DUKE POWER COMPANY

POWER BUILDING, CHARLOTTE, N. C. 28242

June 26, 1984

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. Harold R. Denton, Director

Re: Catawba Nuclear Station, Unit 1
Docket No. 50-413

Subject: Applicants' Application for Partial
Exemption from GDC 17

Dear Mr. Denton:

Pursuant to 10 CFR §50.12, Duke Power Company, et al. (Applicants) hereby request an exemption from the requirement of 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 17, as such relates to fuel load and pre-critical testing activities.

On April 11, 1984, Applicants filed a motion with the Atomic Safety and Licensing Board requesting authorization to load fuel and conduct certain pre-critical testing. On May 1, 1984, the NRC Staff filed a response to this motion which supported the Applicants' conclusion that these activities can be conducted without endangering the health and safety of the public and in compliance with applicable regulations. Staff response at 3.

Applicants' motion discussed in detail the activities to be conducted under the authorization sought. During at least a portion of that time, one or both diesel generators for Catawba Unit 1 will not be available for service. As discussed in meetings with the Staff on March 21 and June 21, 1984, Applicants are now embarked on a test and inspection program which will demonstrate the capability of the diesel generators at Catawba to perform their function. Under the current program, diesel generator 1-A, having completed a 750-hour run and an extensive inspection, is being reassembled and should be available for service again in July 1984. Diesel generator 1-B is undergoing a 750-hour run. Following completion of this run, diesel generator 1-B will be disassembled and inspected. Diesel generator 1-B should be available for service in September 1984. Under the current schedule then, during July 1984, neither of the Unit 1 diesel generators will be available for service. However, as was demonstrated in the

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Applicants' Motion and the NRC Staff response, even in the highly unlikely event of a loss of all offsite power, the diesel generators are not necessary for protection of public health and safety during conduct of the activities for which authorization is sought.

The safety functions which are to be performed by the Catawba diesel generators are described in GDC 17, which states in part:

> Electric power systems. An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

As noted in the NRC Staff response at 9,

> ...GDC-17 is concerned with assuring the safety of the plant during postulated accidents and anticipated operational occurrences upon normal critical power operation of the plant. The activities for which authorization is sought here -- fuel load and non-critical testing -- do not involve critical power operation.

On May 30, 1984, the Atomic Safety and Licensing Board issued a Memorandum and Order which authorized the Director of Nuclear Reactor Regulation, upon making findings on all applicable matters specified in 10 CFR §50.57(a), to issue to the Applicants a license to load fuel and conduct certain pre-critical testing at the Catawba facility.

In order for the NRC Staff to make such a determination under Section 50.57a, it is recognized that there must be a reasonable assurance of compliance with applicable regulations. Although GDC-17 calls for both onsite and offsite electrical power to be provided, and requires that both the offsite and onsite power system be capable of performing specified safety functions, assuming the absence of the other system in the event of certain specified accident conditions, this requirement must be read together with the provisions of 10 CFR 50.57, which contemplates a finding of whether there is reasonable assurance that the activities authorized by the operating license can be conducted without endangering the health and safety of the public and in compliance with the regulations.

In Shoreham, Commission stated that:

> In addressing the determinations to be made under 10 C.F.R. 50.12(a), the applicant should include a discussion of the following:
>
> > Its basis for concluding that, at the power levels for which it seeks authorization to operate, operation would be as safe as under the conditions proposed by it, as operation would have been with a fully qualified onsite A/C power source.

As described in the attached analysis (Attachment 1), a review of the safety functions described in GDC-17 shows that they are not required where operations do not create either significant fission products or decay heat. In Applicants view, then, compliance with the terms of GDC-17 for the activities sought is not required. Notwithstanding that fact, however, because Catawba Unit 1 will not have a qualified source of onsite AC electric power and will therefore not be in literal compliance with GDC-17, the Applicants request an exemption from the portion of GDC-17 requiring an onsite electric system.

The Commission in Shoreham also stated that an exemption request on Applicant should discuss:

> The 'exigent circumstances' that favor the granting of an exemption under 10 C.F.R. 50.12(a) should it be able to demonstrate that, in spite of its noncompliance with GDC 17, the health and safety of the public would be protected.

With respect thereto, Applicants Attachment 2 sets forth the current schedule which calls for commencement of fuel loading on June 29, 1984. It is obvious from the schedule set forth in Attachment 2, that a delay in loading fuel attributable to obtaining the requested exemption will result in substantial financial and/or economic hardship. Given this prospect, and the fact that all parties to the proceeding have stipulated to the issuance of a license authorizing the activities sought, it is clear that the public interest lies in granting the exemption.

