Docket File
50-348

FEBRUARY 13 1979

Docket No. 50-348

Mr. Alan R. Barton
Senior Vice President
Alabama Power Company
P. O. Box 2641
Birmingham, Alabama 35291

Gentlemen:

The Commission has issued the enclosed Amendment No. 8 to Facility Operating License No. NPF-2 for the Farley Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your applications dated November 4 and December 14, 1977 and August 9, 1978.

The amendment includes the following revisions to the Appendix A Technical Specifications:

1. Modification to Specification Figure 3.2-3 and Bases 2.1.1 relating to the fuel rod bow penalty for 17x17 fuel designed by Westinghouse.

2. Modifications to Specification 4.5.2 to increase the surveillance on ECCS subsystems (high pressure and low pressure safety injection) in response to our request dated August 30, 1977.

3. Administrative changes to Specifications 5.6.1 and 5.6.3 relating to storage of the spent fuel assemblies.

4. Modifications to Specification 4.3.3.2 and Bases 4.3.3.2 to authorize use of quarter-core flux maps for excore neutron flux detection system calibration.

5. Deletion of Specifications 3.7.1.6, and 4.7.1.6, Tables 3.7-3, 3/4.7-3 and Bases 3/4.7.1.6 and adding specification 6.10.2.m and license condition 2.C.(3)(g), all relating to secondary water chemistry.

6. Revision of Specification 3.4.9.2 to change the pressurizer heatup rate in response to our letter dated November 23, 1977.

CP 1

GN

Some modifications to your proposals were necessary to meet our requirements. These modifications were discussed with and agreed to by your staff.

7908140325

OFFICE➤						
SURNAME➤						
DATE➤						

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original Signed By

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 8 to NPF-2
2. Safety Evaluation
3. Notice of Issuance

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TACS# 08057C, 08165C & 07255C

see note to Reeves dtd 2/1/79

OFFICE▸	DOR:ORB1	DOR:ORB1	STS	~~XXXXXXX~~	OELD	DOR:ORB1
SURNAME▸	EAReeves:jb	CSParrish	D Brinkman		D Swanson	ASchwencer
DATE▸	12/6/78 & 1/25/79	12/4/78	1/25/79		2/5/79	2/1?/78

#1 7AI-16

February 1, 1979

NOTE TO: Ed Reeves
FROM: Dan Swanson DS
RE.: Farley unit 1 proposed tech spec changes

Done on pg 1

I have two minor comments on the safety evaluation prepared for the Farley Unit 2 proposed tech spec changes regarding fuel rod bow penalty, etc. On page 2 of the evaluation you start using several acronyms such as FNP, DNB, and DNBR which are undefined and, with the exception of FNP, are not at all obvious to the lay reader of the document. These should be defined.

Also, on page 6, 2d line from the bottom, there is another typo: an extra "t" in chemistry.

Judy - Plse to SE pg 2 & correct 2 typos pg 6 -
add * after DNB and * footnote Departure from Nucleate Boiling

Add ** after DNBR and ** footnote
** Departure from Nucleate Boiling Ratio

spell out

cc: Ruble A. Thomas, Vice President
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State Department of Public Health
ATTN: State Health Officer
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Montgomery, Alabama 36104

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U.S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, NE
Atlanta, Georgia 30308

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

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 8
License No. NPF-2

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

   A. The applications for amendment by the Alabama Power Company (the licensee) dated November 4 and December 14, 1977 and August 9, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commissions rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment

and paragraph 2.C(2) of Facility Operating License No. NPF-2 is hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 8 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license is further amended by adding the following new paragraph 2.C.(3)(g):

2.C.(3)(g) Alabama Power Company shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

2. Identification of the procedures used to quantify parameters that are critical to control points;

3. Identification of process sampling points;

4. Procedure for the recording and management of data;

5. Procedures defining corrective actions for off control point chemistry conditions; and

6. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events required to initiate corrective action.

4. The license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 13, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 8

FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

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 area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

B 2-2
3/4 2-10a
3/4 3-39
3/4 4-29
3/4 5-5
3/4 5-5a (added)
3/4 7-11
3/4 7-12
3/4 7-13
B 3/4 3-2
B 3/4 3-3
B 3/4 3-4
B 3/4 5-2
B 3/4 5-3 (added)
B 3/4 7-3
5-5
6-19

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

BASES

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} \leq 1.55\ [1 + 0.2\ (1-P)]\ [1-RBP(BU)]$$

where: $RBP(BU) = .01\ (-1.1667 + 0.05833\ \sqrt{BU})$ for $BU > 400\ \frac{MWD}{MTU}$

and $RBP\ (BU) = 0$ for $BU \leq 400\ \frac{MWD}{MTU}$

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the $f_1\ (\Delta I)$ function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel pressurizer and the reactor coolant system piping and fittings are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure.

