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October 14, 1982

Docket No. 50-369

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Issuance of Amendment No. 17 to Facility Operating License NPF-9 - McGuire Nuclear Station, Unit 1

The Nuclear Regulatory Commission has issued Amendment No. 17 to Facility Operating License NPF-9 for the McGuire Nuclear Station, Unit 1, located in Mecklenburg County, North Carolina.

This amendment is in response to your letters dated March 2 and March 9, 1982. The amendment permits reduction in boron concentration in the boron injection tank from a nominal 20,000 ppm to 2,000 ppm, deletion of the Technical Specification on heat tracing for the boron injection tank, and revises minimum limit for primary containment upper compartment average air temperature. The amendment is effective 72 hours after its date of issuance.

A copy of the related safety evaluation report supporting Amendment No. 17 to Facility Operating License NPF-9 is enclosed. Also enclosed is a copy of a related notice which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

"Original Signed By:

Elinor G. Adensam, Chief
~~Licensing Branch No. 4~~
Division of Licensing

Enclosures:
1. Amendment No. 17
2. Safety Evaluation
3. Federal Register Notice

cc w/encl:
See next page

OFFICE	LA:DL:LB #4	DL:LB #4	DL:LB #4				
SURNAME	MDuncan/hmc	RBirkel	EAdensam				
DATE	10/7/82	10/7/82	10/14/82				

McGuire

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

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Raleigh, North Carolina 27603

County Manager of Mecklenburg County
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Mr. Bruce Blanchard
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Department of the Interior
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345 Courtland Street, N.E.
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Chairman, North Carolina
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430 North Salisbury Street
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Raleigh, North Carolina 27602

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DUKE POWER COMPANY

DOCKET NO. 50-369

MCGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.17
License No. NPF-9

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

   A. The applications for amendment to the McGuire Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-9 filed by the Duke Power Company (licensee) dated March 2 and March 9, 1982, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-9 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 17, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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DA					

3. This license amendment is effective 72 hours after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

"Original Signed By:
Elinor G. Adensam, Chief
~~Licensing Branch No.~~ 4
Division of Licensing

Attachment:
Technical Specification
Changes

Date of Issuance: October 14, 1982

No legal objection to form of amendment

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SURNAME	MDuncan/hmc	RBirkel	BSheron	GCotchin	EAdensam	TNovak	
DATE	10/7/82	10/7/82	10/8/82	10/12/82	10/14/82	10/14/82	

ATTACHMENT TO LICENSE AMENDMENT NO. 17

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Amended Page	Overleaf Page
3/4 5-11	
3/4 5-12	
B3/4 5-2	B3/4 5-1
B3/4 6-2	B3/4 6-1
3/4 6-11	3/4 6-12

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3/4.5.4 BORON INJECTION SYSTEM

BORON INJECTION TANK

LIMITING CONDITION FOR OPERATION

3.5.4.1 The boron injection tank shall be OPERABLE with:

a. A minimum contained borated water volume of 900 gallons, and

b. Between 2,000 and 4,000 ppm of boron.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the boron injection tank inoperable, restore the tank to OPERABLE status within 1 hour or be in HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.1 The boron injection tank shall be demonstrated OPERABLE by:

a. Verifying the contained borated water volume at least once per 7 days, and

b. Verifying the boron concentration of the water in the tank at least once per 7 days.

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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 from the cold leg injection accumulators and directly into the reactor vessel from the upper head injection accumulators 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.

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.

ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 300°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

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.

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 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.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 1.5 psig and 2) the containment peak pressure does not exceed the design pressure of 15 psig during LOCA conditions.