Very truly yours,

Hal B. Tucker

Hal B. Tucker
Vice President
Nuclear Production

Attachments

cc: Mr. James P. O'Reilly
Regional Administrator
U.S. Nuclear Regulatory Commission

NRC Resident Inspector
Catawba Nuclear Station

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Carolina Environmental Study Group
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Mr. Robert Guild, Esq.
Attorney-at-Law
P.O. Box 12097
Charleston, South Carolina 29412

HAL B. TUCKER, being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this Amendment 31 to its application and documents appended thereto; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

Hal B. Tucker

Hal B. Tucker, Vice President

Subscribed and sworn to before me this 26th day of June, 1984.

Barbara J. Hawkins

Notary Public

My Commission Expires:

Barbara J. Hawkins
Notary Public, Montgomery County, Maryland
My Commission Expires July 1, 1986

ATTACHMENT 1

Safety Analysis Review

Catawba Unit 1

License to Load Fuel and Conduct Precritical Testing

1.0 INTRODUCTION

Duke Power Company is requesting a license to load fuel and conduct certain testing activities at Catawba Unit 1 prior to initial criticality. This, in support of that request, is a safety review of the Catawba Nuclear Station Final Safety Analysis Report, Chapter 15 transient and accident analyses. This review examines the Chapter 15 transients and accidents, within the plant conditions that are expected during fuel loading and the precritical testing activities. These plant conditions include no fission product inventory, a subcritical reactor, no decay heat, and a positive moderator temperature coefficient. The absence of a fission product inventory will preclude any environmental consequences and eliminate one of the primary concerns for all Chapter 15 transients and accidents. The boron concentration will be maintained high enough such that the reactor will remain subcritical even with all of the control rods out of the core. During fuel loading and precritical testing the reactor will not be a potential heat source. There will be no decay heat produced by the core, so the potential for DNB and a loss of core integrity are eliminated. With a subcritical reactor and no decay heat, reactor coolant is not required, and even a total loss of coolant will not affect core integrity. With the only heat sources being the reactor coolant pumps, the only credible temperature transient would be an overcooling transient. During fuel loading and precritical testing, the reactor will have a positive moderator temperature coefficient, so that a temperature decrease will only serve to increase the shutdown margin. The integral effect of these factors will be evaluated for each of the Chapter 15 events.

2.0 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

2.1 Feedwater System malfunction causing a reduction in feedwater temperature.

During normal operation steam is extracted from various stages in the turbine for use in preheating feedwater. During fuel loading and precritical testing, these extractions are not used to preheat the feedwater. Therefore, it will not be possible for a malfunction, such as the low pressure heater bypass analyzed in the Catawba FSAR, to cause a reduction in feedwater temperature.

2.2 Feedwater System malfunction causing an increase in feedwater flow

The primary concern for this transient, is a reduction of reactor coolant temperature potentially causing the reactor to increase to power. But since the moderator temperature coefficient is positive during fuel loading and precritical testing, this reduction will cause an increase in the shutdown margin. The continuous addition of excessive feedwater is prevented by the steam generator high-high level trip, which closes the feedwater valves. If offsite power is lost and the diesel generators are unavailable, feedwater flow will be terminated and no adverse consequences would occur.

2.3 Excessive increase in secondary steam flow

The primary concern for this transient is the potential for causing the reactor to increase to power. Any increase in steam flow will cause a decrease in reactor coolant temperature, and due to the positive moderator temperature coefficient will cause an increase in the shutdown margin. If offsite power is lost and the diesel generators are unavailable, no adverse consequences will occur.

2.4 Inadvertent opening of a steam generator relief or safety valve

The inadvertent opening of a relief or safety valve will depressurize the Main Steam System, and core conditions will be dominated by the rapid cooldown that will occur. Due to the positive moderator temperature coefficient, a decrease in reactor coolant temperature will cause an increase in the shutdown margin. If, during this transient, there was leakage from the primary side to the secondary side, no environmental consequences would result since there would be no fission product inventory to be released. If offsite power is lost and the diesel generators are unavailable, no adverse consequences will occur.