The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

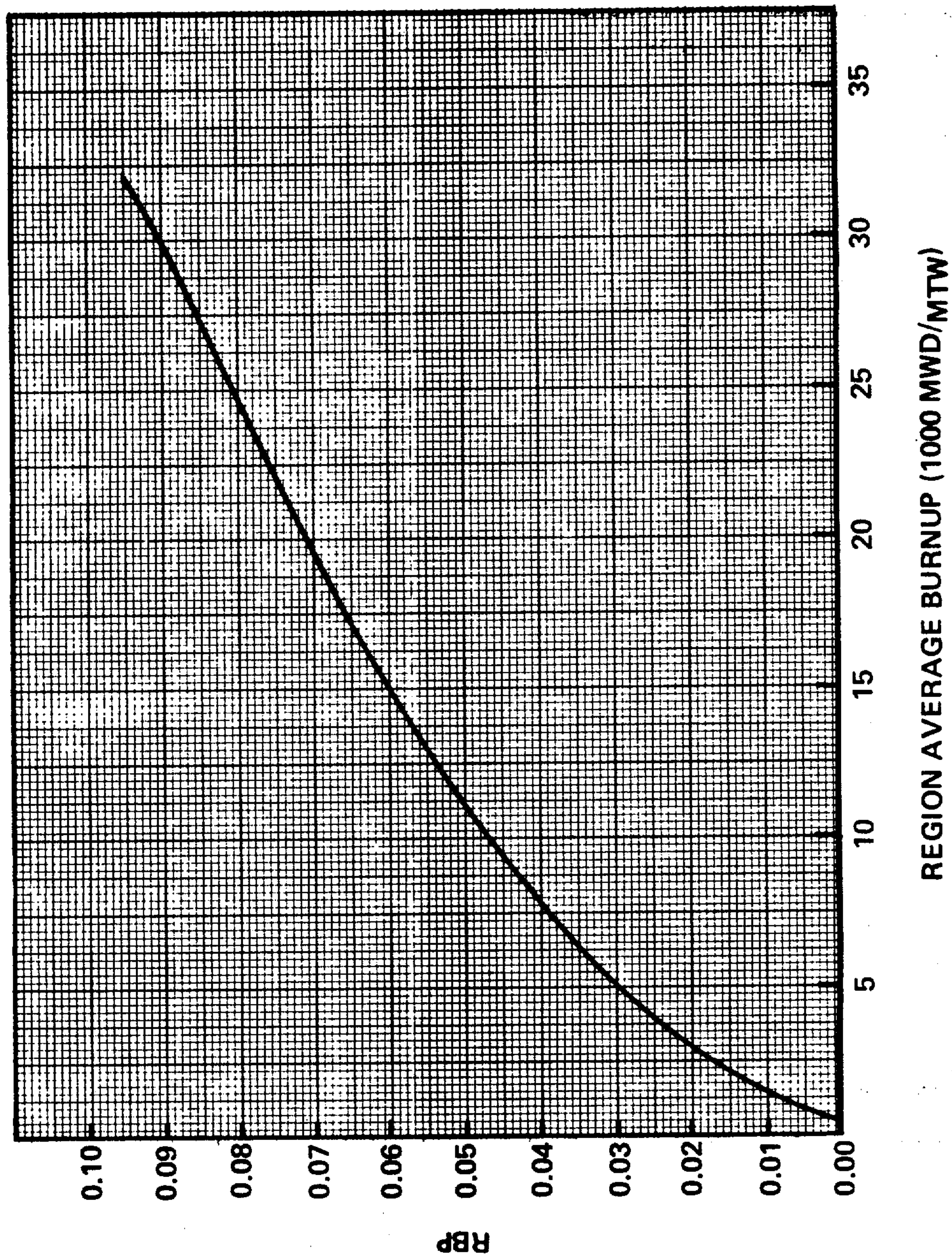

Figure 3.2-3 Rod Bow Penalty (RBP)

versus

Region Average Burnup Average

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 80% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY:

When the movable incore detection system is used for:

a. Recalibration of the excore neutron flux detection system,

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}

ACTION:

With the movable incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The movable incore detection system shall be demonstrated OPERABLE by normalizing each detector output when required for:

a. Recalibration of the excore neutron flux detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}.