INTERNAL PRESSURE (Continued)

The maximum peak pressure expected to be obtained from a LOCA event is 14.5 psig. The limit of 0.3 psig for initial positive containment pressure will limit the total pressure to 14.8 psig which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limits of 100°F for the lower compartment, 75°F for the upper compartment, and 60°F when less than or equal to 5% of RATED THERMAL POWER will limit the peak pressure to 11.8 psig, which is less than the containment design pressure of 12 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 15 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.7 REACTOR BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment reactor building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide 1) protection for the steel vessel from external missiles, 2) radiation shielding in the event of a LOCA, and 3) an annulus surrounding the steel vessel that can be maintained at a negative pressure during accident conditions. A visual inspection is sufficient to demonstrate this capability.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall be maintained:

a. between 75°F* and 100°F in the containment upper compartment, and

b. between 100°F* and 120°F in the containment lower compartment.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature not conforming to the above limits, restore the air temperature to within the limits within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The primary containment upper compartment average air temperature shall be the weighted average** of all ambient air temperature monitoring stations located in the upper compartment. As a minimum, temperature readings will be obtained at least once per 24 hours from the following locations:

Location

a. Elev. 826' at the inlet of upper containment ventilation Unit 1A.

b. Elev. 826' at the inlet of upper containment ventilation Unit 1B.

c. Elev. 826' at the inlet of upper containment ventilation Unit 1C.

d. Elev. 826' at the inlet of upper containment ventilation Unit 1D.

*Lower limit may be reduced to 60°F in MODE 2, 3 and 4.

**The weighted average is the sum of each temperature multiplied by its respective containment volume fraction. In the event of inoperable temperature sensor(s), the weighted average shall be taken as the reduced total divided by one minus the volume fraction represented by the sensor(s) out of service.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.5.2 The primary containment lower compartment average air temperature shall be the weighted average* of all ambient air temperature monitoring stations located in the lower compartment. As a minimum, temperature readings will be obtained at least once per 24 hours from the following locations:

Location

a. Elev. 745' at the inlet of lower containment ventilation Unit 1A.

b. Elev. 745' at the inlet of lower containment ventilation Unit 1B.

c. Elev. 745' at the inlet of lower containment ventilation Unit 1C.

d. Elev. 745' at the inlet of lower containment ventilation Unit 1D.

*The weighted average is the sum of each temperature multiplied by its respective containment volume fraction. In the event of inoperable temperature sensor(s), the weighted average shall be taken as the reduced total divided by one minus the volume fraction represented by the sensor(s) out of service.

SAFETY EVALUATION BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 17

TO LICENSE NPF-9

DUKE POWER COMPANY

PART A: Dilution of Boron Concentration in the Boron Injection Tank

PART B: Containment Upper Compartment Temperature

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SAFETY EVALUATION REPORT

DILUTION OF BORON CONCENTRATION IN THE BORON INJECTION TANK

McGUIRE NUCLEAR STATION UNIT 1

BACKGROUND

Westinghouse has incorporated a Boron Injection Tank, containing a highly concentrated boron solution (20000 ppm), into their nuclear steam supply system design to meet the requirements of the Standard Review Plan Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)." The acceptance criteria for this event seek assurances that the capability to cool the core is maintained and that the resulting offsite dosage complies with the requirements set forth in 10 CFR 100. For postulated steam line break events, a potential for return to criticality exists as the moderator temperature decreases. The Boron Injection Tank (BIT) was specifically designed to mitigate the consequences of this event with the high-head safety injection system (HHSI) by purging the highly concentrated boron solution (20,000 ppm) into the primary system.

Experience with the BIT has placed excessive maintenance requirements upon the plant operators and technicians. As a result, the licensee has proposed to reduce the BIT boron concentration, remove the heat tracing, and change the boron injection system technical specification. (Ref. 1).

EVALUATION

The BIT was designed to mitigate the consequences of postulated steam line break events. During these events, the high head safety injection pumps automatically align to discharge through the BIT, which contains 900 gallons of highly concentrated boric acid solution (20,000 ppm). This solution is then flushed into the primary system to assure adequate shutdown reactivity. The current requirement for a high boron concentration in the BIT was a result of conservatism in the previous safety analysis. To justify the reduction in BIT boron concentration, the licensee reanalyzed the following events assuming a BIT concentration of only 2000 ppm boron: (i) rupture of a main steam line, (ii) accidental depressurization of the main steam system, and (iii) inadvertent operation of the ECCS during power (Ref. 1).

