May 11, 1992

Docket Nos. 50-269, 50-270
and 50-287

Distribution
See next page

Mr. J. W. Hampton
Vice President, Oconee Site
Duke Power Company
P. O. Box 1439
Seneca, South Carolina 29679

① See Correction ltr of 5/27/92
② See correction ltr of 6/5/92
③ See correction ltr of 7/13/93

Dear Mr. Hampton:

SUBJECT: ISSUANCE OF AMENDMENTS - OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3 (TACS M66390, M66391 AND M66392)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 195, 195 , and 191 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55, respectively, for the Oconee Nuclear Station, Units 1, 2, and 3. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated July 26, 1985, as supplemented August 14, 1987, August 12 and November 28, 1988, August 21, 1990, March 5, 1991, March 24 and April 9, 1992.

The amendments revise the TS to add Limiting Conditions for Operation, surveillance requirements and bases, and manpower requirements for the operation of the Standby Shutdown Facility.

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,
/S/
Leonard A. Wiens, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 195 to DPR-38
2. Amendment No. 195 to DPR-47
3. Amendment No. 191 to DPR-55
3. Safety Evaluation

cc w/enclosures:
See next page

OFC	:PDII-3/LA	:PDII-3/PM	:OGC	:PDII-3/D	:	:
NAME	:LBERRY	:LAWIENS	:RBachmann	:DMATTHEWS	:	:
DATE	:4/24/92	:4/24/92	:5/ /92	:5/11/92	:	:

OFFICIAL RECORD COPY
File Name: OC066390.AMD

DF01

140010

cc:
Mr. A. V. Carr, Esquire
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242-0001

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

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Division
Suite 525
1700 Rockville Pike
Rockville, Maryland 20852

Manager, LIS
NUS Corporation
2650 McCormick Drive, 3rd Floor
Clearwater, Florida 34619-1035

Senior Resident Inspector
U. S. Nuclear Regulatory Commission
Route 2, Box 610
Seneca, South Carolina 29678

Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, NW. Suite 2900
Atlanta, Georgia 30323

Mr. Heyward G. Shealy, Chief
Bureau of Radiological Health
South Carolina Department of Health
and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

County Supervisor of Oconee County
Walhalla, South Carolina 29621

Mr. M. E. Patrick
Compliance
Duke Power Company
Oconee Nuclear Site
P. O. Box 1439
Seneca, South Carolina 29679

Mr. Alan R. Herdt, Chief
Project Branch #3
U. S. Nuclear Regulatory Commission
101 Marietta Street, NW. Suite 2900
Atlanta, Georgia 30323

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

Mr. R. L. Gill, Jr.
Licensing
Duke Power Company
P. O. Box 1007
Charlotte, North Carolina 28201-1007

DATED: May 11, 1992

AMENDMENT NO. 195 OCONEE UNIT 1
AMENDMENT NO. 195 OCONEE UNIT 2
AMENDMENT NO. 191 OCONEE UNIT 3

DISTRIBUTION:

Docket File	
NRC & Local PDRs	
PD II-3 R/F	
Oconee R/F	
S. Varga	14-E-4
G. Lainas	14-H-3
D. Matthews	14-H-25
L. Berry	14-H-25
L. Wiens	14-H-25
OGC-WF	15-B-18
D. Hagan	MNBB 4702
G. Hill (12)	P1-22
W. Jones	MNBB 7103
C. Grimes	11-F-23
ACRS (10)	P-135
OPA	2-G-5
OC/LFMB	
L. A. Reyes	RII

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 195
License No. DPR-38

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Facility Operating License No. DPR-38 filed by the Duke Power Company (the licensee) dated July 26, 1985, as supplemented August 14, 1987, August 12 and November 28, 1988, August 21, 1990, March 5, 1991, March 24 and April 9, 1992, 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 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. **195**, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: **May 11, 1992**

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 195
License No. DPR-47

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Facility Operating License No. DPR-47 filed by the Duke Power Company (the licensee) dated July 26, 1985, as supplemented August 14, 1987, August 12 and November 28, 1988, August 21, 1990, March 5, 1991, March 24 and April 9, 1992, 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 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. **195**, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: **May 11, 1992**

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. **191**
License No. DPR-55

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Facility Operating License No. DPR-55 filed by the Duke Power Company (the licensee) dated July 26, 1985, as supplemented August 14, 1987, August 12 and November 28, 1988, August 21, 1990, March 5, 1991, March 24 and April 9, 1992, 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 3.B of Facility Operating License No. DPR-55 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. **191**, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: **May 11, 1992**

ATTACHMENT TO LICENSE AMENDMENT NO. 195

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 195

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 191

FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

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

Remove Pages	Insert Pages
iv	iv
v	v
vi	vi
-	3.18-1
-	3.18-2
-	3.18-3
-	3.18-4
-	3.18-5
-	3.18-6
-	4.20-1
-	4.20-2
-	4.20-3
-	4.20-4
-	4.20-5
-	4.20-6
6.1-6a	6.1-6a