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum cooldown of 200°F in any one hour period,

b. A maximum heatup of 100°F in any one hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per hour during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected components to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be demonstrated;

SURVEILLANCE REQUIREMENTS (Continued)

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosions.

e. At least once per 18 months, during reactor shutdown and within 4 hours following completion of maintenance on or stroking operation of the following manual valves, verify that these valves are in the proper position for injection.

Valve Number

CVC-V-8991 A/B/C
CVC-V-8989 A/B/C
CVC-V-8996 A/B/C
CVC-V-8994 A/B/C

f. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

a) Centrifugal charging pump

b) Residual heat removal pump

g. By verifying that each of the following pumps develops a discharge pressure (after subtracting suction pressure) on recirculation flow when tested pursuant to specification 4.0.5:

1. Centrifugal charging pump $\geq$ 2458 psig.

2. Residual heat removal pump $\geq$ 136 psig.

h. Prior to entry into Mode 3 from Mode 4, verify that the mechanical stops on low lead safety injection valves RHR-HV 603 A/B are intact.

SURVEILLANCE REQUIREMENTS (Continued)

i. A flow balance test shall be conducted during shutdown to confirm the following minimum injection flow rates following completion of HPSI or LPSI system modifications that alter system flow characteristics:

HPSI System - Single Pump	LPSI System - Single Pump
$\geq$ 193 gpm (each injection leg)	$\geq$ 3981 gpm (total injection)

ECCS SUBSYSTEMS - T_{avg} < 350°F

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

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

APPLICABILITY: MODE 4.

ACTION:

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

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

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

PLANT SYSTEMS

SECONDARY WATER CHEMISTRY

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3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of the primary coolant and feedwater shall be > 70°F when the pressure of either fluid in the steam generator is > 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to $\leq$ 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generator shall be determined to be < 200 psig at least once per hour when the temperature of the primary coolant or feedwater is < 70°F.

BASES

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served

RADIATION MONITORING INSTRUMENTATION (Continued)

by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

Alarm/trip setpoints for the containment purge have been established for a purge rate of 5,000 scfm in all modes and for purge rates of 25,000 scfm and 50,000 scfm in modes 4,5 and 6. The containment purge setpoints are based on a release in which Xe-133 and Kr-85 are the predominent isotopes, on the 10 CFR 20, Appendix B, Table 2, MPC values for these isotopes and on a X/Q of 5.6×10^{-6} sec/m^3 at the site boundry.

The Alarm/trip setpoint for the fuel storage pool area has been established based on a flow rate of 13,000 scfm; a release in which Xe-133 and Kr-85 are the predominent isotopes, on the 10 CFR 20, Appendix B, Table 2, MPC values for these isotopes and on a X/Q of 5.6×10^{-6} sec/m^3 at the site boundry.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is generally consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Program," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the chlorine detection system ensures that sufficient capibility is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975.

3/4.3.3.7 HIGH ENERGY LINE BREAK ISOLATION SENSORS

The high energy line break isolation sensors are designed to mitigate the consequences of the discharge of steam and/or water to the affected room and other lines and systems contained therein. In addition, the sensors will initiate signals that will alert the operator to bring the plant to a shutdown condition.

3/4.3.3.8 POST ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available for selected plant parameters to monitor and assess these variables following an accident.

3/4.3.3.9 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each RCS accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 BORON INJECTION SYSTEM

The OPERABILITY of the boron injection system as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident or a steam line rupture.

The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensure that the solubility of the boron solution will be maintained above the solubility limit of 135°F at 21,000 ppm boron.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration

3/4.5.5 REFUELING WATER STORAGE TANK (Continued)

ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

ACTIVITY (Continued)

of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.1.6 SECONDARY WATER CHEMISTRY

THIS SPECIFICATION DELETED

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator average impact values taken at 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.5 RIVER WATER SYSTEM

The OPERABILITY of the River water system ensures that sufficient cooling capacity is available to the service water system for continued operation of vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.6.1 ULTIMATE HEAT SINK (River)

The limitations on the ultimate heat sink level ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level is based on providing a 30 day cooling water supply to safety related equipment.