2.5 Steam System piping failure

A steam system piping failure will cause a rapid cooldown of the Reactor Coolant System. Due to the positive moderator temperature coefficient present during fuel loading and precritical testing, a decrease in reactor coolant temperature will cause an increase in the shutdown margin. If, during this transient, there was leakage from the primary side to the secondary side, no environmental consequences would result since there would be no fission product inventory to be released. If offsite power is lost and the diesel generators are unavailable, no adverse consequences will occur.

3.0 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

3.1 Steam pressure regulator malfunction

There are no pressure regulators in the Catawba plant whose failure or malfunction could cause a steam flow transient.

3.2 Loss of external load

During fuel loading and precritical testing, the turbine will not be on line, so a loss of external load is not possible.

3.3 Turbine trip

During fuel loading and precritical testing, the turbine will not be on line, so a turbine trip is not possible.

3.4 Inadvertent closure of the Main Steam Isolation valves (MSIV'S)

The inadvertent closure of a Main Steam isolation valve will prevent steam from being dumped into the main condenser from the corresponding steam generator. In the event that all MSIV'S close, the main steam safety and relief valves will be available to limit any transient. With no decay heat and the reactor subcritical, DNB concerns are eliminated as the heat sources in the Reactor Coolant System are the reactor coolant pumps. Reactor coolant flow is not required, during fuel loading and precritical testing. If there was leakage from the primary side to the secondary side, no environmental consequences would result since there will be no fission product inventory in the core. If offsite power is lost and the diesel generators are unavailable, the heat sources in the Reactor Coolant System will be eliminated and no adverse consequences will occur.

3.5 Loss of condenser vacuum and other events causing a turbine trip

During fuel loading and precritical testing, the turbine will not be on line, so a turbine trip is not possible.

3.6 Loss of non-emergency A-C power to the station auxiliaries

A complete loss of non-emergency A-C power may result in the loss of all power to the plant auxiliaries, such as the reactor coolant pumps and condensate pumps. This will not cause an increase in reactor coolant temperature, because without decay heat, the reactor coolant pumps are the primary heat source. With no decay heat and the reactor subcritical, DNB concerns are eliminated. The loss of offsite power when the diesel generators are unavailable would result in the loss of all power to the plant auxiliaries and would cause no adverse consequences to occur. Reactor coolant pump seal injection flow could be provided by the Standby Shutdown Facility to prevent reactor coolant pump seal damage. If the secondary safety and relief valves actuate and there was leakage from the primary side to the secondary side, there will be no environmental consequences since there will be no fission product inventory in the core.

3.7 Loss of normal feedwater

A complete loss of normal feedwater automatically starts the auxiliary feedwater pumps, and if sufficient feedwater is not supplied to the steam generators, will cause a reactor trip. The auxiliary feedwater pumps are more than capable of removing the heat generated by the reactor coolant pumps. With no decay heat and the reactor subcritical, DNB concerns are eliminated. This transient is of no consequence for the fuel loading and precritical testing since there are ample means of supplying sufficient feedwater, or the reactor coolant pumps can be turned off eliminating the need for feedwater. A loss of offsite power when the diesel generators are unavailable will eliminate the reactor coolant pumps as a heat source and as a result no adverse consequences will occur.

3.8 Feedwater System pipe break

A major feedwater line rupture is a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell side fluid inventory in the steam generators. With no decay heat and the reactor subcritical, DNB concerns are eliminated as reactor coolant flow is not required. With the heat source during fuel loading and precritical testing being the reactor coolant pumps, both normal and auxiliary feedwater will be capable of supplying sufficient feedwater. So this transient is of no consequence. If there was leakage from the primary side to the secondary side, there would be no environmental consequences since there will be no fission product inventory in the core. If offsite power is lost and the diesel generators are unavailable, no adverse consequences will occur.

4.0 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

4.1 Partial loss of forced reactor coolant flow

In the absence of decay heat and with the reactor subcritical, DNB concerns are eliminated. No reactor coolant flow is necessary to prevent a reactor coolant temperature increase as the only heat sources are the reactor coolant pumps, and a decrease in forced flow is a decrease in the heat source. If offsite power is lost and the diesel generators are unavailable, a complete loss of forced reactor coolant flow will occur and is discussed in Section 4.2. Therefore, this transient is of no consequence during fuel loading and precritical testing.

4.2 Complete loss of forced reactor coolant flow

A complete loss of forced flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. In the case of no decay heat and with the reactor subcritical, DNB concerns are eliminated, and no reactor coolant flow is necessary to prevent a reactor coolant temperature increase. With the loss of the reactor coolant pumps, the heat source in the reactor coolant flow is removed. During fuel loading and precritical testing reactor coolant flow is not required. If offsite power is lost and the diesel generators are unavailable, no adverse consequences will occur. Therefore, this transient is of no consequence.

