of the steam generator less than or equal to 50°F above the cold leg temperatures, or (2) the start of a Safety Injection pump and its injection into a water solid Reactor Coolant System.

The Maximum Allowed PORV Setpoint for the Low Temperature Overpressure Protection System (LTOPS) is derived by analysis which models the performance of the LTOPS assuming various mass input and heat input transients and incorporates instrument uncertainties as well as corrections for Reactor Coolant Pump operation and the static pressure difference between the Reactor Vessel Beltline Region and the location of the pressure transmitters used for LTOP. Operation with a PORV Setpoint less than or equal to the maximum allowable value of 425 psig (as found) ensures that Appendix G criteria will not be violated with consideration for a maximum pressure overshoot beyond the PORV Setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all but one Safety Injection pump and all but one centrifugal charging pump while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of a RCP if secondary temperature is more than 50°F above primary temperature.

The Maximum Allowed PORV setpoint for the LTOPS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 4.4-5.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 162 TO FACILITY OPERATING LICENSE NPF-52

DUKE ENERGY CORPORATION, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated September 15, 1997, as supplemented by letters dated March 5, April 27, June 15, July 22, and August 10, 1998, Duke Energy Corporation, et al. (the licensee, DEC) submitted a request for amendments to the Catawba Nuclear Station, Units 1 and 2, Technical Specifications (TS), revising the units' pressure-temperature (P-T) limits curves in the TS, and extending the period of their applicability from 10 to 15 effective full power years (EFPYs) of operation. On February 6 and March 26, 1998, the staff issued requests for additional information. On March 5, April 27, June 15, July 22, and August 10, 1998, the licensee supplemented the amendment request. The June 15, 1998, supplement also expanded the scope of the licensee's amendment request, proposing changes and additional requirements to address low-temperature overpressure protection of the primary coolant system.

The March 5, April 27, July 22, and August 10, 1998, letters provided additional information that did not change the scope of the September 15, 1997, application and the initial proposed no significant hazards consideration determination.

2.0 DISCUSSION AND EVALUATION

The staff reviewed the licensee's submittals in three specific areas. These areas are addressed as follows in Sections 2.1, 2.2, and 2.3.

2.1 Neutron Fluence

The licensee reported that the fluence evaluation, which is the basis for the current as well as the proposed revised P-T curves, was performed when the first four surveillance capsules were removed and evaluated at the end of about 1 and 4 EFPYs. The results are documented in Westinghouse Topical Reports WCAP-13720 ("Analysis of Capsule Y from the Duke Power Company Catawba Unit 1 Reactor Vessel Radiation Surveillance Program" by J. M. Chicots, et al., Westinghouse Electric Corporation, June 1993), and WCAP-13875 ("Analysis of Capsule X from the Duke Power Company Catawba Unit 2 Reactor Vessel Radiation Surveillance Program," by E. Terek, et al., Westinghouse Electric Corporation, February

9809040139 980828
PDR ADOCK 05000413
P PDR

1994). Both these topical reports were previously submitted to the NRC docket files and are available to the public.

At the time these evaluations were performed, ENDF/B-IV based nuclear cross sections were used. Since that time, the staff has identified a number of nonconservative values in the iron inelastic cross sections. Currently the staff-recommended cross sections are based on the ENDF/B-VI cross section file. For plants like Catawba, which are equipped with neutron pads (partial thermal shields), ENDF/B-VI based cross sections yield calculated fluence values about 15 percent higher than those based on ENDF/B-IV. This observation is based on the staff's experience with a large number of computations of fluence values.

In the calculation of the current P-T curves, the licensee assumed fluence values based on the measured values, which were higher than the calculated fluence. The new proposed fluence values are based on the measured values. However, had the fluences been recalculated using cross sections based on ENDF/B-VI, the resulting values would have been up to 3 percent lower than the measured values. Therefore, the licensee's proposed values are up to 3 percent more conservative compared to the calculated values.

