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

MISSISSIPPI POWER & LIGHT COMPANY

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 35
License No. NPF-29

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

A. The application for amendment by Mississippi Power & Light Company, System Energy Resources, Inc. (formerly Middle South Energy, Inc.) and South Mississippi Electric Power Association, (the licensees) dated June 3, 1987, as supplemented June 22, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-29 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 35 , are hereby incorporated into this license. System Energy Resources, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 10, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 35

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf page(s) have been provided to maintain document completeness.

Remove	Insert
1-2	1-2
3/4 1-1	3/4 1-1
3/4 1-7	3/4 1-7
3/4 3-4	3/4 3-4
3/4 7-11	3/4 7-11
3/4 9-3	3/4 9-3
3/4 9-7	3/4 9-7
B3/4 7-3	B3/4 7-3

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

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

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of the SRMs, IRMs, LPRMs, TIPs, or special movable detectors is not considered to be CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CRITICAL POWER RATIO

1.8 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the XN-3 correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

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

DRYWELL INTEGRITY

1.10 DRYWELL INTEGRITY shall exist when:

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

    1. Capable of being closed by an OPERABLE drywell automatic isolation system, or

    2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.4-1 of Specification 3.6.4.

b. The drywell equipment hatch is closed and sealed.

c. The drywell airlock is in compliance with the requirements of Specification 3.6.2.3.

d. The drywell leakage rates are within the limits of Specification 3.6.2.2.

e. The suppression pool is in compliance with the requirements of Specification 3.6.3.1.

f. The sealing mechanism associated with each drywell penetration; e.g., welds, bellows or O-rings, is OPERABLE.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. 0.38% delta k/k with the highest worth rod analytically determined, or
b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle:

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity difference between the monitored core k_{eff} and the predicted core k_{eff} shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity difference greater than 1% delta k/k:

a. Within 12 hours, perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity difference between the monitored core k_{eff} and the predicted core k_{eff} shall be verified to be less than or equal to 1% delta k/k:

a. During the first startup following CORE ALTERATIONS, and

b. At least once per 1000 MWD/T during POWER OPERATION.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. With a "slow" control rod(s) not satisfying ACTION a.1, above:

   1. Declare the "slow" control rod(s) inoperable, and

   2. Perform the Surveillance Requirements of Specification 4.1 3.2.c at least once per 60 days when operation is continued with three or more "slow" control rods declared inoperable.

   Otherwise, be in at least HOT SHUTDOWN within 12 hours.

c. With the maximum scram insertion time of one or more control rods exceeding the maximum scram insertion time limits of Specification 3.1.3.2 as determined by Specification 4.1.3.2.c, operation may continue provided that:

   1. "Slow" control rods, i.e., those which exceed the limits of Specification 3.1.3.2, do not make up more than 20% of the 10% sample of control rods tested.

   2. Each of these "slow" control rods satisfies the limits of ACTION a.1.

   3. The eight adjacent control rods surrounding each "slow" control rod are:

      a) Demonstrated through measurement within 12 hours to satisfy the maximum scram insertion time limits of Specification 3.1.3.2, and

      b) OPERABLE.

   4. The total number of "slow" control rods, as determined by Specification 4.1.3.2.c, when added to the sum of ACTION a.3, as determined by Specification 4.1.3.2.a and b, does not exceed 7.

   Otherwise, be in at least HOT SHUTDOWN within 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

   a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS* or after a reactor shutdown that is greater than 120 days,

   b. For specifically affected individual control rods** following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

   c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

*Except normal control rod movement.

**The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 provided this surveillance is completed prior to entry into OPERATIONAL CONDITION 1.

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.3 All control rod scram accumulators shall be OPERABLE:

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITIONS 1 and 2:

1. With one control rod scram accumulator inoperable, within 8 hours:

a) Restore the inoperable accumulator to OPERABLE status, or

b) Declare the control rod associated with the inoperable accumulator inoperable.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

2. With more than one control rod scram accumulator inoperable, declare the associated control rods inoperable and:

a) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one control rod drive pump is operating by inserting at least one withdrawn control rod at least one notch or place the reactor mode switch in the Shutdown position.

b) Insert the inoperable control rods and disarm the associated directional control valves either:

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. In OPERATIONAL CONDITION 5*:

1. With one withdrawn control rod with its associated scram accumulator inoperable, insert the affected control rod and disarm the associated directional control valves within one hour, either:

a) Electrically, or

b) Hydraulically by closing the drive water and exhaust water isolation valves.