4.3 Reactor Coolant pump shaft seizure

This transient is expected to cause a faster decrease in Reactor Coolant System flowrate than a partial or complete loss of forced reactor coolant flow, due to the elimination of the pump coastdown. In the absence of decay heat and with the reactor subcritical, DNB concerns are eliminated, and no reactor coolant flow is necessary to prevent a reactor coolant temperature increase. Without decay heat, reactor coolant flow is not required. With the loss of a reactor coolant pump, the heat source in the reactor coolant system is reduced, as the reactor coolant pumps are the only heat source during fuel loading and precritical testing. If offsite power is lost and the diesel generators are unavailable, a complete loss of forced reactor coolant flow will occur and is discussed in Section 4.2. Therefore this transient is of no consequence.

4.4 Reactor coolant pump shaft break

The accident is postulated as an instantaneous failure of a reactor coolant pump shaft. Flow through the affected reactor coolant loop is rapidly reduced, but not as rapidly as in the reactor coolant pump shaft seizure event. This transient is otherwise the same as that discussed in Section 4.3 for fuel loading and precritical testing.

5.0 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

5.1 Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition.

During fuel loading and precritical testing the boron concentration in the Reactor Coolant System will be maintained high enough to keep the reactor subcritical even with all control rods out of the core. Therefore, if the RCCA's were to be withdrawn in an uncontrolled manner, the reactor will remain subcritical, eliminating any potential for this type of transient.

5.2 Uncontrolled rod cluster control assembly bank withdrawal at power

During fuel loading and precritical testing the reactor will not be operated at power, so this type of transient cannot occur.

5.3 Rod cluster control assembly misoperations

RCCA misoperation accidents include one or more dropped RCCA's within the same group, a dropped RCCA bank, a statically misaligned RCCA, and the withdrawal of a single RCCA. During fuel loading and precritical testing the boron concentration will be maintained high enough to keep the reactor subcritical with all of the control rods out of the core. A loss of offsite power with the diesel generators unavailable, will cause the RCCA's to drop into the core. Therefore, none of these accidents will challenge the required shutdown margin or cause undesireable power distributions.

5.4 Startup of an inactive reactor coolant pump at an incorrect temperature

During fuel loading and precritical testing, the core ΔT will be zero and the loop ΔT will be small as the heat source will be the reactor coolant pumps. The primary concern is that the startup of a reactor coolant pump in an inactive loop will send a slug of cooler water into the core. However, with the temperature differentials that will exist during fuel loading and precritical testing, this transient will be of no consequence. If offsite power is lost and the diesel generators are unavailable, this transient cannot occur.

5.5 A malfunction or failure of the flow controller in a BWR Loop that results in an increased reactor coolant flowrate

This transient does not apply to Catawba.

5.6 Chemical and Volume Control System malfunction that results in a decrease in boron concentration in the reactor coolant

A boron dilution event during fuel loading and precritical testing is very unlikely for the following reasons:

1) A dilution accident would require multiple manual operator actions to initiate it.

2) Reaching criticality would require that the operator take no action to stop the dilution during the time that shutdown margin is being reduced.

3) The boron concentration will not have to be changed until initial criticality, so a dilution would be an unusual event and therefore even more unlikely.

Even assuming that a dilution event occurred, the Reactor Coolant System would initially be at a conservatively high boron concentration. This ensures that the operator would have ample time to mitigate any postulated boron dilution accident before the shutdown margin was completely lost. It is very unlikely that a loss of offsite power would prevent the operator from taking action to restore adequate shutdown margin since such a loss of power would terminate the dilution process. Therefore, a loss of offsite power would have to selectively happen many minutes after the initiation of the event, i.e., immediately after criticality, to prevent mitigation of the dilution.

5.7 Inadvertent loading and operation of a fuel assembly in an improper position

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. These procedures assure that the fuel assembly will be identified several times and check that the fuel assembly will be placed in its proper position. After all fuel loading has been completed, a thorough inspection is conducted to assure that all fuel assemblies and control components are located in their proper position.

5.8 Spectrum of rod cluster control assembly ejection accidents

During fuel loading and precritical testing, the boron concentration in the Reactor Coolant System will be maintained such that the reactor will remain subcritical even with all of the control rods out of the core. Therefore, the rupture of a RCCA drive mechanism and subsequent ejection of the associated RCCA will not challenge the shutdown margin.