The steam line rupture accident was analyzed with the assumption of a stuck RCCA, with or without offsite power, a single failure in the safety injection system and a break area of 1.4 ft^2, (i.e., the flow restrictor area). The LOFTRAN code had been used to calculate the core heat flux and RCS temperature and pressure. The

minimum DNBR was calculated from the THINC code. The results indicated that the reactor returned to power and the maximum heat flux was 19.9% of the design value, the maximum RCS pressure was 1303 psia, well below the 110% design pressure. With regard to the DNBR, the licensee, in a letter dated April 19, 1982, (Ref. 2) responded to our request for additional information and indicated that the DNBR evaluated shows the same general trend as the plot of DNBR vs. time in WCAP-9225 (Steamline Break Topical Report) for similar plant types. The DNBR did not decrease below 1.3.

The event of accidental depressurization of the main steam system was analyzed with the LOFTRAN code assuming a stuck RCCA, with or without offsite power, and a single failure in the Engineered Safety Features. The case analyzed was a steam flow of 248 lbs/sec at 1100 psia from one steam generator with offsite power available. The steam flow rate was the maximum capacity of any single steam dump, relief or safety valve. The results indicated that with one charging pump in operation supplying a boron solution at 2000 ppm to the RCS, sufficient negative reactivity was provided to prevent the reactor from returning to power. Although only five state points were evaluated, the licensee stated that the DNBR showed the same general trend as the main steam line break event and did not fall below the value of 1.3. This event is less limiting than the steam line rupture accident.

The inadvertent operation of ECCS during the power operation event was analyzed with the digital computer program LOFTRAN assuming initial reactor power at 102%, and a low absolute value of the Doppler Power coefficient. The licensee stated that because of the power and temperature reduction during the transient, operating conditions did not approach the core limits and the results were relatively independent of time to reactor trip. The licensee further stated that spurious safety injection with or without reactor trip would not affect the integrity of the reactor coolant system. The DNBR was always greater than the initial value of 1.62.

With regard to the deletion of the Technical Specification on heat tracing for the BIT, the licensee stated that the current requirement for heat tracing was due to high boron concentration in the BIT and associated piping. Reduction of boron concentration to less than 4000 ppm would eliminate the need for heat tracing. Heat tracing would be required for boron concentration above 4 weight percent, corresponding to approximately 7000 ppm.

CONCLUSION

The staff has reviewed Duke Power Company's submittal for dilution of boron concentration in the BIT and related Technical Specification changes for McGuire Nuclear

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Station Unit 1. The supporting analysis demonstrated compliance to Sections 15.1.1, 15.1.2, 15.1.3, 15.1.4, 15.1.5, 15.5.1, and 15.5.2 of the Standard Review Plan, while assuming 2000 ppm boron concentration in the BIT.

The analytical methodology (i.e., use of LOFTRAN) for evaluating the accident events discussed previously is presently undergoing staff review. Our review at this time indicates reasonable assurance that the conclusions based on the licensee's submittal will not be appreciably changed by completion of review. Although limited clad perforation following a steam line break event is permitted by the SRP, the licensee has demonstrated that no clad perforation is calculated to occur. Therefore, there exists adequate margin of safety to acceptable limits as specified in the SRP. Moreover, we conclude that because the acceptance criterion of a DNBR greater than 1.3 is met both for the low and high boron concentration, the safety margin has not been significantly reduced. Based on our review of the licensee's evaluation, the staff concludes that the licensee's proposed Technical Specification modifications to reduce the allowable boron concentration and remove the heat tracing for the BIT are acceptable.