At the Catawba reactors, dosimeters were located in the surveillance capsules, which were fastened outside the neutron pads (partial thermal shields). Based on the staff's experience, fluence values calculated using the ENDF/B-VI based cross sections and accounting for the presence of the thermal shield, would be about 15 percent higher. WCAP-13720 and WCAP-13875, which describe the analysis of surveillance capsule Y and summary of capsule Z for Catawba Unit 1, and analysis of surveillance capsule X and summary of capsule Z for Unit 2, established that the measured values were indeed higher than the corresponding calculated values by 15 to 18 percent. As previously stated, the fluence for the current P-T curves and the proposed P-T curves is based on the measured values. Therefore, the proposed fluence values would be up to 3 percent higher than the values calculated using the ENDF/B-VI cross sections. The staff, therefore, concludes that the proposed fluence values are conservative and are, thus, acceptable.

2.2 Proposed New P-T Limits Curves

The staff evaluated the P-T limits based on the following NRC regulations and guidance: Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix G, "Fracture Toughtness Requirements"; Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations"; GL 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity"; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2), "Radiation Embrittlement of Reactor Vessel Materials;" and Standard Review Plan (SRP) Section 5.3.2, "Pressure-Temperature Limits." GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Revision 1, requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the staff. GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the

staff as the basis for the staff's review of P-T limit curves, and as the basis for the staff's review of pressurized thermal shock (PTS) assessments (10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code).

SRP 5.3.2 provides an acceptable method of calculating the P-T limits for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to one-fourth of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T limit curves are the one-fourth thickness (1/4T) and three-fourth thickness (3/4T) locations, which correspond to the depth of the maximum postulated flaw, if initiated and grown from the inside and outside surfaces of the RPV, respectively.

The Appendix G, ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term.

The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

Unit 1

For the Unit 1 reactor vessel, the licensee determined that the most limiting material at the 1/4T and 3/4T locations is the lower shell forging plate 04. This plate was fabricated using plate heat 527708. The licensee calculated an ART of 43 °F at the 1/4T location and 26 °F at the 3/4T location at 15 EFPYs. The neutron fluence used in the ART calculation was 7.1×10^{18} n/cm^2 at the 1/4T location and 2.57×10^{18} n/cm^2 at the 3/4T location. The initial RT_{NDT} for the limiting plate was -13 °F. The margin term used in calculating the ART for the limiting plate was 28 °F at the 1/4T location and 19.6 °F at the 3/4T location, as permitted by Position 1.1 of RG 1.99, Rev. 2.

The ART is determined using the chemistry values for each beltline material of Unit 1. The Reactor Vessel Integrity Database (RVID) contains chemistry values for each beltline material for all light water reactors in the U.S. The licensee provided updated chemistry data for the

beltline materials of Unit 1 by letters dated August 12, 1993 (D. L. Rehn to NRC, transmitting WCAP-13720) and July 3, 1997 (M. S. Tuckman to NRC). It should be noted that the staff and the licensee used the updated chemistry values to determine the ART for the beltline materials in Unit 1.

The beltline welds in the Unit 1 RPV were all fabricated using weld wire heat 895075. The staff's review of data in the RVID indicates that the initial RT_{NDT} for weld wire heat 895075, as reported by the licensee for Unit 1, may be nonconservative. The staff does not agree with the initial RT_{NDT} of -51 °F for weld wire heat 895075, because this initial RT_{NDT} value is nonconservative when compared to data from other plants for this weld wire heat. However, the ART for the subject weld using the most conservative value of the initial RT_{NDT}, as indicated in the RVID, for heat number 895075, is -40 °F, which still results in an ART that is less than the ART for the limiting material in the RPV. Therefore, using that value will have no impact on the P-T limits evaluation.

The staff performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Rev. 2. Based on these calculations, the staff verified that the licensee's limiting material for the Unit 1 reactor vessel is the lower shell forging plate 04 that was fabricated using plate heat 527708. The staff's calculated ART value for the limiting material agreed with the licensee's calculated ART value. Substituting the ART values for the Unit 1 limiting plate into the equations in SRP 5.3.2, the staff verified that the proposed P-T limits satisfy the requirements in Paragraph IV.A.2 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. Based on the flange RT_{NDT} of 10 °F for Unit 1 provided by the licensee, the staff has determined that the proposed P-T limits, as depicted in the proposed Figures 3.4-2 and 3.4-3, satisfy the requirement for the closure flange region during normal operation and hydrostatic pressure test and leak test.