2. With more than one withdrawn control rod with the associated scram accumulator inoperable or with no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

c. The provisions of Specification 3.0.4 are not applicable.

*At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
9. Scram Discharge Volume Water Level - High			
a. Transmitter/Trip Unit	1, 2,	2	1
	$5^{(g)}$	2	3
b. Float Switch	1, 2,	2	1
	$5^{(g)}$	2	3
10. Turbine Stop Valve - Closure	$1^{(h)}$	4	6
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	$1^{(h)}$	2	6
12. Reactor Mode Switch Shutdown Position	1, 2	2	1
	3, 4	2	7
	5	2	3
13. Manual Scram	1, 2	2	1
	3, 4	2	8
	5	2	9

Amendment No. 21
Effective Date: OCT 2 0 1986

TABLE 3.3.1.1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the SHUTDOWN position within one hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS*, and insert all insertable control rods within one hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to less than the automatic bypass setpoint within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within one hour.

ACTION 8 - Lock the reactor mode switch in the SHUTDOWN position within one hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS*, and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within one hour.

*Also suspend replacement of LPRM strings unless SRM instrumentation is OPERABLE per Specification 3.9.2.

SURVEILLANCE REQUIREMENTS (Continued)

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans for each type of snubber. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.4.f, an additional 5% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7.4-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.4.f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7.4-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, $1 + C/2$, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested. The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor* (SRM) channels shall be OPERABLE and inserted to the normal operating level with:

a. Continuous visual indication in the control room,

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

c. Unless adequate shutdown margin has been demonstrated, the shorting links shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn.#

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS and insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours:

1. Performance of a CHANNEL CHECK,
2. Verifying the detectors are inserted to the normal operating level, and
3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant.

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*The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

#Not required for control rods removed per Specification 3.9.10.1 and 3.9.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. Performance of a CHANNEL FUNCTIONAL TEST:

   1. Within 24 hours prior to the start of CORE ALTERATIONS, and

   2. At least once per 7 days.

c. Verifying that the channel count rate is at least 0.7 cps*:

   1. Prior to control rod withdrawal,

   2. Prior to and at least once per 12 hours during CORE ALTERATIONS, and

   3. At least once per 24 hours,

   except that:

   1. During spiral unloading, the required count rate may be permitted to be less than 0.7 cps*.

   2. Prior to and during spiral loading, until sufficient fuel has been loaded to maintain at least 0.7 cps*, the required count rate may be achieved by:

      a) Use of portable external source, or

      b) Loading up to 2 fuel assemblies### in cells containing inserted control rods around an SRM.

d. Verifying that the RPS circuitry "shorting links" have been removed within 8 hours prior to and at least once per 12 hours during:

   1. The time any control rod is withdrawn,## or

   2. Shutdown margin demonstrations.

*Provided signal to noise ratio ≥ 2; otherwise use 3 cps.

##Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

###These fuel assemblies may be loaded with the SRM count rate less than 0.7 cps.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling platform personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.*

ACTION:

When direct communication between the control room and refueling platform personnel cannot be maintained, immediately suspend CORE ALTERATIONS.*

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling platform personnel shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.*

*Except movement of control rods with their normal drive system.

3/4.9.6 REFUELING EQUIPMENT

REFUELING PLATFORM

LIMITING CONDITION FOR OPERATION

3.9.6.1 The refueling platform shall be OPERABLE and only the main hoist shall be used for handling fuel assemblies.

APPLICABILITY: During handling of fuel assemblies or control rods in the primary containment with the refueling platform.

ACTION:

With the requirements for refueling platform OPERABILITY not satisfied, suspend use of any inoperable refueling platform equipment from operations involving the handling of fuel assemblies or control rods after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each refueling platform hoist to be used for handling fuel assemblies or control rods shall be demonstrated OPERABLE within 7 days prior to the handling of fuel assemblies or control rods:

a. In the containment fuel pool, reactor cavity or reactor pressure vessel by:

   1. Demonstrating operation of the slack cable cutoff on the main hoist when the total cable load is 50±10 pounds.

   2. Demonstrating operation of the grapple engaged loaded interlock on the main hoist before the total cable load exceeds 535 pounds.