6.0 INCREASE IN REACTOR COOLANT

6.1 Inadvertent operation of Emergency Core Cooling System during power operation

The primary concern in this transient is a reduction in core power and resultant power mismatch that causes a reduction in Tavg and shrinkage of the reactor coolant. During fuel loading and precritical testing the reactor will not be in power operation so the only effect that an inadvertent ECCS actuation could have is an increase in reactor coolant inventory with cold, highly borated water. This could potentially cause the pressurizer relief valves to lift. There will be no environmental consequences since there will be no fission product inventory in the core. If offsite power is lost and the diesel generators are unavailable, inadvertent ECCS actuation would be precluded or terminated and no adverse consequences would occur.

6.2 Chemical and Volume Control System malfunction that increase reactor coolant inventory

An increase in reactor coolant inventory by the addition of cold, unborated water to the Reactor Coolant System is analyzed in Section 5.6. An increase in reactor coolant inventory by the addition of highly borated water to the Reactor Coolant System is analyzed in Section 6.1.

7.0 DECREASE IN REACTOR COOLANT INVENTORY

7.1 Inadvertent opening of a pressurizer safety or relief valve

The loss of reactor coolant will serve to cool the Reactor Coolant System and due to the positive moderator temperature coefficient, increase the shutdown margin. With no decay heat and the reactor subcritical, the reactor coolant is not required, so the loss thereof is of no consequence. There will be no environmental consequences since there will be no fission product inventory in the core. If offsite power is lost and the diesel generators are unavailable, no adverse consequences will occur.

7.2 Break in instrument line or other lines from reactor coolant pressure boundary that penetrate Containment

Once detected, these breaks are all isolatable. The most severe pipe rupture is a complete severance of the letdown line just outside the Containment building. This break would result in a loss of reactor coolant at a rate no greater than 140 gpm. This release rate is within the capability of the normal reactor makeup system. The primary concern with this type of transient is the environmental consequences posed by the escaping reactor coolant. There will be no environmental consequences since there will be no fission product inventory in the core. If offsite power is lost and the diesel generators are unavailable, no adverse consequences will occur.

7.3 Steam generator tube rupture

The primary concern involved with steam generator tube rupture is the potential for environmental consequences caused by reactor coolant going through the rupture and escaping to the atmosphere. The loss of coolant is of no consequence because with no decay heat and the reactor subcritical, the reactor coolant is not required. There will be no environmental consequences since there will be no fission product inventory in the core. If offsite power is lost and the diesel generators are unavailable, no adverse consequences will occur.

7.4 BWR piping outside Containment

This section does not apply to Catawba.

7.5 Loss of coolant accidents

A loss of coolant accident is the result of a pipe rupture of the reactor coolant system pressure boundary. The primary concerns for LOCA's are the peak clad temperature and environmental consequences. During fuel loading and precritical testing there will be no decay heat, eliminating the peak clad temperature as a concern. Reactor coolant will not be required and even a total loss of coolant will not affect core integrity. There will be no environmental consequences since there will be no fission product inventory in the core. If offsite power is lost and the diesel generators are unavailable, no adverse consequences will occur.

8.0 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

There will be no environmental consequences since there will no fission product inventory in the core.

9.0 ANTICIPATED TRANSIENTS WITHOUT SCRAM

The concern in ATWS transients is that a reactor continues to operate at some power level even after a trip setpoint has been exceeded. During initial fuel loading and precritical testing, the reactor will be subcritical even with all of the control rods out of the core. Therefore, continued power operation following a reactor trip signal is not possible.

10.0 LOSS OF STATION POWER

During fuel loading and precritical testing, there will be a period of time during which both diesel generators will be disassembled and undergoing inspection. Without decay heat and with the reactor subcritical, with the loss of station power there will be no heat source in the Reactor Coolant System. The plant could remain in this condition indefinitely with the only potential for damage being to the pumps seals. Damage to the pump seals could occur if the temperature in the Reactor Coolant System were high enough, and a loss of seal injection and component cooling water would occur. If such cooling water is lost for an extended period of time, a loss of coolant accident may occur, but the probability of such an occurrence happening is extremely small. Seal injection flow to the reactor coolant pumps can be provided by the Standby Shutdown Facility. The turbine driven auxiliary feedwater will be available to supply feedwater in the event of a total loss of station power. LOCA's and their consequences are addressed in Section 7.5.