Section		Page
3.10	GAS STORAGE TANK AND EXPLOSIVE GAS MIXTURE	3.10-1
3.11	(Not Used)	3.11-1
3.12	REACTOR BUILDING POLAR CRANE AND AUXILIARY HOIST	3.12-1
3.13	SECONDARY SYSTEM ACTIVITY	3.13-1
3.14	SNUBBERS	3.14-1
3.15	CONTROL ROOM PRESSURIZATION AND FILTERING SYSTEM AND PENETRATION ROOM VENTILATION SYSTEMS	3.15-1
3.16	HYDROGEN PURGE SYSTEM	3.16-1
3.17	FIRE PROTECTION AND DETECTION SYSTEMS	3.17-1
3.18	STANDBY SHUTDOWN FACILITY	3.18-1
4	SURVEILLANCE REQUIREMENTS	4.0-1
4.0	SURVEILLANCE STANDARDS	4.0-1
4.1	OPERATIONAL SAFETY REVIEW	4.1-1
4.2	STRUCTURAL INTEGRITY OF ASME CODE CLASS 1, 2 AND 3 COMPONENTS	4.2-1
4.3	TESTING FOLLOWING OPENING OF SYSTEM	4.3-1
4.4	REACTOR BUILDING	4.4-1
4.4.1	Containment Leakage Tests	4.4-1
4.4.2	Structural Integrity	4.4-14
4.4.3	Hydrogen Purge System	4.4-17
4.4.4	Reactor Building Purge System	4.4-20
4.5	EMERGENCY CORE COOLING SYSTEMS AND REACTOR BUILDING COOLING SYSTEMS PERIODIC TESTING	4.5-1
4.5.1	Emergency Core Cooling Systems	4.5-1
4.5.2	Reactor Building Cooling Systems	4.5-6
4.5.3	Penetration Room Ventilation System	4.5-10
4.5.4	Low Pressure Injection System Leakage	4.5-12
4.6	EMERGENCY POWER PERIODIC TESTING	4.6-1
4.7	REACTOR CONTROL ROD SYSTEM TESTS	4.7-1
4.7.1	Control Rod Trip Insertion Time	4.7-1
4.7.2	Control Rod Program Verification	4.7-2
4.8	MAIN STEAM STOP VALVES	4.8-1

Section		Page
4.9	EMERGENCY FEEDWATER PUMP AND VALVE PERIODIC TESTING	4.9-1
4.10	REACTIVITY ANOMALIES	4.10-1
4.11	(Not Used)	4.11-1
4.12	CONTROL ROOM PRESSURIZATION AND FILTERING SYSTEM	4.12-1
	(INTENTIONALLY BLANK)	4.13-1
4.14	REACTOR BUILDING PURGE FILTERS AND SPENT FUEL POOL VENTILATION SYSTEM	4.14-1
4.15	(Not Used)	4.15-1
4.16	RADIOACTIVE MATERIALS SOURCES	4.16-1
4.17	STEAM GENERATOR TUBING SURVEILLANCE	4.17-1
4.18	SNUBBERS	4.18-1
4.19	FIRE PROTECTION AND DETECTION SYSTEM	4.19-1
4.20	STANDBY SHUTDOWN FACILITY	4.20-1
4.21	(Not Used)	4.21-1
5	DESIGN FEATURES	5.1-1
5.1	SITE	5.1-1
5.2	CONTAINMENT	5.2-1
5.3	REACTOR	5.3-1
5.4	NEW AND SPENT FUEL STORAGE FACILITIES	5.4-1
6	ADMINISTRATIVE CONTROLS	6.1-1
6.1	ORGANIZATION, REVIEW, AND AUDIT	6.1-1
6.1.1	Organization	6.1-1
6.1.2	Technical Review and Control	6.1-2
6.1.3	Nuclear Safety Review Board	6.1-3a
6.2	ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE OCCURRENCE	6.2-1
6.3	ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED	6.3-1
6.4	STATION OPERATING PROCEDURES	6.4-1

LIST OF TABLES

Table No.		Page
2.3-1	Reactor Protective System Trip Setting Limits - Units 1,2 and 3	2.3-5
3.5.1-1	Instruments Operating Conditions	3.5-4
3.5-1	(Not Used)	3.5-14
3.5.5-1	(Not Used)	3.5-39
3.5.5-2	(Not Used)	3.5-41
3.5.6-1	Accident Monitoring Instrumentation	3.5-45
3.7-1	Operability Requirements for the Emergency Power Switching Logic Circuits	3.7-14
3.17-1	Fire Protection & Detection Systems	3.17-5
3.18-1	SSF Minimum Instrumentation	3.18-6
4.1-1	Instrument Surveillance Requirements	4.1-3
4.1-2	Minimum Equipment Test Frequency	4.1-9
4.1-3	Minimum Sampling Frequency and Analysis Program	4.1-10
4.1-4	(Not Used)	4.1-16
4.4-1	List of Penetrations with 10CFR50 Appendix J Test Requirements	4.4-6
4.11-1	(Not Used)	4.11-3
4.11-2	(Not Used)	4.11-5
4.11-3	(Not Used)	4.11-8
4.17-1	Steam Generator Tube Inspection	4.17-6
4.20-1	SSF Instrumentation Surveillance Requirements	4.20-5
6.1-1	Minimum Operating Shift Requirements with Fuel in Three Reactor Vessels	6.1-6

3.18 STANDBY SHUTDOWN FACILITY

Applicability

Applies to the Oconee Standby Shutdown Facility (SSF) consisting of the SSF Auxiliary Service Water (ASW), SSF Portable Pumping System, and SSF Reactor Coolant (RC) Makeup Systems, associated instrumentation, electrical generation and distribution, support systems, and the interfaces with normal in-plant systems.