3/4.7.6.2 ULTIMATE HEAT SINK (POND)

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits.

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 9,723 ± 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new fuel storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between fuel assemblies placed in the storage racks. The spent fuel storage racks are designed and shall be maintained with a nominal 13 inch center-to-center distance between fuel assemblies placed in the storage racks. This will ensure a K_{eff} equivalent of $\leq$ 0.95 for either storage pool filled with unborated water. The K_{eff} of $\leq$ 0.95 includes a conservative allowance of 3.84% ΔK for uncertainties.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 149.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 675 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.9-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.9-1.

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

f. Records of transient or operational cycles for those facility components identified in Table 5.7-1.

g. Records of reactor tests and experiments.

h. Records of training and qualification for current members of the plant staff.

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

j. Records of Quality Assurance activities required by the QA Manual.

k. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

l. Records of meetings of the PORC and the NORB.

m. Records of secondary water sampling and water quality.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c) (2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

*Health Physics personnel or personnel escorted by health physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation Work Permit.

6.12.2 The requirements of 6.12.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or the Chemistry and Health Physics Supervisor.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 8 TO FACILITY OPERATING LICENSE NO. DPR-66

ALABAMA POWER COMPANY

FARLEY NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-348

INTRODUCTION

By letters dated November 4 and December 14, 1977 and August 9, 1978, Alabama Power Company (APC) proposed changes to the Appendix A Technical Specifications for Farley Nuclear Plant (FNP), Unit No. 1. The proposals included the following:

1. Modification to Specification Figure 3.2-3 and Bases 2.1.1 relating to the fuel rod bow penalty for 17x17 fuel designed by Westinghouse.

2. Modifications to Specification 4.5.2 to increase the surveillance on ECCS subsystems (high pressure and low pressure safety injection) in response to our request dated August 30, 1977.

3. Administrative changes to Specifications 5.6.1 and 5.6.3 relating to storage of the spent fuel assemblies.

4. Modifications to Specification 4.3.3.2 and Bases 4.3.3.2 to authorize use of quarter-core flux maps for excore neutron flux detection system calibration.

5. Deletion of Specifications 3.7.1.6, and 4.7.1.6, Tables 3.7-3, 3/4.7-3 and Bases 3/4.7.1.6 and adding specification 6.10.2.m and license condition 2.C.(3)(g), all relating to secondary water chemistry.

6. Revision of Specification 3.4.9.2 to change the pressurizer heatup rate in response to our letter dated November 23, 1977.

Some modifications to APC's proposals were necessary to meet our requirements. These modifications were discussed with and agreed to by the APC staff.

DISCUSSION AND EVALUATION

1. Rod Bow Penalty versus Region Average Burnup (Figure 3.2-1 and Bases 2.1.1)

By letter dated August 9, 1978 APC proposed revisions to the rod bow penalty (RBP) versus core region average burnup, Specification

Figure 3.2-3. The existing figure was derived based on a conservative extrapolation of 15x15 fuel data rather than data for the 17x17 fuel existing in the FNP core.

The RBP became of interest to the NRC staff in August 1976 when Westinghouse reported data to the staff from Departure from Nucleate Boiling Ratio experiments using electrically heated simulated fuel rods. Data indicated that a fuel rod which bowed to contact with a thimble tube or with a thimble tube present in the subchannel would experience a significant reduction in Departure from Nucleate Boiling Ratio over that of the unbowed geometry[1]. A calculational model to predict the amount of Departure from Nucleate Boiling Ratio reduction in 17x17 fuel (such as that in the Farley reactor) was proposed by Westinghouse[2] and accepted by the staff with some modifications[3]. This model was based on the extrapolation of rod bowing data from Westinghouse 15x15 fuel. The existing Specification Figure 3.2-3 was derived from this data.

APC proposed a new equation (see Bases 2.1.1) derived by Westinghouse for predicting the amount of rod bow in 17x17 fuel. This new equation is developed from rod bow data from 17x17 lead test assemblies irradiated in the Surry Nuclear Plant Unit No. 1 and Unit No. 2 reactors. The staff has previously accepted this model for the Trojan reactor[4].

[1]Letter to V. Stello, Jr., USNRC from C. Eicheldinger, Westinghouse Electric Corporation, NS-CE-1161, August 13, 1976.

[2]Reavis, J. R., et al., "Fuel Rod Bowing", Westinghouse Electric Corporation, WCAP 8691, December 1975.

