UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
DUKE POWER COMPANY, et al.	)	Docket Nos. 50-413
	)	50-414
(Catawba Nuclear Station, Units 1 and 2)	)	

AFFIDAVIT OF P. M. ABRAHAM
AND WILLIAM R. McCOLLUM, JR.

P. M. Abraham and William R. McCollum, Jr., being duly sworn, depose and state as follows:

(1) My name is P. M. Abraham. My business address is Duke Power Company, Nuclear Production Department, P. O. Box 33189, Charlotte, N.C., 28242. I am the supervisor of the Reactor Safety Section in the Nuclear Production Department of Duke Power Company. In this capacity I direct the nuclear safety efforts involving safety analysis, nuclear safety event analysis, and probabilistic risk assessment within the Nuclear Production Department. Prior to becoming the supervisor of Reactor Safety, in 1980, I was assigned to the Licensing Unit with the responsibility of performing the review and analysis of reactor safety matters involving transient and accident analysis, plant startup testing, core design and generic safety issues. I have been employed by Duke Power Company since July 1974.

(2) From 1965 through 1966, I was a lecturer in physics at St. Thomas College in Kerala, India, and from 1970 to 1972 I was

employed as an assistant professor at Belmont Abbey College in North Carolina teaching physics and mathematics.

(3) I have both a Bachelors and a Masters Degree in Physics from the University of Kerala (1963 and 1965, respectively), a Masters Degree in Nuclear Engineering from North Carolina State University (1974), and a Doctorate Degree in Nuclear Physics from the University of Colorado (1970).

(4) I currently serve on the Duke Power Company Corporate Research and Development Committee and the American Nuclear Society Reactor Safety Division Program Committee.

(5) My name is William R. McCollum, Jr. My business address is Duke Power Company, Catawba Nuclear Station, P.O. Box 256, Clover, S.C. 29710. I am the Catawba Unit 1 Schedule Engineer. In this capacity I am cognizant of the activities involved in fuel loading and precritical testing. Prior to assuming my present position in February, 1984 I held the position of Performance Engineer in the Technical Services Group at Catawba. In this capacity I was responsible for the development and execution of the preoperational and startup testing programs at Catawba which included development and execution of such programs for Catawba's diesel generators. I have been employed by Duke Power Company since September, 1974.

(6) I have a Bachelors Degree in Electrical Engineering and a Masters Degree in Nuclear Engineering from Georgia Institute of Technology (1973 and 1974, respectively).

(7) The purpose of this joint affidavit is to describe the steps involved in fuel loading and precritical testing, to

describe the safety implications of these activities, and to demonstrate that, for these activities availability of onsite electrical power as required by General Design Criterion 17 is not necessary.

(8) During the initial fuel loading, 193 cold, clean (unirradiated) fuel assemblies and specified control components are loaded into the reactor vessel in accordance with a written, approved procedure. An initial nucleus of eight fuel assemblies, the first of which contains an activated neutron source, is loaded into the reactor vessel to permit meaningful inverse count-rate monitoring of additional fuel insertions. This initial nucleus is determined by calculation and previous experience to be markedly subcritical ($K_{eff} < .95$) under the required conditions of fuel loading. Each subsequent fuel addition is accompanied by detailed neutron count rate monitoring to determine that the just-loaded fuel assembly does not excessively increase the count rate and that the extrapolated inverse neutron count rate is not decreasing for unexplained reasons.

(9) During core loading at least one path for boron addition to the Reactor Coolant System is available at all times. Uniform boron concentration in the Reactor Coolant System is maintained by circulation with at least one residual heat removal pump as required by Technical Specifications and is sufficient to assure K_{eff} less than or equal to 0.95 during fuel loading. Containment integrity is established and maintained in accordance with the Technical Specifications.

(10) Upon completion of core loading, the reactor upper internals and the reactor vessel head are installed. At this time the unit is ready for testing in successively higher modes. The Reactor Coolant System is heated up as necessary using heat from the reactor coolant pumps. Additional mechanical and electrical tests are performed, including the following tests which are discussed below and more fully described in Table 14.2.12-2 of the Catawba FSAR:

a) Movable Incore Detector Functional Test

The purpose of this test is to assure proper alignment and operation of the movable incore detector drive system and readout equipment.

The system is operated manually and automatically in all modes after setting the indexing and limit switches. The response of each channel to simulated detector movement is verified.

b) Incore Thermocouple and RTD Cross - Calibration (Optional)

The purpose of this test is to determine the response characteristics of each RTD and the response characteristics and isothermal correction factor for each incore thermocouple and to demonstrate the proper operation of temperature readout and compensating equipment. This test was preformed initially during hot functional testing, and is repeated anytime RTDs are replaced following hot functional testing.