Objective

To specify minimum conditions necessary to assure the operability of the Standby Shutdown Facility when any Oconee unit Reactor Coolant System temperature is at or above 250°F.

Specification

3.18.1 a. The Oconee SSF consisting of the SSF ASW, SSF Portable Pumping System, SSF RC Makeup Systems, associated instrumentation, electrical generation and distribution, support systems, and the interfaces with normal in-plant systems shall be operable at any time an Oconee Unit is at or above 250°F, except as permitted by Specifications 3.18.2, 3.18.3, 3.18.4, 3.18.5, 3.18.6, 3.18.7, and 3.18.8.

b. The Provisions of Specification 3.0 do not apply.

3.18.2 SSF Auxiliary Service Water System

a. The SSF ASW System, consisting of SSF ASW pump and a flow path capable of taking suction from the Unit 2 CCW line and discharging into the secondary side of each steam generator, shall be operable for each Unit at or above 250°F, except as permitted by part (b) or Specification 3.18.7.

b. If the SSF ASW system is inoperable, it shall be restored to operable status within 7 days, or the affected unit(s) shall be in hot shutdown within the next 12 hours, and below 250°F within the following 72 hours.

3.18.3 SSF Portable Pumping System

a. The SSF Portable Pumping System, consisting of SSF Submersible Pump and a flow path capable ofpumping water from the intake canal into the Unit 2 CCW line, shall be operable when any unit is at or above 250°F, except as permitted by part (b) or 3.18.7.

b. If the SSF Portable Pumping System is inoperable, it shall be restored to operable status within 7 days, or all unit(s) shall be in hot shutdown within the next 12 hours, and below 250°F within the next 72 hours.

3.18.4 SSF Reactor Coolant Makeup System

a. The SSF RC Makeup System, consisting of the SSF RC makeup pump, a flow path from the spent fuel pool and discharging into the Reactor Coolant System shall be operable for each unit at or above 250°F, except as permitted by part (b) or by Specification 3.18.7.

b. If the SSF RC Makeup is inoperable, it shall be restored to operable status within 7 days or the affected unit(s) shall be in hot shutdown within the next 12 hours, and below 250°F with the following 72 hours.

3.18.5 SSF Power System

a. The SSF Power System consisting of the SSF Diesel Generator(SSF DG), diesel support systems, 4160 VAC, 600 VAC, 208 VAC, 120 VAC, 125 VDC systems, shall be operable for any unit at or above 250°F, except as permitted by part (b) or by Specification 3.18.7.

(1) The SSF DG and support systems consists of the diesel generator, fuel oil transfer system, air start system, diesel engine service water system, as well as associated controls and instrumentation.

(2) The power system consists of 4160V switchgear OTS1; 600V load center OXSF; 600V motor control centers XSF, 1XSF, 2XSF, 3XSF; 208V motor control centers 1XSF, 1XSF-1, 2XSF, 2XSF-1, 3XSF, 3XSF-1; 120V panelboards KSF, KSFC.

(3) The DC power system consists of two batteries and associated chargers, 125VDC distribution centers DCSF, DCSF-1, and power panelboard DCSF. Only one battery and associated charger is required to be operable and connected to the 125VDC distribution center.

b. If the SSF Power System is inoperable, it shall be restored to operable status within 7 days or the affected unit(s) shall be in hot shutdown within the next 12 hours and below 250°F within the following 72 hours.

3.18.6 SSF Associated Instrumentation

a. The associated instrumentation for the SSF, consisting of the instrumentation specified in Table 3.18.1, shall be operable for each unit at or above 250°F, except as permitted by part (b) or by Specification 3.18.7.

b. With less than the minimum SSF instrumentation in Table 3.18.1 operable, it shall be restored to operable status within 7 days or the affected unit(s) shall be in hot shutdown within the next 12 hours, and below 250°F with the following 72 hours.

3.18.7 Special inoperability periods are allowed for maintenance on the SSF, with the following restrictions.

a. Special inoperability periods are independent of the degraded mode periods allowed by Specifications 3.18.2, 3.18.3, 3.18.4, 3.18.5, and 3.18.6.

b. The special inoperability periods shall total no more than 45 days per year.

3.18.8 While the SSF or any of its major subsystems is in a degraded mode or a special inoperability period allowed by Specifications 3.18.2, 3.18.3, 3.18.4, 3.18.5, 3.18.6, and/or 3.18.7, any Oconee unit may be heated above 250°F, permitted to remain critical, or restarted.