REFERENCE

1. Letter to H. R. Denton from W. O. Parker, McGuire Nuclear Station Unit 1, Docket No. 50-369, Proposed Amendment to License NPF-9, March 2, 1982.

2. Letter to H. R. Denton from W. O. Parker, McGuire Nuclear Station Unit 1, Docket No. 50-369, Proposed Amendment to License NPF-9, April 19, 1982.

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SAFETY EVALUATION REPORT

CONTAINMENT UPPER COMPARTMENT TEMPERATURE

McGUIRE STATION UNIT 1

BACKGROUND

The McGuire Nuclear Station utilizes an ice condenser as a pressure mitigating system in containment. As shown on the attached figure the McGuire containment is physically divided into an upper and lower compartment. The barrier is designed to prevent steam from bypassing the ice condenser in the event of a postulated pipe rupture. Thus steam emanating from either a loss of coolant accident or a main steam line break inside containment will be forced to flow through the ice baskets which are located along the containment's perimeter. Since the steam resulting from a pipe break is largely condensed in the ice condenser, the containment internal design pressure is only 15 psig.

The air temperature inside containment is closely monitored during operating modes 1, 2, 3 and 4. The containment peak pressure analyses in the FSAR assumes a minimum upper compartment temperature of 75°F and a minimum lower compartment temperature of 100°F. These minimum temperatures maximize the initial mass of air in containment. The initial mass of air is important in calculating the limiting containment internal pressure.

McGuire Unit 1 was shut down for most of December 1981 due to repairs of the main turbine. No heat was generated in the Reactor Building during this time, and thus the cold weather was able to slowly cool the building. Also, the equipment hatch to the outside was briefly opened to allow some equipment to be taken into containment. Due to the great amount of thermal mass in the reactor building, it took a period of time for the cold to affect the temperature in containment. When the licensee attempted to bring the Unit back to power in early January 1982, the upper compartment average temperature fell below the minimum allowable. Technical Specification 3.6.1.5 conservatively requires a minimum upper compartment temperature of 85°F and a minimum lower compartment temperature of 100°F.

When the average temperature in containment began reaching the lower limit and after it dropped below the limit, the temperature was increased by using the H_2 recombiner heaters. On January 1 and 2, 1982 the hyrdogen recombiners were operated four separate times in order to increase the upper compartment temperature above the minimum valve of 85°F (see LER 82-03, Reference 1).

On March 2, 1982, the licensee submitted a proposed Technical Specification change to lower the minimum upper compartment temperature from 85°F to 75°F.

EVALUATION

Due to the ice condenser system the McGuire containment is only designed to 15.0 psig. The peak calculated containment pressure is 14.8 psig. This calculation was performed using the Westinghouse LOTIC-3 computer code which has been reviewed and found to be acceptable by the staff.

As stated previously the initial mass of air in containment can have a significant affect on the peak pressure transient calculation. This is particularly true for small containments incorporating low design pressures. By placing a technical specification limit on the minimum operating temperature, the initial mass of air present in containment is limited. If a large pipe break is postulated inside containment, the containment pressure is increased by both the partial pressure of steam and the partial pressure of air as it is heated and expands.

The upper and lower compartment temperature limits are not standard for all ice condensers and are calculated on a plant specific bases. Generally speaking, plants that have a small margin between the peak calculated and design pressure for containment must maintain a relatively high minimum operating temperature in order to reduce the partial pressure of air. This can be seen in the cases of McGuire and Sequoyah in the table below. Bounding calculations using the ideal gas law show that both the McGuire and Sequoyah peak calculated pressures inside containment would approach design conditions if the minimum operating temperatures were lowered by approximately 10°F. Conversely, plants with a relatively large margin between calculated and design pressures such as D.C. Cook are able to operate with lower initial containment temperatures.