[3]Memorandum for D. B. Vassallo and K. R. Goller, USNRC, from D. F. Ross and D. G. Eisenhut, USNRC, "Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors", February 16, 1977.

[4]Safety Evaluation supporting Amendment 30 to License NPF-1 for Trojan Nuclear Plant, June 22, 1978.

Based on our review of the APC proposal and on the previous evaluation done during the Trojan Nuclear Plant Amendment No. 30 review, we agree the changes are also acceptable for the FNP reactor since it gives a sufficiently conservative estimate of the DNBR reduction.

2. HPSI and LPSI Surveillance (Specification 4.5.2)

In our August 30, 1977 letter we requested APC consider increased surveillance on safety injection system valves. We also requested flow balance tests as an added surveillance check when system alterations occur. By letter dated November 4, 1977 APC proposed changes to Technical Specification 4.5.2.e and adding 4.5.2.h relating only to system valves. However, following discussion with our staff, APC proposed flow balance checks (Specification 4.5.2.i) by letter dated August 9, 1978.

The High and Low Pressure Safety Injection system (HPSI and LPSI) designs of many Pressurized Water Reactors (PWR) utilize a common low pressure and a common high pressure header to feed the several cold (and in some cases hot) leg injection points. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration; (2) provide a proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. On many plants, there are motor operated valve(s) in the lines to each injection point that have stops which are set during pre-operational flow testing of the plant to insure that these flow requirements are satisfied. On other plants, electrical or mechanical stops on the Safety Injection System's isolation valve(s) are used for this purpose. FNP utilizes mechnical stops to satisfy these ECCS flow requirements.

While pre-operational HPSI/LPSI flow testing is utilized to assure that the valves used to throttle flow have been properly set, the NRC staff has concluded that periodic surveillance requirements are needed to assure that these settings are maintained throughout the life of the plant. Consequently, we requested all PWR licensees to propose changes to their Technical Specifications, as appropriate, to incorporate periodic surveillance requirements for these valves. Sample surveillance requirements, developed by the NRC staff, were provided to licensees for guidance in developing proposed changes.

The sample requirements include periodic verification of throttle valve position stop settings, and verification of proper ECCS flow rates whenever system modifications are made that could alter flow characteristics. Our request for proposed Technical Specification changes was sent to APC as noted above.

APC's response, as amended by letter of August 9, 1978, to our request with respect to Farley Nuclear Plant, Unit No. 1 contains proposed changes to the Technical Specifications that are in essential agreement with the staff's requirements. Based on our review, we have concluded that the licensee's proposed increased surveillance requirements would provide sufficient additional assurance that proper valve settings for the ECCS flows and flow distributions will be maintained throughout plant life. Thus, the proposed changes are acceptable.

3. Administrative Changes (Specification 5.6.1 and 5.6.3)

By letter dated August 9, 1978 APC proposed changes to update Technical Specifications 5.6.1 and 5.6.3, Design Features. The updating would correct the specifications to be consistent with the Final Safety Analysis Report (FSAR) description of the spent fuel storage facility. In addition, a slight increase in the uncertainty allowance used by Westinghouse for $\Delta k/k$ was proposed.

The changes include the following:

(1) Racks center to center: 21 inch to 13 inch
(2) Number of storage racks: 212 to 675
(3) $\Delta k/k$ uncertainty: 3.3% to 3.84%

Amendment No. 55 dated May 4, 1976 to the FSAR for Farley describes changes from the original design proposed by APC. Changes (1) and (2) above were reviewed by the NRC staff prior to issuance of License NPF-2 on June 25, 1977. Therefore, the corresponding Technical Specification changes are editorial in nature and are acceptable. The increase in the $\Delta k/k$ uncertainty assumption made by Westinghouse provides added conservatism to criticality calculations. Thus, this minor administrative change is acceptable.

4. <u>Quarter-Core Flux Maps for Calibration of Excore Neutron Flux Detection System (Specifications 3.3.3.2, 4.3.3.2 and Bases 4.3.3.2)</u>

By letter dated August 9, 1978 APC proposed changes which would allow use of quarter-core flux maps for calibration of the excore neutron flux detection system. The NRC staff has reviewed and approved Westinghouse Topical Report, WCAP-8648, "Excore Detector Calibration Using Quarter Core Flux Maps". Our letter dated November 11, 1977 approved the Topical Report.