Bases

The SSF is designed to mitigate the consequences of postulated fire or flooding incidents, or acts of industrial sabotage to one or more of the three units at Oconee. The SSF contains, within seismically designed structures a reactor coolant volume control system for maintenance of primary system coolant during hot shutdown conditions; a steam generator volume control system for secondary system heat removal capabilities; independent emergency sources of AC and DC electrical power and associated electrical distribution systems; and various support systems. The SSF is designed to provide an alternate and independent means to achieve and maintain hot shutdown conditions for one or more of the three Oconee units. The SSF is in addition to and supplements the current shutdown capability described in the Oconee FSAR. It would be operated only in the event installed normal and emergency systems are inoperable. Manual operator action is required to actuate the systems.

The SSF Auxiliary Service Water System is a high head, high volume system designed to provide sufficient steam generator inventory for adequate decay heat removal for three units during a loss of normal AC power in conjunction with the loss of the normal and Emergency Feedwater Systems.

The SSF Portable Pumping System is designed to provide a backup supply of water to the SSF in the event of loss of CCW and subsequent loss of CCW siphon flow. The SSF Portable Pumping System is not safety grade and is installed manually according to Emergency procedures.

The SSF RC Makeup System is designed to supply makeup to the Reactor Coolant System (RCS) in the event that normal makeup systems are unavailable. The capacity of this system is sized to account for normal RCS leakage and shrinkage which results from going from a hot power operating condition to hot shutdown.

The SSF power supply is designed to provide normal and independent emergency sources of AC and DC electrical power, their associated electrical distribution systems and various support systems in the SSF. The SSF diesel generator would be operated only in the event installed normal power systems are inoperable. Manual operator action is required to actuate this system.

The SSF power supply includes 4160VAC, 600VAC, 208VAC, 120VAC and 125VDC power. This system supplies power necessary for the hot shutdown of the reactor in the event of loss of power from all other power systems. It consists of switchgear, a load center, motor control centers, panelboards, remote starters, batteries, battery chargers, inverters, a diesel powered electrical generator unit, relays, control devices, and interconnecting cable supplying the appropriate loads.

The 125VDC SSF Power System consists of two 125 VDC batteries and associated chargers, two DC distribution centers, and a DC power panelboard. This system is designed to provide an uninterruptible source of power for the SSF equipment controls and instrumentation.

Normally, one 125 VDC battery and its associated charger are connected to the 125VDC distribution center to supply the 125VDC loads. In this alignment, the battery is floated on the distribution center and is available to assure load without interruption upon loss of its associated battery charger or AC power source. The other 125VDC battery and its associated charger are in a standby mode and are not normally connected to the 125VDC distribution center. However, they are available via manual connection to the 125VDC distribution center to supply SSF loads, if required.

Although it is desirable to maintain the SSF operable to mitigate design basis events, short periods of inoperability are necessary for testing and maintenance to assure a high degree of reliability for the SSF. The 7 day limiting condition for operation (LCO) will be sufficient for routine testing and maintenance; however, inoperability periods of greater than 7 days are also allowed. To minimize the number and duration of extended outages associated with the special inoperability periods, outages of greater than 7 days on any unit shall not total more than 45 days per year without prior NRC approval.

<u>References</u>

1) NRC to Duke Power letter, April 28, 1983, "Safety Evaluation Report for the Oconee SSF".

2) FSAR, Section 9.6.1.

TABLE 3.18-1
SSF
MINIMUM INSTRUMENTATION

	Instrument	Readout Location	Minimum Channels Operable
1.	Reactor Coolant System Pressure	SSF Control Panel	1/Unit
2.	Reactor Coolant System Temperature (Tc)	SSF Control Panel	1/Loop/Unit
3.	Reactor Coolant System Temperature (Th)	SSF Control Panel	1/Loop/Unit
4.	Pressurizer Water Level	SSF Control Panel	1/Unit
5.	Steam Generator Level(Loop A&B)	SSF Control Panel	1/Steam Generator/ Unit
6.	D/G Air Start System Pressure	SSF D/G Room	1

4.20 STANDBY SHUTDOWN FACILITY

Applicability

Applies to the periodic surveillance testing requirements for the Standby Shutdown Facility (SSF) consisting of the SSF Auxiliary Service Water, SSF Portable Pumping System, SSF RC Makeup Systems, associated instrumentation, electrical generation and distribution, support systems, and the interfaces with normal in-plant systems.

Objective

To verify that the systems and components associated with the SSF are operable.

Specification

4.20.1 SSF Pumps and Valves

a. Inservice testing of SSF ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50, §50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components with the exception as permitted by Specification 4.20.1.c.

b. In the event that a pump or valve is determined to be inoperable by the performance of a surveillance test, then actions shall be taken for the affected system as required by Specification 3.18.

c. Inservice testing of the submersible pump for the SSF Portable Pumping System will be performed on a two (2) year frequency and will consist of testing developed head and flow.

4.20.2 SSF Instrumentation

a. The frequency and type of surveillance required for SSF instrumentation shall be as stated in Table 4.20-1.

b. In the event that an instrument is determined to be inoperable by the performance of a surveillance test, then actions shall be taken for the affected system as required by Specification 3.18.