	D.C. Cook	Sequoyah	McGuire
Containment Design Pressure (psig)	12.0	12.0	15.0
Calculated Containment Pressure (psig)	9.4	11.8	14.8
Upper Compartment Temp. Range (°F)	60-100	85-110	75*-100
Lower Compartment Temp. Range (°F)	60-120	100-120	100-120

*Proposed

Since the limiting containment pressure calculations found in the McGuire FSAR consistently assume 75°F as the minimum upper compartment temperature, there is no loss of margin between the proposed technical specification and that found acceptable in the staff's Safety Evaluation Report. The current value of 85°F found in the McGuire Technical Specifications is unnecessarily conservative and has no apparent basis.

In addition, during the course of our review, we realized that McGuire's Technical Specification Bases section 3/4.6.1.5 inadvertently quotes the wrong peak calculated and design pressure for containment. These values have been corrected in the Bases.

CONCLUSION

Based on our review of the licensee's submittal, we conclude that the proposed Technical Specification change reducing the minimum average air temperature in the primary containment upper compartment from 85°F to 75°F does not have any adverse effect on safety of plant operation or the health and safety of the public.

The proposed Technical Specification 3.6.1.5 and the revised Bases section 3/4.6.1.5 is attached.

REFERENCES

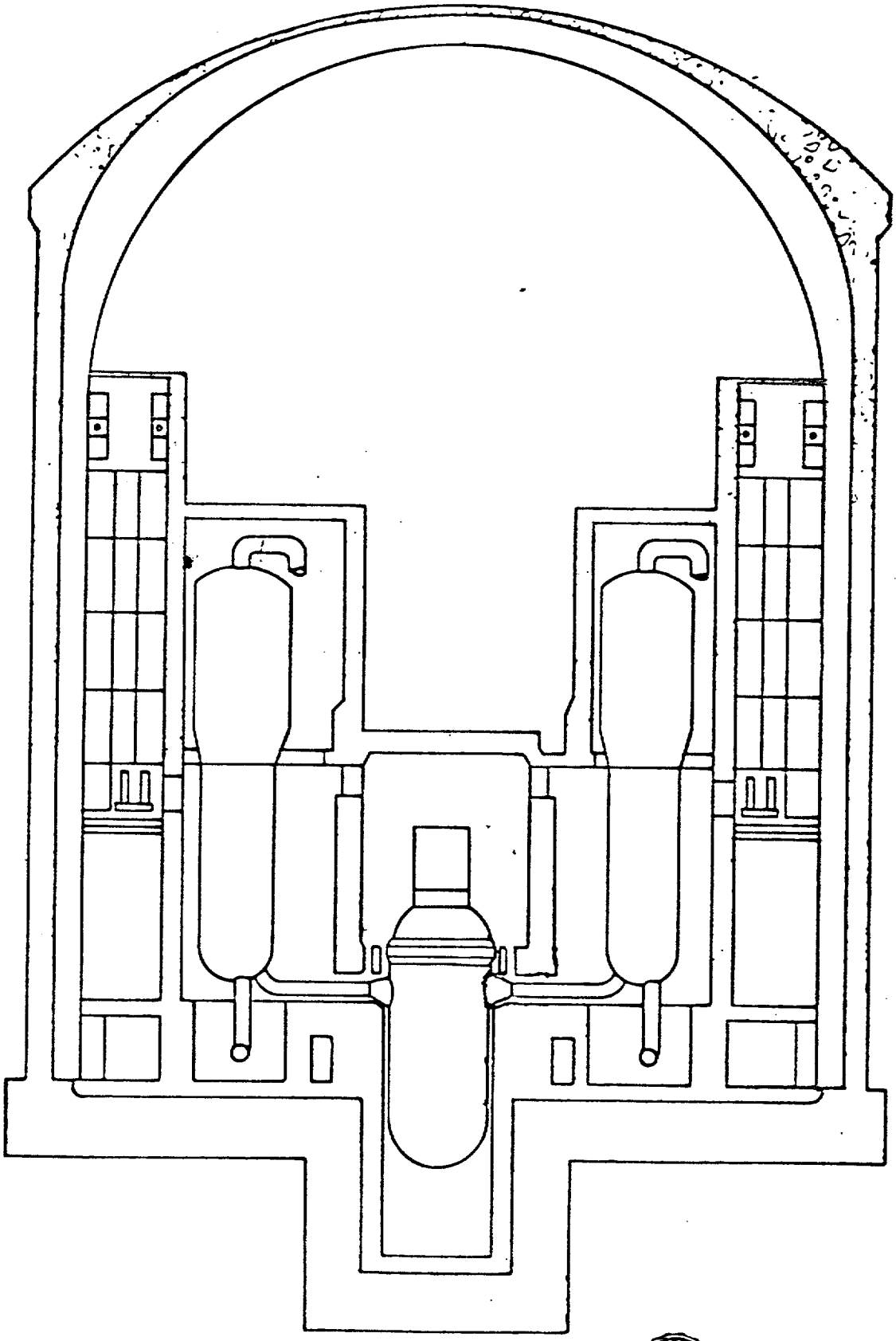
1. Letter from Duke Power Company to NRC Region II (William O. Parker to James P. O'Reilly) dated February 1, 1982. Includes Reportable Occurrence Report RO-369/82-03.