Our intention is that WCAP-8648 is acceptable for reference to justify using quarter-core flux maps for such calibrations at reactor plants designed by Westinghouse. Since the technique presented in WCAP-8648 is applicable to the reactor design of FNP, the proposed specifications are acceptable.

5. <u>Deletion of Secondary Water Chemistry (Specifications 3.7.1.6, 4.7.1.6 and Bases 3/4.7.1.6)</u>

By letter dated August 9, 1978 APC proposed deletion of certain secondary water chemistry specifications. The existing specifications contain a limiting condition of operation, yet actual limitations do not exist. APC was to determine the limiting values after about six months of operation. Then, the values would have been reviewed by the NRC staff and put into existing blank tables in Specification 3.7.1.6 if appropriate. However, APC proposed not to establish secondary water chemistry limits.

<u>Evaluation</u>

The NRC staff recognizes that different utilities use different secondary water treatment methods to limit steam generator tube corrosion. Moreover, we recognize that a licensee's choice of a particular water treatment method, including specific values of operating limits for chemistry parameters, is governed by plant and site characteristics that are unique to each facility. In addition, we do not believe at this time that sufficient service experience exists to conclude that any particular method is superior to another for controlling impurities that may be introduced into the secondary coolant. Such experience would be necessary to assure that Technical Specifications on secondary water chemistry will ensure minimum tube degradation.

Restricting the amount of chemical additions to control the water chemistry parameters would not ensure the desired steam generator operating conditions. Realizing that meeting the secondary coolant water quality criteria would not be possible during all periods of operation, it is necessary that the most effective procedure for reestablishing out-of-specification chemistry parameters be available without unduly restricting plant operations. This can be accomplished most rapidly by continuing to operate the unit so that chemical additives to the secondary water can be made to achieve a balanced chemistry. During discussions with APC personnel, we were advised that permanent records are kept of all chemical additives used. Such records would be available if needed for our future evaluations. We consider that these permanent plant records on a sampling program may be useful in the future. Thus, Specification 6.10.2.m was added identifying records of secondary water sampling and water quality for retention. The APC staff agreed to this addition since they already retain such records for the life of the plant.

In particular, Technical Specifications 3.7.1.6 and 3/4.7.1.6 for secondary water chemistry do not provide adequate flexibility to allow desired water conditions to be achieved gradually or ensure long term tube integrity. In addition, these specifications may not limit specific types of severe tube degradation, particularly "denting". Furthermore, the possible adverse effects of any secondary water parameter limits on the steam purity that could lead to potential failure of turbine components must also be considered before specific limits are required.

We believe that other methods for reducing the impurity concentration in the steam generator such as periodic chemical cleaning for long term solution, flushing or free surface boiling for an intermediate term solution, or the use of chelating agents for the control of secondary water purity are more practical. These methods are likely to be more effective in limiting corrosion than specific Technical Specifications that may lack the flexibility needed for proper control of secondary water chemistry. The NSSS vendors are now considering these alternate methods in lieu of restrictive secondary water chemistry for assuring steam generator tube integrity. We have also added license condition 2.C.(3)(g) requiring APC to implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. The APC staff has agreed to the program.

In addition, other existing Technical Specification limiting conditions for operation and surveillance requirements for secondary water monitoring requirements provide assurance that steam generator tube integrity is not reduced below an acceptable level for adequate margins of safety. These specifications are:

1. Technical Specification 3.7.1.4 - Secondary Water Monitoring Requirements

2. Technical Specification 3.4.6.2 - Primary to Secondary Leakage Rates

3. Technical Specification 3.4.5 - Steam Generator Tube Surveillance and Plugging Criterion

Since the current state-of-the-art for steam generator secondary water treatment has not proven to be the ultimate solution in limiting various forms of tube degradation to ensure tube integrity, we do not believe that a specific Technical Specification for the control of secondary water quality would be appropriate at this time. Consequently, we agree that Technical Specifications 3.7.1.6 and 4.7.1.6 be deleted.

Conclusion

Based on the discussion above we consider that the deletion of the secondary water chemistry Technical Specification will not cause a significant decrease in margin of safety or involve a significant hazard consideration. There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

6. Pressurizer Heat-Up Rate (Specification 3.4.9.2)

By letter dated December 14, 1977 APC proposed to reduce the maximum pressurizer heatup rate from 200°F per hour to 100°F per hour. This proposal was in response to our letter dated November 23, 1977.