4.20.3 SSF Electrical Power Systems

a. Diesel Generator

1. Monthly, or after maintenance or modification that could affect its operability the SSF diesel generator shall be verified operable by:

   a. Verifying the fuel inventory in the day tank is greater than or equal to 200 gallons and,

   b. Verifying the fuel inventory in the underground oil storage tank is greater than or equal to 25,000 gallons and,

   c. Verifying the diesel starts from standby condition and runs according to the procedures and requirements recommended by the manufacturer.

2. Quarterly verify that:

   a. The SSF diesel generator can be operated for a least 60 minutes with a load of greater than or equal to 3000 KW. This test may be preceded by an engine prelube period and/or other warm-up procedures recommended by the manufacturer.

   b. The fuel oil transfer pump starts and transfers fuel from the storage system to the day tank. This test will be performed per Specification 4.20.1.a.

3. Quarterly, diesel fuel from the day tank and the underground storage tank shall be sampled and analyzed for viscosity, water and sediment in accordance with applicable ASTM Specifications for Diesel Fuel Oil.

4. Annually, the SSF diesel generator shall be demonstrated operable by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

5. In the event the SSF diesel generator is determined to be inoperable by the performance of a surveillance test, then actions shall be taken as required by Specification 3.18.

b. DC Power System

Batteries in the SSF shall have the following periodic inspections performed to assure maximum battery life. Any battery or cell not in compliance with these periodic inspection requirements shall be corrected to meet the requirements within 90 days or the battery shall be declared inoperable.

1. Weekly, verify that:

   a. The electrolyte level of each pilot cell is in between the minimum and maximum level indication marks.

   b. The pilot cell specific gravity, corrected to 77 °F and full electrolyte level is ≥ 1.200.

   c. The pilot cell float voltage is ≥ 2.12 VDC.

   d. The overall battery float voltage is ≤ 125 VDC.

2. Quarterly, verify that:

   a. The specific gravity of each cell corrected to 77°F and full electrolyte level, is ≥ 1.200 and is not less than 0.010 below the average of all cells measured.

   b. The voltage of each cell under float charge is ≥ 2.12 VDC.

   c. The electrolyte level of each connected cell is between the minimum and maximum level indication marks.

3. Annually, verify that:

   a. The cells, end-cell plates and battery racks show no visual indication of structural damage or degradation.

b. The cell to cell and terminal connections are clean, tight and coated with anti-corrosion material.

4. Annually, a one hour discharge service test at the required maximum load shall be conducted.

5. In the event an SSF battery is declared to be inoperable by the performance of a surveillance test, then actions shall be taken as required by Specification 3.18.

TABLE 4.20-1
SSF INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

		Check	Calibrate	Remarks
1.	RCS Pressure (3)	WE	RF	Loop A, B
2.	SSF RC Makeup Pump (3)			
	Suction Pressure	QU(1)	RF	
	Discharge Pressure	QU(1)	RF	
	Suction Temperature	QU(1)	RF	
	Discharge Flow	QU(1)	RF	
3.	RC System Temperature(3)	NA(2)	RF	Loop A, B Hot, Cold
4.	Pressurizer Water Level(3)	WE	RF	
5.	SSF Auxiliary Service Water Pump			
	Suction Pressure	QU(1)	AN	
	Discharge Pressure	QU(1)	AN	
	Unit 1 Discharge Pressure	NA	AN	
	Unit 2 Discharge Pressure	NA	AN	
	Unit 3 Discharge Pressure	NA	AN	
	Discharge Test Flow	QU(1)	AN	
	Suction Temperature	QU(1)	AN	
6.	Steam Generator Levels (3)	WE	RF	A,B
7.	Underground Fuel Oil Storage Tank Inventory	NA	AN	
8.	D/G Service Water Pump			
	Discharge Flow	QU(1)	AN	
	Discharge Pressure	QU(1)	AN	
9.	D/G Air Start System Pressure	WE	AN	

(1) Check when pump operated/tested per IST.
(2) This instrumentation is normally aligned through a transfer/isolation device to each Unit Control Room and is thus checked in accordance with Specification 4.1, Table 4.1-1, Item 7. Each refueling outage, the instrument string to the SSF Control Room will be checked and calibrated.
(3) Units 1, 2, 3.

Bases

Surveillance requirements contained in this specification are provided to assure the SSF would be capable of performing its design function if demanded, and are consistent with the surveillance requirements for other equipment contained in Technical Specifications. All inservice testing of pumps and valves will be done in accordance with the provisions of ASME Section XI, Subsections IWP and IWV with the exception of the SSF Portable Pumping System Submersible Pump. This pump is not QA Grade and will be tested on a two-year frequency for developed head and flow, using calibrated test instrumentation.

The surveillance requirements for the SSF Instrumentation are based on experience in operation of both conventional and nuclear systems. The minimum checking frequency stated is deemed adequate for SSF Instrumentation. Calibration is performed to assure the presentation and acquisition of accurate information. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals specified.

The testing of the SSF electrical power systems are based upon a review of the surveillance requirements of other similar type of equipment contained within the technical specifications, manufacturer recommendations, and appropriate NRC guidelines. The quarterly verification of the diesel fuel oil from the day tank and the underground storage tank shall be sampled and verified to be within the acceptable limits specified in Table 1 of ASTM D 975-89, Standard Specification for Diesel Fuel Oils, when checked for viscosity, water and sediment. Inspection requirements of the batteries in the DC Power System for electrolyte level, specific gravity, and voltages are based upon manufacturer's recommendations.