   3. Demonstrating operation of the jam cutoff on the main hoist before the total cable load exceeds 1250 pounds.

   4. Demonstrating operation of primary and redundant overload cutoff on the auxiliary hoists before the load exceeds 550 pounds.

b. In or over the reactor pressure vessel by:

   1. Demonstrating operation of the downtravel cutoff on the main hoist when the bottom of the grapple is 3.5 ± 0.5 inches below the top of the fuel assembly handles in the reactor core.

   2. Demonstrating operation of the primary and redundant fuel load interlocks on the main hoist before the total cable load exceeds 600 pounds.

3/4.7.4 SNUBBERS (Continued)

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing.

To provide assurance of snubber functional reliability one of three functional testing methods is used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 5% tested for each functional testing failure, or
2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7.4-1, or
3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7.4-1 was developed using "Wald's Sequential Probability Ratio Plan" described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.5 SEALED SOURCE CONTAMINATION

The limitation on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism, i.e., sealed sources within radiation monitoring or boron measuring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.6 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, CO_2 systems, halon systems and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurances that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of halon in the halon storage tanks by verifying the weight and pressure of the tanks.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

The surveillance requirements for spray and sprinkler systems provide for periodic visual inspections to ensure that temporary structures/objects do not impair the spray patterns which have been established in accordance with the GGNS fire protection design requirements.

3/4.7.7 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4.7.8 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause loss of its OPERABILITY. The temperature limits include allowance for instrument error.

L. Kintner

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

September 10, 1987

Docket No.: 50-416

Mr. Oliver D. Kingsley, Jr.
Vice President, Nuclear Operations
System Energy Resources, Inc.
Post Office Box 23054
Jackson, Mississippi 39205

Dear Mr. Kingsley:

SUBJECT: CHANGES TO TECHNICAL SPECIFICATIONS REGARDING CORE ALTERATIONS AND SNUBBER SAMPLE SIZE (TAC NO. 65511)

RE: GRAND GULF NUCLEAR STATION, UNIT 1

The Commission has issued the enclosed Amendment No. 35 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated June 3, 1987, as supplemented June 22, 1987.

This amendment changes the definition of core alteration in the TSs to include certain exceptions and changes footnotes in the TSs to be consistent with the new definition. This amendment also changes a snubber surveillance test sample plan in the TSs by decreasing the number of additional snubbers required to be tested from 10% to 5% for each snubber in the initial test sample that fails to meet specified functional test criteria.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

L L Kintner

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Enclosures:
1. Amendment No. 35 to NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 35 TO FACILITY OPERATING LICENSE NO. NPF-29

MISSISSIPPI POWER & LIGHT COMPANY

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

INTRODUCTION

By letter dated June 3, 1987, as supplemented June 22, 1987, System Energy Resources, Inc., (the licensee) requested an amendment to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1 (GGNS-1). The proposed amendment would (1) change the definition of core alteration in the Technical Specifications (TSs) to include certain exceptions and change footnotes in the TSs to be consistent with the new definition; and (2) change a snubber surveillance test sample plan in the TSs by decreasing from 10% to 5% the number of additional snubbers required to be tested for each snubber in the initial test sample that fails to meet specified functional test criteria.

EVALUATION

(1) Definition of Core Alteration

The following changes to the TSs would be made:

a. The definition of core alteration would be modified to exclude normal movement of the source range monitors (SRMs), intermediate range monitors (IRMs), local power monitors (LPRMs), traversing in-core probes (TIPs) or special movable detectors.

b. The "*" footnote to Specification 3.1.1 on shutdown margin would be deleted. This footnote provides an exception to the core alteration definition for movement of IRMs, SRMs or special movable detectors.

c. The "*" footnote to Surveillance Requirement 4.1.3.2.a would be modified by deleting the exception to the core alteration definition for the movement of SRMs, IRMs or special movable detectors. The exception for normal control rod movement remains and is not affected by this proposed change.

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d. The "*" footnote to Table 3.3.1-1 would be modified by deleting the exceptions to the core alteration definition for IRMs, SRMs or special movable detectors. The part of the "*" footnote requiring operable SRM instrumentation for replacement of LPRM strings would be retained.

e. The "**" footnote to Specification 3.9.2 on refueling operations instrumentation would be deleted. This footnote provides an exception to the core alteration definition for movement of IRMs, SRMs, or special movable detectors.

f. The "*" footnote to Specification 3.9.5 would be modified by deleting the exception to the core alteration definition for incore instrumentation. The part of the "*" footnote that allows an exception for control rod movement with their normal drive system remains and is not affected by this proposed change.