2. Letter from Duke Power Company to NRC (William O. Parker to Harold R. Denton) dated March 2, 1982.

3. McGuire Nuclear Station FSAR, Chapter 6 and Technical Specifications.

4. D.C. Cook Technical Specifications.

5. Sequoyah Technical Specifications.



McGUIRE NUCLEAR STATION

ENVIRONMENTAL CONSIDERATION

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 environmental impact and, pursuant to 10 CFR Section 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 consideration discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered, does not create the possibility of an accident of a type different from any evaluated previously 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: October 14, 1982

Principal Contributors: V. Leung, RSB
D. Pickett, ORAB
R. Birkel, LB #4

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-369

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT

FACILITY OPERATING LICENSE NO. NPF-9

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 17 to Facility Operating License No. NPF-9, issued to Duke Power Company (licensee) for the McGuire Nuclear Station, Unit 1 (the facility) located in Mecklenburg County, North Carolina. The amendment is effective 72 hours after its date of issuance.

The amendment permits reduction in boron concentration in the boron injection tank from a nominal 20,000 ppm to 2,000 ppm, deletion of the Technical Specification on heat tracing for the boron injection tank, and revises minimum limit for primary containment upper compartment average air temperature.

Issuance of this amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's 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.

OFFICE							
SURNAME							
DATE							

For further details with respect to this action, see (1) Duke Power Company letters dated March 2 and March 9, 1982, (2) Amendment No. 17 to Facility Operating License No. NPF-9 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 the Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223. A copy of these items may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 14th day of October 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

"Original Signed By:

Elinor G. Adensam, Chief
~~Licensing Branch No.~~ 4
Division of Licensing, NRR

No legal objection to form of FR notice

OFFICE	LA:DL:LB #4	DL:LB #4	OELD	DL:LB #4			
SURNAME	MDuncan/hmc	RBirkel	CUTCHIN	EAdensam			
DATE	10/7/82	10/7/82	10/12/82	10/14/82			

October 13, 1982

Docket No. 50-369

MEMORANDUM FOR: Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

THRU: Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

FROM: Ralph A. Birkel, Project Manager
Licensing Branch No. 4
Division of Licensing

SUBJECT: ISSUANCE OF AMENDMENT NO. 17 TO FACILITY OPERATING LICENSE NPF-9 McGUIRE NUCLEAR STATION, UNIT 1

There is no known public correspondence or irreversible impact associated with the issuance of the subject amendment.

Ralph A. Birkel, Project Manager
Licensing Branch No. 4
Division of Licensing

OFFICE	LA:DL:LB #4	DL:LB #4	DL:LB #4				
SURNAME	MDuncan/hmc	RBirkel	EAdensam				
DATE	10/7 /82	10/7 /82	10/13/82				