As the unit is heated up from cold shutdown, isothermal conditions are established at selected intervals. At

these isothermal plateaus resistance versus temperature is measured and plotted for all RTDs and the variation between each RTD and the average of the RTD readings is calculated and recorded. Also, the voltage versus temperature for each incore thermocouple is measured and plotted at each plateau to generate individual isothermal correction factors. The operation of remote and local temperature instrumentation is observed. Cold junction box temperatures are recorded at each plateau.

c) Rod Position Indication Check

The purpose of this test is to verify that the Digital Rod Position Indication System satisfactorily performs the required indication and alarm functions for each individual rod under hot shutdown conditions, and to demonstrate that all full length rods operate satisfactorily over their entire range of travel.

Each full length rod cluster control assembly is pulled to its fully withdrawn position and inserted to its fully inserted position in discrete increments. Indication and alarms are observed for proper operation.

d) Rod Cluster Control Assembly Drop Time Test

The purpose of this test is to verify the drop time for each full-length rod cluster control assembly under no-flow and full-flow conditions, with the reactor in the cold shutdown and in hot standby conditions.

Each rod cluster control assembly for each unit condition is individually withdrawn, then the drop time is determined by monitoring the rod position indication signal

following deenergization of the stationary winding of the rod cluster control assembly drive mechanism.

e) <u>Rod Control System Alignment Test</u>

The purpose of this test in the cold shutdown condition is to assure proper connection, identification and continuity of Rod Control System power and control cabling. The purpose in the hot standby condition is to adjust Rod Control System bank-overlap setpoints and to demonstrate proper system control and indication and to verify control rod withdrawal interlocks.

With the reactor in the cold shutdown condition, the connection and identification of each power and control cable are visually checked and the resistance of each measured. With the reactor in the hot standby condition, the Rod Control System is operated in various modes and indications and alarms observed. Bank start and stop positions during insertion, withdrawal and overlap operations are recorded. Simulated control rod inhibit signals are injected into the control system and the rod withdrawal interlocks verified. Setpoint adjustments are made as required.

f) <u>Rod Drive Mechanism Timing Test</u>

The purpose of this test is to demonstrate proper operation and timing of each rod drive mechanism.

With the reactor in the cold shutdown condition, the timing for each slave cycler is set, measured and reset as necessary. Each rod drive mechanism is manually operated with a rod cluster control assembly attached, checking the latching and

releasing features of each. The test is repeated for each rod drive mechanism with the reactor in the hot standby condition.

g) Reactor Coolant System Flow Test

The purpose of this test is to verify predicted Reactor Coolant System flow rates at normal no-load operating temperature and pressure and to align the Reactor Coolant System flow instruments.

The output voltage of each NC loop differential pressure transmitter is measured using a digital voltmeter. The output voltages are averaged and converted on equivalent differential pressure which is then converted to flow using a vendor supplied, plant specific graph. The loop flows are summed to give the total system flow. The flow transmitters are adjusted for 100 percent flow at normal operating conditions and zero output at zero flow.

h) Reactor Coolant System Flow Coastdown Test

The purpose of this test is to measure the rate at which reactor coolant flow rate decreases, subsequent to reactor coolant pump trips, from various flow configurations in order to verify the assumptions made in the analysis of the loss of flow accident.

One reactor coolant pump is tripped and flow coast down data is recorded. As a separate portion of the tests, all reactor coolant pumps are tripped simultaneously. On a high-speed strip chart recorder, for each transient, one elbow tap differential pressure cell for each loop, and each reactor coolant pump breaker position.

i) <u>RTD Bypass Flow Verification</u>

The purpose of this test is to determine the flowrate necessary to achieve the required reactor coolant transport time in each RTD bypass loop (time from NC loop to last RTD well), to verify that the coolant transport times are acceptable and to verify the low flow alarm setpoint and reset for the total RTD bypass flow in each reactor coolant loop.

The flow required to achieve the required reactor coolant transport time is determined by accurately measuring and recording the lengths of installed piping from the bypass loop inlet connections on each reactor coolant loop to the last downstream RTD of both the cold and hot leg bypass loops, and then calculating the flow necessary to achieve less than or equal to 1.0 second transport time. The total bypass flowrate is then measured with both loops in service, and the actual bypass loop transport time is calculated. The low flow alarm setpoint and reset are verified by sequentially throttling the hot and cold leg manifold isolation valves in each loop and noting the flow when the alarm point(s) are reached.

j) <u>Pressurizer Functional Test</u>

The purpose of this test is to establish the continuous spray flow rate, determine the effectiveness of the pressurizer normal control spray and of the pressurizer heaters, and verify the response time of the pressurizer power operated relief valves.

While maintaining pressurizer level constant, spray bypass valves are adjusted until a minimum flow is achieved which

maintains less than a 125°F temperature difference between the spray line and the pressurizer steam space.