Unit 2

For the Unit 2 reactor vessel, the licensee determined that the most limiting material at the 1/4T and 3/4T locations is the intermediate shell plate B8605-2. The licensee calculated an ART of 113 °F at the 1/4T location and 96 °F at the 3/4T location at 15 EFPYs. The neutron fluence used in the ART calculation was 7.0×10^{18} n/cm^2 at the 1/4T location and 2.5×10^{18} n/cm^2 at the 3/4T location. The initial RT_{NDT} for the limiting plate was 33 °F. The margin term used in calculating the ART for the limiting plate was 34 °F at the 1/4T location and 32.2 °F at the 3/4T location, as permitted by Position 1.1 of RG 1.99, Rev. 2.

The ART is determined using the chemistry values for each beltline material of Unit 2. The RVID contains chemistry values for each beltline material for all light water reactors in the U.S. The licensee provided updated chemistry data for the beltline materials of Unit 2 by letters

dated March 2, 1994 (D. L. Rehn to NRC, transmitting WCAP-13875) and July 3, 1997 (M. S. Tuckman to NRC). It should be noted that the staff and the licensee used the updated chemistry values to determine the ART for the beltline materials in Unit 2. In addition, the staff compared the licensee's best estimate chemistry data for weld wire heat 83648 against the best estimate chemistry values in the Combustion Engineering Owners Group (CEOG) Report CE NPSD-1039, Revision 2. The staff verified that the licensee's best estimate Cu and Ni values were the same as the values in the CEOG Report.

The beltline welds in the Unit 2 RPV were all fabricated using weld wire heat 83648. The staff reviewed the initial RT_{NDT} values in the RVID for welds made of weld wire heat 83648 for all plants. The staff found that the initial RT_{NDT} value of -80°F for the Unit 2 axial and girth welds was acceptable, since there were no other plants with the same weld wire heat.

The staff performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Rev. 2. Based on these calculations, the staff verified that the licensee's limiting material for the Unit 2 reactor vessel is the intermediate shell plate B8605-2. The staff's calculated ART value for the limiting material agreed with the licensee's calculated ART value. Substituting the ART values for the Unit 2 limiting plate into the equations in SRP 5.3.2, the staff verified that the proposed P-T limits satisfy the requirements in Paragraph IV.A.2 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. Based on the flange RT_{NDT} of 10 °F for Unit 2 provided by the licensee, the staff has determined that the proposed P-T limits satisfy the requirement for the closure flange region during normal operation and hydrostatic pressure test and leak test.

Proposed TS Changes

The staff concludes that the proposed P-T limits for the Catawba Nuclear Station, Units 1 and 2, reactor coolant systems for heatup, cooldown, leak test, and criticality satisfy the requirements in Appendix G to Section XI of the ASME Code and Appendix G of 10 CFR Part 50 for 15 EFPYs. The proposed P-T limits also satisfy GL 88-11 because the method in RG 1.99, Rev. 2, was used to calculate the ART. Therefore, the proposed P-T limits, depicted in revised TS Figures 3.4-2 and 3.4-3, may be incorporated into the TS.

The licensee also proposed to revise Table 4.4-5, "Reactor Vessel Material Surveillance Schedule," modifying the actual capsule number, lead factors and withdrawal schedule. For Unit 1, the proposed withdrawal schedule is in accordance with the ASTM E185-73 Standard. Appendix H of 10 CFR Part 50 states that "...the design of the surveillance program and the withdrawal schedule must meet the requirements of the edition of ASTM E185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased. Later editions of ASTM E 185 may be used, but including only those editions through 1982." The staff

concludes that the revised Catawba Unit 1 surveillance capsule withdrawal schedule satisfies the requirements of Appendix H to 10 CFR Part.50, and is acceptable.