ADDITIONAL REQUIREMENTS

1. One licensed operator per unit shall be in the Control Room at all times when there is fuel in the reactor vessel.

2. Two licensed operators shall be in the Control Room during startup and scheduled shutdown of a reactor.

3. At least one licensed operator shall in the reactor building when fuel handling operations in the reactor building are in progress.

4. An operator holding a Senior Reactor Operator license and assigned no other operational duties shall be in direct charge of refueling operations.

5. At least one person per shift shall have sufficient training to perform routine health physics requirements.

6. If the computer for a reactor is inoperable for more than eight hours, an operator, in additional to those required (1) and (2) above, shall supplement the Control Room shift crew.

7. A fire brigade of 5 members shall be maintained on site at all times. This excludes 3 member of the minimum operating shift requirements that are required to be present in the Control Rooms.

8. An operator holding a Senior Reactor Operator's license shall be in the Control Room from which the unit is operated whenever the unit is above cold shutdown.

9. Temporary deviations from the requirements of Table 6.6-1 may be allowed in cases of sudden illness, injury or other similar emergencies provided replacement personnel are notified immediately and are on site as soon as possible to return shift manning to minimum.

10. The qualified manpower necessary for achieving alternate shutdown using the SSF will be available at the plant at all times. The manpower necessary to operate the SSF will be exclusive of the fire brigade and the 3 member minimum operating shift that is required to be present in the Control Room.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 195 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 195 TO FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 191 TO FACILITY OPERATING LICENSE DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By letter dated April 28, 1983, the NRC staff transmitted a safety evaluation report (SER) to the Duke Power Company (the licensee) in which it concluded that the design of the Oconee Nuclear Station's Standby Shutdown Facility (SSF) was acceptable. However, the staff requested that the licensee provide Technical Specifications (TS) for the operability of the SSF components to ensure that the SSF will meet its intended safety function. By letter dated July 26, 1985, the licensee submitted proposed changes to the Oconee Nuclear Station, Units 1, 2, and 3 TS for the SSF.

In a letter of January 23, 1987, the NRC staff reviewed the submittal and concluded that the proposed SSF TS outage time limit of 60 days was unacceptable because the licensee relies on the SSF for mitigating design basis events such as fires, flooding, and sabotage. The staff suggested revising the limiting conditions for operation (LCO) of the proposed TS to make them equivalent to those of a safety-related system specified in the standard technical specifications (STS). In a letter of August 14, 1987, the licensee proposed revised SSF TS to address the staff's concerns on the 60-day outage time. In a submittal of August 12, 1988, the licensee added requirements for the hot leg temperature of the reactor coolant system (RCS) to the operability and surveillance requirements for instrumentation, as originally proposed in its submittals of July 26, 1985, and August 14, 1987. The NRC staff sent a request for additional information (RAI) dated September 20, 1988, after reviewing the licensee's August 14, 1987, submittal. The staff requested the following information:

1. The required mode of operation for the reactor during scheduled and unscheduled SSF outage times,

2. The impact that the failure of Keowee Hydroelectric Station and other emergency power sources will have on the capability of the SSF to perform its intended safety functions,

3. The rationale for deleting the originally proposed compensatory measures in the revised SSF TS proposal, and

4. The rationale for allowing an additional 7-day hot shutdown period before being required to enter Cold Shutdown mode, which is not provided in the Oconee TS for other similar safety systems.

Upon reviewing the licensee's response of November 28, 1988, to these concerns, the NRC staff issued a letter of February 21, 1990, in which it requested the licensee to clarify the following issues:

1. The allowed outage time of the SSF power system for planned reasons when the normal standby power supply (Keowee Hydroelectric Station) is out of service for planned or unscheduled inoperability and one or more of the units are in a condition requiring SSF operability,

2. Justification for including a 45-day combined special inoperability period for any SSF systems other than the SSF power system, and

3. The compensatory measures used between the 7th and 14th day of the proposed hot standby period when the SSF systems could still be required to be functional.

On August 21, 1990, the licensee submitted the requested information.

On September 17 and 18, 1990, the NRC staff met with the licensee to clarify previous concerns. Following the meeting, the licensee provided the staff with an amended TS submittal, dated March 5, 1991. The licensee also submitted SSF and diesel generator availability records in letters of August 1 and August 12, 1991.

The NRC staff reviewed these submittals, including the most recent submittals of March 5, 1991, and April 9, 1992, to verify that the SSF TS would meet the requirements for safe shutdown. This SER does not address acceptability of the TS for station blackout (SBO). As a separate activity, the staff will review the TS requirements for using the SSF for a station blackout (SBO) to verify that these requirements comply with the SBO rule and associated guidance. The August 14, 1987, August 12 and November 28, 1988, August 21, 1990, March 5, 1991, March 24 and April 9, 1992, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The SSF is a single train system consisting of a seismically designed, reinforced concrete structure and the components and systems contained therein. These components and systems provide an alternative and independent means to achieve and maintain hot shutdown in one, two, or three of the Oconee units for approximately 3 days following a loss of normal ac power. The SSF was designed to provide Oconee Nuclear Station with a means to meet the safe shutdown requirements for fire protection, turbine building flooding, and physical security. The licensee will operate the SSF only if installed normal and emergency systems become inoperable. The licensee personnel must manually actuate the SSF subsystems. The SSF is not designed to meet the single failure criterion but is designed to ensure that failures do not cause failures or inadvertent operation in existing plant systems. If one or more major subsystems and components become inoperable, they could cause the SSF to become inoperable.