11.0 SUMMARY

Each of the Chapter 15 transients and accidents from the Catawba FSAR has been reviewed with respect to the plant conditions that are anticipated during fuel loading and precritical testing. The primary concerns that dominate transient and accident analyses, decay heat and fission product inventory, will not be present, during fuel loading and precitical testing, and without a fission product inventory the potential for any environmental consequences is eliminated. The reactor will be subcritical at all times and not even the removal of all control rods will bring the reactor to power. Without any heat source in the core, reactor coolant is not required so the loss of reactor coolant is of no consequence. The absence of a heat source also mitigates the potential for an overheating type of transient. The only credible temperature transient is an overcooling of the Reactor Coolant System, but with a positive moderator temperature coefficient, the decrease in temperature will increase the shutdown margin.

In summary, a review of Chapter 15 of the FSAR based on the core conditions that will be maintained during the Catawba Unit 1 precritical testing program has been performed. The consequences of all transients and accidents have been determined to be insignificant, and will result in no risk to the health and safety of the public.

12.0 REFERENCES

1. Catawba Nuclear Station FSAR Chapter 15.

2. WCAP-10422 The Nuclear Design and Core Physics Characteristics of the Catawba Unit 1 Nuclear Power Plant Cycle 1.

3. TP/1/A/2650/01 Initial Fuel Loading Procedure Catawba Unit 1.

ATTACHMENT 2

JUN 20 1984

DUKE POWER COMPANY

LEGAL DEPARTMENT
P. O. Box 33189
CHARLOTTE, N. C. 28242

(704) 373-2570

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June 18, 1984

James L. Kelley, Chairman
Atomic Safety and Licensing Board Panel
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. Paul W. Purdom
235 Columbia Drive
Decatur, Georgia 30030

Dr. Richard F. Foster
P. O. Box 4263
Sunriver, Oregon 97702

Re: Duke Power Company, et al.
(Catawba Nuclear Station, Units 1 and 2)
Docket Nos. 50-413, 50-414

Gentlemen:

Attached is an affidavit of Warren H. Owen which reports a change in the previously-scheduled fuel load date. The prior date was June 16, 1984. The new date for start of fuel loading is now June 29, 1984.

Sincerely,

AVCarr Jr

Albert V. Carr, Jr.

AVCjr/dm
c: (w/enclosures)
Judges Margulies, Lazo, Hooper
All Parties

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
DUKE POWER COMPANY, et al.	)	Docket No. 50-413
	)	50-414
(Catawba Nuclear Station	)	
Units 1 and 2)	)	

AFFIDAVIT OF WARREN H. OWEN

My name is Warren H. Owen. I am currently employed by the Duke Power Company as Executive Vice President, Engineering and Construction. I am a member of the Board of Directors and of the Executive Committee. In these capacities I am responsible for, among other things, scheduling of the fuel loading, testing and power ascension phases for the Catawba nuclear units. Schedules have been developed for these phases which take into account Duke's experience with such activities at its five operating nuclear reactors.

The purpose of this affidavit is to inform the Licensing Board of a change in our fuel load date, and to explain briefly the reasons for such change.

Our current schedule for Catawba Unit 1 reflects the following:

Fuel Loading	June 29 thru July 9
Pre-Critical Testing	July 9, 1984 thru Sept. 14, 1984
0-5% power testing (plant critical)	Sept. 14, 1984 thru Sept. 27, 1984
5-100% power testing	Sept. 27, 1984 thru Feb. 6, 1985

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Our previously scheduled date for starting fuel loading was June 16, 1984. As of this date, all construction on all systems necessary for fuel loading and pre-critical testing is complete. However, a significant number of systems required for start of fuel loading have only recently been turned over to the Nuclear Production Department by the Construction Department. There remains, for some of those systems, certain functional and surveillance testing to be done, as well as completion of a volume of paperwork. Because of the tests which remain to be done, and the volume of paperwork which remains to be completed, we believe it prudent to move our target date for start of fuel loading from June 16 to June 29, 1984.

We feel confident that we can complete the work outlined above and, barring unforeseen events, will meet our projected date of June 29 for start of fuel loading.

I, Warren H. Owen, of lawful age, being first duly sworn, state that I have reviewed the foregoing affidavit and that the statements contained therein are true and correct to the best of my knowledge and belief.

Warren H. Owen

Warren H. Owen

Sworn to and subscribed before me
this 18th day of June, 1984.

Gail I. Christenbury
Notary Public
Commission Expires 7-28-86