For Unit 2, the proposed withdrawal schedule in the revised Table 4.4-5 meets the intent of the ASTM E185-82 Standard. Since this edition of the standard is a later edition of the standard that was current on the issue date of the ASME Code to which the reactor vessel was purchased and is included through the 1982 Edition, the Catawba revised surveillance capsule withdrawal schedule satisfies the requirements of Appendix H to10 CFR Part 50 and is acceptable.

2.3 Low-Temperature Overpressure (LTOP) Considerations

The LTOP system mitigates overpressure transients at low temperatures so that the integrity of the reactor coolant pressure boundary is not compromised by violating 10 CFR Part 50, Appendix G. Catawba's LTOP system uses the pressurizer power-operated relief valves (PORVs) to accomplish this function. The system is manually enabled by operators and uses a single setpoint as the lift pressure for the PORVs. The design basis of Catawba's LTOP system considers both mass-addition and heat-addition transients during water-solid reactor coolant system (RCS) conditions. Two mass-addition cases are analyzed. One analysis accounts for injection from a single safety injection pump while the other accounts for injection from a single centrifugal charging pump. The heat-addition analysis accounts for heat input from the secondary sides of all steam generators (SGs) into the RCS, upon starting a single reactor coolant pump (RCP). The heat-addition transient analysis assumes the SGs' secondary side temperatures are 50 °F higher than the RCS temperature.

<u>LTOP Enable Temperature</u>

The LTOP enable temperature is the temperature below which the LTOP system is required to be operable. The licensee's proposed enable temperature accounts for instrumentation uncertainties associated with the instrumentation used to enable the LTOP system and the temperature difference between the reactor coolant and the metal at a distance one-fourth of the vessel wall thickness from the inside surface in the beltline region. Therefore, the minimum allowed enable temperature is calculated as RT_{NDT} + 90 °F + E_{inst} + delta-$T_{1/4t}$; where, E_{inst} refers to instrument error and delta-$T_{1/4t}$ refers to the temperature difference between the reactor coolant and the metal at a distance one-fourth of the vessel wall thickness from the inside surface in the beltline region. This method is consistent with Branch Technical Position (BTP) RSB 5-2, which is attached to SRP Section 5.2.2, "Overpressure Protection," Revision 2.

The licensee proposed an LTOP enable temperature of ≥ 285 °F for both units. Based on vessel material data (RT_{NDT} values of 43 °F for Unit 1 and 112.6 °F for Unit 2), instrumentation uncertainties (the licensee conservatively assumed 10 °F) , and delta-$T_{1/4t}$ (the licensee used 19.1 °F based on a Westinghouse evaluation), the minimum allowed enable temperatures for Catawba Units 1 and 2 are 162.1 °F and 231.7 °F, respectively. Limiting RT_{NDT} determinations were reviewed by the staff (see above). However, for the values presented above, the proposed LTOP enable temperature of ≥ 285 °F is conservative with respect to the calculated minimum LTOP enable temperatures based on BTP RSB 5-2. Therefore, the staff finds the licensee's proposed enable temperature acceptable.

LTOP Actuation Setpoint

For the mass-addition cases for Catawba Units 1 and 2 and for the heat-addition case for Catawba Unit 2, the proposed LTOP actuation setpoint of ≤ 400 psig was verified to protect the P-T limits using the methods provided in Westinghouse reports, "Pressure Mitigating System Transient Analysis Results," July 1977, and "Supplement to the July 1977 Report, Pressure Mitigating System Transient Analysis Results," September 1977. These reports were utilized in determining the initial LTOP actuation setpoint. For the heat-addition case for Catawba Unit 1, the licensee performed plant-specific analyses using the RETRAN-02 MOD 5.1 Code to verify that the proposed setpoint for Unit 1 protects the P-T limits. This additional analysis was performed because Unit 1 has SGs with heat transfer areas that placed them outside of the limitations placed on the heat-addition analyses in the previously mentioned Westinghouse reports.