The SSF includes the following major subsystems:

(1) The auxiliary service water (ASW) system. This high volume system consists of one high head ASW pump and is designed to provide sufficient coolant to the steam generator to ensure that each of the three units has adequate decay heat removal during loss of normal ac power and normal emergency feedwater system. The ASW system receives its supply of water from the Unit 2 buried condenser recirculating water (CCW) piping.

(2) The portable pumping system. This system is designed to provide a backup supply of water to the SSF ASW system if the CCW piping fails and causes a loss of CCW siphon flow from Unit 2.

(3) The reactor coolant (RC) makeup system. This system consists of three independent pumps. One pump is provided for each of the three units to supply coolant to the reactor coolant system (RCS) if normal makeup systems become unavailable. This system has sufficient capacity to compensate for normal RCS leakage and shrinkage which results from going from a hot power operation to hot shutdown. The RC makeup system obtains its coolant from the spent fuel pool, thus ensuring a supply of borated water.

(4) The electrical power system. This system provides normal independent emergency sources of ac and dc electrical power, their associated electrical distribution systems, and various support systems. The SSF standby power supply consists of an independent diesel generator unit rated at 3500 kw, 0.8 PF, 4160 volts. The dc power supply consists of two 125v dc batteries and associated chargers, two dc distribution centers, and a dc power panel board. This system is designed to provide an uninterruptible source of power for the SSF equipment controls and instruments.

(5) The SSF associated instrumentation. The SSF panel provides accurate and reliable information to ensure safe plant operation and shutdown conditions for all three units. Monitoring capability is provided for needed plant parameters to achieve and maintain hot shutdown conditions in all three units for approximately 3 days.

The licensee is proposing the following TS for Section 3.18, "Standby Shutdown Facility," to ensure that the operability of the SSF components is compatible with fire, flooding, and security assumptions used in the design.

3.18.1, Summary of the limiting conditions for operation requirements for SSF subsystems

3.18.2, "SSF Auxiliary Service Water System"

3.18.3, "SSF Portable Pumping System"

3.18.4, "SSF Reactor Coolant Makeup System"

3.18.5, "SSF Power System"

3.18.6, "SSF Associated Instrumentation"

3.18.7, LCO requirements for 45 day special inoperability periods

3.18.8, LCO requirements for allowing changing mode(s) of operation while in action statement

The licensee is proposing the following surveillance requirements (SRs) for Section 4.20, "Standby Shutdown Facility:"

4.20.1, "SSF Pumps and Valves"

4.20.2, "SSF Instrumentation"

4.20.3, "SSF Electrical Power Systems"

The LCO defined in TS 3.18.2 through 3.18.6 require that any inoperable system must be restored to operable status within 7 days or that one of two options must be followed: (1) the affected unit shall be in hot shutdown within the next 12 hours, and below 250°F within the following 72 hours, or (2) the 45-day special inoperability period as defined in TS 3.18.7 may be used to extend the outage time of the SSF inoperable subsystem.

The licensee may use the 45-day special inoperability period to extend the outage time of an SSF subsystem(s). Each Oconee unit has a 45-day special inoperability period. Certain subsystems, such as the RC makeup system, affect only one unit. If one of these systems is inoperable for 8 days, the licensee must reduce the 45-day special inoperability period by 8 days for that unit alone. Other subsystems, such as the diesel generator (DG) affect all units. If one of these systems is inoperable for 8 days, the licensee must reduce the 45-day special inoperability period for each unit by 8 days. If any unit uses its entire 45-day special inoperability period before the end of the year, the licensee will be required to shut down that unit if a subsystem is inoperable for more than 7 days. In the August 21, 1990, and November 28, 1988, submittals, the licensee described its method for calculating the 45-day combined special inoperability periods in TS 3.18.2 through 3.18.6.