To determine pressurizer heater and spray capability, all pressurizer spray valves are closed. All pressurizer heaters are then energized and the time to reach a 2300 psig system pressure is measured and recorded. Bypass spray valves are then returned to their previously determined setting and full spray is initiated through each spray valve individually and in parallel. Pressure versus time is recorded for each transient. The transient is terminated at a Reactor Coolant System pressure of approximately 2000 psig by shutting the spray valves.

With the Unit at normal operating no load temperature and pressure, each PORV is cycled for response time testing. The 2185 psig interlock closes the valve and original conditions are re-established.

This test is performed following initial fuel loading due to the need to establish the effectiveness of actual spray flow with core pressure drop acting as the driving head.

(11) During fuel loading and precritical testing activities the concentration of boron within the reactor coolant system will be maintained at a level sufficient to ensure that subcriticality is maintained even in the unlikely event all control rods are withdrawn from the core. This level will be in compliance with that required by the facility Technical Specifications. The concentration of boron will be checked at least once per day in order to ensure that the level is being maintained. In the event

that boron concentrations decreased, the operator would be alerted, during fuel movement by a change in the monitored source range count rate, and at all other times by the source range channel alarm. The operator would then take positive action to borate the reactor coolant system using the available boron addition flow path which will be available in compliance with Technical Specifications.

(12) The potential impact of a nuclear power plant on the public health and safety, whether appreciable or insignificant, is assessed in terms of the potential for accidents which could result in the release of significant quantities of radioactive fission products. In other words, the public health and safety risk associated with a nuclear power plant is characterized in terms of the likelihood of accidents involving plant systems which contain large quantities of radioactive fission products and the amount of fission products which are actually released as a result of such accidents. In a typical operating nuclear power plant fission products are contained in radwaste systems, spent fuel assemblies outside the reactor core, and the reactor core itself. Since the fission products are the byproducts of the fission process taking place in the reactor core, the inventory of radioactive fission products contained in these systems would vary from zero for a plant which has not attained initial criticality to an equilibrium value for a plant which has been in full power operation for a long period of time.

(13) Fuel loading and precritical test activities involve no sustained fission reactions in the facility, and therefore, no

fission products such as would follow criticality are generated at the facility. Consequently, these activities pose no impact on the public health and safety.

(14) There is, yet, another aspect of the proposed activities which further makes any concern on the public health and safety moot. This is the amount of core decay heat available in the core after reactor shutdown following any postulated accident. The principal mechanism for significant release of fission products for the reactor core during postulated accidents is core damage from inadequate decay heat removal. The amount of core decay heat is a function of the amount of fission products in the core. During the fuel loading and precritical test activities decay heat is non-existent.

(15) Chapter 15 of the Catawba FSAR contains the safety analysis of transients and accidents to demonstrate that the operation of the plant at its full power level does not pose any undue radiological consequences to the public. In addition to the loss of non-emergency power event, certain design basis accidents (steamline break, feedwater line break, and loss of coolant) analyzed in the FSAR involve consideration of a loss of offsite power. For these accidents the safety analysis assumed that one diesel generator would be available to provide power to one train of the emergency AC power system. Under the situation postulated AC power is considered essential in order to remove core decay heat and to prevent any significant release of fission products to the environment when considering accidents with reactor operation at high power levels.

(16) Fission products of concern are generated in the reactor core as a result of sustained fission reaction in the core; the amount of fission product accumulated in the core is proportional to the power level of the reactor. If the fission product inventory in the core is large, as is the case after continuous operation of the reactor at high power levels for a period of time, the decay heat associated with these fission products would have to be removed in order to prevent excessive breakup of the fuel rods, thereby creating the potential for release of the fission products from the fuel rods into the reactor coolant. Under these circumstances, should one of the accidents postulated in Chapter 15 of the FSAR occur, the onsite source of electric power to be provided by the Catawba diesel generators might be needed to meet the requirements of General Design Criterion 17, which states in pertinent part:

> An onsite electric power system . . .shall be provided . . . 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.

10 CFR Part 50, Appendix A.

(17) However, for conducting the activities described in ¶¶ 8 - 11 of this Affidavit, and for which permission is sought, the onsite AC power to be provided by the Catawba diesel generators is not necessary. As noted therein, the initial fuel loading and the pre-critical tests are activities performed under subcritical conditions. They involve no sustained fission

reactions in the reactor, and therefore no fission products are generated in the reactor core. Accordingly, for postulated accidents there is no need for the ability to remove decay heat and there is no release of fission products. Therefore, the initial fuel loading and pre-critical tests activities pose no threat to the health and safety of the public even if the diesel generators are unavailable during these activities.

P. M. Abraham

P. M. Abraham

William R. McCollum, Jr.

William R. McCollum, Jr.

Subscribed and sworn to
before me this 11
day of April 1984.

Jo Ann D. Bowman

Notary Public

My Commission expires: 7-12-88