For the mass-addition case, the licensee performed two analyses. One with injection from a safety injection pump and one with injection from a centrifugal charging pump. For the heat-addition case, the licensee assumed the start of a single RCP with the temperature of the secondary side of the SGs 50 °F higher than the RCS. In its analyses, the licensee assumed that only one PORV was available for pressure relief. Additionally, the licensee did not credit the RHR suction relief valves for mitigating the pressure transient in the analyses. The licensee evaluated the results of the heat-addition and mass-addition cases with a water solid RCS and determined that the mass-addition cases remained limiting. The licensee considered the impact of SG tube plugging and accounted for static and dynamic head effects as well as instrumentation uncertainties in the final determination of the LTOP setpoints.

The licensee's analyses did not account for injection from the cold leg accumulators. Additionally, the dynamic head effect was divided into several temperature regions with different RCP operational restrictions. For Unit 1, the licensee used the following assumptions with regard to RCP operation:

RCS Cold Leg Temperature	Maximum Number of RCPs Allowed in Operation
≥67 °F	1
≥68 °F	2
≥73 °F	4

For Unit 2, the licensee used the following assumption:

RCS Cold Leg Temperature	Maximum Number of RCPs Allowed in Operation
≥67 °F	1
≥73 °F	2
≥95 °F	4

The licensee's analyses demonstrated that the proposed actuation setpoint of ≤ 400 psig is adequate to protect the P-T limits. The staff has reviewed the licensee's analyses and finds the licensee's proposed LTOP actuation setpoint acceptable. Additionally, the licensee proposed additional TS restrictions on the number of pumps capable of injection, accumulator discharge valves, and RCP operation. The staff has reviewed the proposed additional TS changes. The staff finds that the proposed changes are necessary to bring the TS into agreement with the assumptions in the analyses and, therefore, are acceptable.

With regard to the use of the RETRAN-02 MOD5.1 Code to perform the heat-addition transient for Catawba Unit 1, the staff has reviewed and accepted the use of this Code for non-LOCA transient analyses in letters dated April 12, 1994, November 1, 1991, October 19, 1988, and September 4, 1984. The RETRAN RCS model used for this analysis has also been approved by the NRC and is described in Duke Power Company Topical Report DPC-NE-3000-PA, Revision 1, dated December 1997. The licensee has reviewed the restrictions and limitations placed in the staff's safety evaluation reports associated with RETRAN-02 MOD5.1 computer Code and the RETRAN RCS model described in DPC-NE-3000-PA. The licensee concluded that the use of these methods for the heat-addition analysis was appropriate. The staff reviewed the associated reports and concurs with the licensee's assessment.

Minimum Temperature

The licensee evaluated the minimum temperature at which the LTOP system can protect the P-T limits. In this evaluation the licensee considered the mass-addition transient since it was more limiting and accounted for static and dynamic head effects as well as instrumentation uncertainties. Additionally, since the licensee's P-T curves did not extend below 85 °F, the licensee used extrapolated data for the lower temperatures. The extrapolations were conservatively calculated based on a constant slope of a straight line using the last two data points from the steady state curve. The licensee's evaluation determined that the minimum temperatures at which the LTOP system can protect the P-T curves are 63.2 °F for Unit 1 and 62.1 °F for Unit 2. Accordingly, the licensee proposed a note to TS 3.4.9.3 that limits the temperature to a minimum of 65 °F when the PORVs are relied upon for LTOP. Extrapolation of the P-T curves should be reviewed by the staff. However, for the values provided in the licensee's submittals, the staff finds the minimum temperature of 65 °F acceptable.

Summary of LTOP Review

The staff has reviewed the licensee's proposed LTOP system enable temperature and actuation setpoint. The staff has also reviewed the licensee's proposed TS related to the number of pumps capable of injection into the RCS, accumulator discharge valves, RCP operation, and the minimum temperature protected by the LTOP system. Based on the evaluation provided in Section 2 of this SE, the staff finds the licensee's proposed changes acceptable. The staff further finds acceptable the licensee's use the RETRAN-02 MOD 5.1 Code for analyzing the heat-addition scenario for Catawba Unit 1.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, South Carolina State official Virgil Autrey was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve 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 two proposed findings that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (62 FR 52580 dated October 8, 1997; 63 FR 40553 dated July 29, 1998). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Meena K. Khanna
Lambros Lois
Mohammed A. Shuaibi
Peter S. Tam

Date: August 28, 1998