In providing justification for the 45-day special inoperability period, the licensee stated that the need to remove the DG for preventive maintenance and draining of the Unit 2 CCW intake piping are necessary to ensure the long term reliability of the SSF. To perform major activities such as these, the licensee may need more than the 7-day outage time allowed in the LCO. Thus, the licensee would need the 45-day special inoperability provision to avoid shutdown of all three units unnecessarily. The licensee also stated that high radiation levels or problems with operating systems prevent it from performing repairs to certain SSF subsystems, such as the RC makeup pump or associated instrumentation during operation. Therefore, a special inoperability period allows some flexibility in scheduling shutdowns and other activities associated with such repairs. In the submittals of August 1 and 12, 1991, the licensee provided summaries of SSF availability records to support the 45-day special inoperability period. The licensee also stated that it anticipates approaching the end of the 45-day special inoperability period only when it drained the Unit 2 CCW intake piping, which is done once every 3 years. The staff reviewed the availability records and found that the Oconee SSF has only been unavailable for a period greater than 7 days twice since 1985. On August 8, 1985, the licensee had made the SSF unavailable for 7.5 days while making modifications to a diesel oil drain line. On May 22, 1989, the SSF became unavailable for 30.6 days while the licensee drained the Unit 2 CCW intake piping to inspect it and apply a coating. In the August 14, 1987, submittal, the licensee stated that it will perform the scheduled preventive maintenance on the SSF DG, SSF ASW pump, and SSF makeup pump when it drains the Unit 2 CCW piping, if practical, to decrease the outage of the SSF.

In the submittal of March 5, 1991, the licensee also proposed TS 3.18.8, which was previously proposed in the August 14, 1987, and August 21, 1990, submittals. Proposed TS 3.18.8 would allow the licensee to heatup a unit above 250°F, to maintain it critical, or to restart it, if the SSF or any of its major subsystems is in a degraded mode or a special inoperability period allowed by TS 3.18.2 through 3.18.6. The licensee had not provided justification for TS 3.18.8. Thus, in a submittal of April 4, 1992, the licensee justified the mode change LCO based on the following: (1) SSF events are of very low probability; (2) once the SSF is degraded, it usually affects all three units; (3) a unit heating up above 250°F places few additional requirements on the SSF beyond those already required for the operating units; and (4) the remote shutdown TS for many other plants contain similar provisions in the exceptions to LCO 3.0.4.

The NRC staff has reviewed the licensee's submittals requesting the 45-day special inoperability period under TS 3.18.7 and find the period acceptable based on the restrictive and justifiable needs for exceeding normal LCO outage limits provided in TS 3.18.2 through 3.18.6. The licensee has shown, through availability records, that the maximum unavailable period has been 30.6 days, which marginally exceeds the STS limit of 30 days. The licensee can use the special inoperability period of 45 days during certain events such as concurrent maintenance or maintenance of other systems during the same calendar year which require more than 7 days, thus reducing the possibility of unnecessary shutdown of one or more Oconee units. The LCO in TS 3.18.2 through 3.18.6 include an allowable inoperability period of 7 days for the SSF subsystems. This provision is compatible with the STS for emergency feedwater and other safety-related systems. The licensee's proposed LCO in TS 3.18.8 for allowing mode changes (to heat the reactor above 250°F, maintain it critical, or restart it while the SSF is degraded or during a special inoperability allowed by TS 3.18.2 through TS 3.18.6) is acceptable since this LCO is similar to those previously approved by the staff (such as for Catawba) and is consistent with LCOs for other systems which perform similar functions. Therefore, the staff finds that the proposed SSF TS are acceptable because (1) they generally conform with the STS guidelines, and (2) the licensee has prudently used the proposed 45-day special inoperability period under TS 3.18.7.

SR 4.20.1 defines the surveillance testing requirements for the SSF pumps and valves. SR 4.20.1 states that all inservice testing of ASME (American Society of Mechanical Engineers) Code Class 1, 2, and 3 pumps and valves in the SSF shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, and applicable addenda as required by Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50). All of the SSF pumps and valves will be tested in accordance with the above criteria with the exception of the portable pump system, which is not safety-related. This portable pump will be tested on a 2-year frequency for developed head and flow, using calibrated test instrumentation. SR 4.20.2 defines the frequency and type of surveillance required for the SSF instrumentation. Table 4.20.1 lists the frequency in which the SSF instrumentation is to be checked and calibrated.

In its rationale for the instrumentation surveillance requirements, the licensee stated that the surveillance requirements are based on the experience in operating both conventional and nuclear systems. Process system instrumentation errors induced by drift should remain within acceptable tolerances if the licensee calibrates these instruments at the specified intervals. SR 4.20.3 defines the surveillance tests to be performed monthly, quarterly, and after maintenance to verify the operability of the DG. The NRC staff finds that the surveillance requirements are acceptable because they are similar to the surveillance requirements of other similar types of equipment within technical specifications, manufacture recommendations, and appropriate NRC guidelines.

In 1983, the NRC staff found the SSF design acceptable to meet the safe shutdown requirements for fire protection, turbine building flooding, and physical security. The staff reviewed the proposed TS for these issues but did not consider station blackout (SBO).

The NRC staff concludes that the licensee's proposed TS 3.18, "Standby Shutdown Facility," is acceptable because it (1) generally meets the intent of the STS, (2) is similar to TS previously approved by the staff for other systems which perform similar functions, and (3) provides a 45-day special inoperability period that is restrictive and will be used with proper justification to ensure the long term reliability of the SSF.

The elements in SR 4.20, "Standby Shutdown Facility," are acceptable in that they are similar to the corresponding TS for similar plants, add conservatism, and are supported by additional surveillance procedures as appropriate.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements 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 a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (50 FR 43024). 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: J. Raval, SPLB
F. Burrows, SELB
S. Flanders, SPLB
L. Wiens, PDII-3

Date: **May 11, 1992**