The present definition of core alteration is:

"Core alteration shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of core alterations shall not preclude completion of the movement of a component to a safe conservative position."

The proposed change would insert the following after the first sentence:

"Normal movement of the SRMs, IRMs, LPRMs, TIPs, or special movable detectors is not considered a core alteration."

The exception to the present definition of core alteration for the normal movement of the SRMs, IRMs, LPRMs, TIPs, and special movable detectors is needed in certain specifications related to refueling operations in order to preclude unnecessary suspension of the normal movement of these detectors. During a refueling outage, maintenance or modification of equipment can result in TS limiting conditions for operation which require that core alterations be suspended. In the present TSs, exceptions to the definition of core alteration for normal movement of detectors are provided by footnotes in those TSs where a need for the exception was foreseen.

However, some TSs that require suspension of core alterations do not presently have a footnote excepting normal movement of detectors. For example, Specification 3.8.1.2 requires suspension of core alterations with diesel generator 11 or 12 inoperable. With the present TSs, surveillance tests of SRMS and IRMs could not be performed because the tests require movement of the detectors. Making the exception a part of the definition will correct this type of operational problem. Where particular conditions are required for normal movement of detectors, these conditions are retained in the applicable TSs. For example, the requirement for SRMs to be operable when replacing LPRMs is retained in Specification 3/4.3.1, "Reactor Protection System Instrumentation."

The NRC staff has reviewed the proposed changes to the GGNS-1 TSs related to core alterations. The detectors in the SRM, IRM, LPRM, TIP and the special movable detectors are sealed unit fission detectors and their reactivity worth is insignificant with respect to reactivity excursion events. Therefore, allowing the normal movement of these detectors will not significantly increase the probability or consequences of an accident previously analyzed in the Final Safety Analysis Report. The proposed change would only permit normal movement of the incore detectors. Normal movement of these detectors includes insertion and withdrawal using detector drives, replacement of detectors, and movement of special movable detectors in the core region. The addition, removal or relocation of SRMs, IRMs, LPRMs and TIPs would still be prohibited.

The staff concludes that the proposed changes to the definition of core alteration and the deletion of footnotes in the TSs would not significantly reduce the level of safety and would tend to enhance safety by making the TSs more readable. Accordingly, the proposed changes are acceptable.

(2) Snubber Sample Plan

To verify the operability of safety-related snubbers, Surveillance Requirement 4.7.4.e in the TSs requires functional testing to be performed on a periodic basis. The TSs permit the use of any one of three specified sampling plans. Essentially, all three plans require the testing of an initial sample of snubbers from the total population. For every inoperable snubber identified during testing of an initial sample of snubbers, an additional or subsequential sample is required to be tested. For Sample Plan 1, the size of the initial and the subsequential samples is 10% and 10%, respectively. The initial sample size of 10% for Sample Plan 1 was selected on the basis that every snubber in the plant will be tested at least once every 15 years when the associated functional testing period is 18 months. The subsequential sample size of 10% was selected as a conservative value.

For Sample Plans 2 and 3, initial and subsequential sample sizes are both determined by statistical considerations, and the subsequential samples are half that of the initial samples. All three sample plans should yield the same results. Yet for a population that would produce the same initial sample size for Sample Plans 1 and 2 or 1 and 3, the subsequential sample sizes will differ by twice as much. To make all three plans have an equal basis, the conservatively determined subsequential size of 10% for Sample Plan 1 should be reduced to 5%.

The American Society of Mechanical Engineers Operation and Maintenance Working Group 4 Standard (O&M 4 Standard), "Examination and Performance Testing of Nuclear Power Plant Dynamic Restraints (Snubbers)," has taken this into consideration and changed the recommended subsequential sample size from 10% to 5% for Sample Plan 1. The standard was approved by the NRC staff and will be adopted by ASME Boiler & Pressure Vessel Code Section XI for plant surveillance guidance.

In conclusion, the proposed change to Sample Plan 1 would make it consistent with the other two sample plans in the TSs, is in accordance with the requirements recommended by the O&M 4 Standard, and is therefore acceptable.

ENVIRONMENTAL CONSIDERATION

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

CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and the security nor to the health and safety of the public.

Dated: September 10, 1987

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