Background

In August 1977, Mitsubishi Heavy Industries, Ltd., of Japan, noted an inconsistency in the pressurizer heatup rate stated in their Technical Specifications. This discrepancy was reported to the Westinghouse Electric Corporation (Westinghouse), who then reviewed their analysis of the pressurizer heatup rate and determined that the correct heatup rate is

100°F per hour, and that the correct cooldown rate is 200°F per hour. Technical Specification 3.4.9.2.a for FNP specifies pressurizer heatup and cooldown rates of 200°F per hour. Westinghouse then notified the Nuclear Regulatory Commission (the Commission) and the licensee of this problem. The requested amendment would correct the error in the pressurizer heatup rate limit.

Evaluation

In designing the pressurizer, Westinghouse performed a thermal stress analysis which analyzed the fatigue resulting from a heatup rate of 100°F per hour and a cooldown rate of 200°F per hour. This analysis meets the standards of the ASME Code, Section III, which requires that the analysis be based on a usage factor. The usage factor represents the fraction of the fatigue life (the total amount of stress that a particular component is designed to handle), with a usage factor of zero implying that no stress has been exerted on the component, and a usage factor of one implying that the stress exerted on the component is equal to the amount of stress that the component is designed to handle. For any system or large piece of equipment, certain components receive more stress than others. For the pressurizer, the limiting component is the surge nozzle, which has a usage factor of 0.9 for the design numbers listed above. This usage factor is such that if the heatup and cooldown rates used in the analysis were exceeded more than a few times, the actual usage factor for the surge nozzle would exceed 1.0, which is not allowable under the ASME Code.

Because the current FNP Technical Specification 3.4.9.2.a authorized higher rates of pressurizer heatup than the correct limit, the question arose as to whether the correct limit of 100°F per hour has been exceeded in the past. Discussions with Westinghouse indicate that this is unlikely. System capabilities and Technical Specification limits on the rate of reactor coolant system heatup and pressurization effectively preclude pressurizer heatup rates in excess of 50°F to 75°F per hour. Furthermore, APC reviewed startup tests for FNP and performed calculations after being informed of this potential problem. Calculations performed show that the maximum heatup rate possible, assuming the maximum heater power and smallest mass of water in the pressurizer, is less than 100°F per hour. Accordingly, we conclude that the only action required by APC for FNP is modification of the Technical Specifications as proposed to reduce the limiting pressurizer heatup rate of 200°F per hour to 100°F per hour.

On September 25, 1978 Westinghouse was requested to perform an audit review of the stress analyses for components of the reactor coolant pressure boundary to assure that no similar inadvertent error appears in any other portion of the applicable Technical Specifications. By letter dated October 27, 1978 Westinghouse responded by stating that in the past year it had carefully reviewed the stress analysis inputs to the Technical Specifications for five separate plants.
In addition, they are completing a very careful, systematic review on the generic June 15, 1978 version of the Westinghouse Standard Technical Specifications (STS). If any further inconsistencies surface during this review process, suitable action would be taken (in the forum of the Westinghouse STS). We find this response acceptable.

Conclusion

Based on our review of APC's proposal as discussed herein, we conclude that reducing the heatup rate limit from 200°F per hour to 100°F per hour is necessary to maintain thermal stresses in the pressurizer to allowable levels. For the same reasons, we further conclude that the cooldown rate limit presently in the Technical Specifications is adequate.

ENVIRONMENTAL CONSIDERATIONS

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmetnal impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequencs of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: February 13, 1979

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-348

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY

OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 8 to Facility Operating License No. NPF-2, issued to the Alabama Power Company which revised Technical Specifications for operation of the Joseph M. Farley Nuclear Plant, Unit 1 (the facility) located in Houston County, Alabama. The amendment was effective as of its date of issuance.

The amendment revises the Technical Specifications relating to the fuel rod bow penalty for 17x17 fuel, the surveillance of ECCS subsystems (high pressure and low pressure safety injection), administrative changes to the description of the spent fuel storage facility, use of quarter-core flux maps for excore neutron flux detection system calibration, secondary water chemistry, and changing the pressurizer heatup rate. The amendment also adds a license condition to implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated November 4 and December 14, 1977 and August 9, 1978, (2) Amendment No. 8 to License No. NPF-2, and (3) the Commission's related Safety Evaluation. These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the George S. Houston Memorial Library, 212 W. Vurdeshaw Street, Dothan, Alabama 36301. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

7590-01

Dated at Bethesda, Maryland, this 13th day of February, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors