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July 3, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Re: Catawba Nuclear Station
Docket Nos. 50-413 and 50-414

Dear Mr. Denton:

On June 27, 1984, Duke Power Company filed Revision 11 to the Catawba FSAR. As a result of NRC Staff review of this revision and as a result of preoperational testing at Catawba a number of additional revisions to the FSAR have been identified. The following is a brief description of each:

1. Table 1.9-1 (Page 17) item III.D.1.1 was revised to add the Refueling Water System as requested by Ms. E. G. Adensam's letter of June 18, 1984.

2. Section 9.1.4.3.1 (page 9.1-27) was revised to accurately describe load testing of a portion of the fuel handling system.

3. Section 10.4.9 (page 10.4-27 and Tables 10.4.9-1 and -2) was revised to correct a typographical error and to clarify the design flow rates of the auxiliary feedwater pumps.

4. Section 11.2.2.7.2.2 was revised to reflect an improved method of detecting identified leakage inside containment. The revised method has been verified by preoperational testing.

5. Tables 12.3.3-1, -2, -3, -4, and -6 were revised to take exception to the Regulatory Guide 1.52, Rev. 2 requirement which states "After the test is completed, air flow through the unit should be maintained until the residual refrigerant gas in the effluent is less than 0.01 ppm". This is not considered necessary for the following reasons:

   a) R-11 is normally used to challenge carbon because it is not a poison to the carbon. R-11 is not chemically adsorbed, but only lightly physically adsorbed.

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b) The R-11 adsorbed during a test will not in any way decrease the ability of the carbon to adsorb radioiodides.

c) ANSI N510-1980 does not require verification of refrigerant concentration after testing and this guide was published after Reg. Guide 1.52.

The only possible concern for residual refrigerant is for future filter testing, that the carbon bed does not have too much R-11 already present to be able to test. In lieu of verification of 0.01 PPM, steps will be put in filter testing procedures to run the system with preheaters energized for at least 10 hours following the test to drive off any excess refrigerant.

Revised FSAR pages reflecting each of the above changes are attached. These pages will be included in Revision 12 to the FSAR.

Very truly yours,

Hal B. Tucker

H. B. Tucker

ROS/rhs

Attachment

cc: Mr. James P. O'Reilly, Regional Administrator
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NRC Resident Inspector
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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 these proposed FSAR revisions 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 3rd day of July, 1984.

Sue C. Sherrill

Notary Public

My Commission Expires:

Sept. 20, 1984

Westinghouse model is currently scheduled for submittal to the NRC by November 15, 1982 as documented in Letter NS-EPR-2617, dated June 28, 1982, E. P. Rahe (Westinghouse) to D. G. Eisenhut (NRC).

II.K.3.31 PLANT-SPECIFIC CALCULATIONS TO SHOW COMPLIANCE WITH 10 CFR 50.46

See II.K.3.30 above.

III.A.1.1 UPGRADE EMERGENCY PREPAREDNESS

See Section 13.3

III.A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES

See Section 13.3

III.D.1.1 PRIMARY COOLANT SOURCES OUTSIDE CONTAINMENT

A periodic leak rate test will be written for portions of systems which could carry highly radioactive fluids outside of containment following an accident. Portions of the following systems are included: Refueling Water, Safety Injection, Residual Heat Removal, Containment Spray, Containment Hydrogen Sample and Purge, Boron Recycle, Nuclear Sampling System, Chemical Volume and Control, Liquid Waste, and Waste Gas. This test, to be performed before startup and during each refueling outage, or at intervals not to exceed the refueling cycle, will be accomplished by pressurizing a system or part of a system and checking non-welded pipe joints, penetrations, flanges, valve separations, packing, and pump packing for leakage. Where possible, pumps included in the leak test boundary will be run so that a more accurate determination of the leak test may be made.

A separate periodic test procedure will be written to assure that excessive leakage is detected on a timely basis. This test will be run at least weekly and will require that systems carrying radioactive fluids outside of containment be visually inspected for excessive leakage. Appropriate corrective action will be taken if excessive leakage is detected.

III.D.3.3 IN-PLANT RADIATION MONITORING

See Section 12.5.3

III.D.3.4 CONTROL ROOM HABITABILITY

See Section 6.4

The hoist-gripper position interlock consists of two separate circuits that work in parallel such that one circuit must be closed for the hoist to operate. If one or both interlocking circuits fail in the closed position, an audible and visual alarm on the console is actuated. This interlock is provided in both the fuel and RCC hoist drive circuits.

f. An interlock of the bridge and trolley drives prevents the bridge drive from traveling beyond the edge of the core unless the trolley is aligned with the refueling crane centerline. The trolley drive is locked out when the bridge is beyond the edge of the core.

g. Suitable restraints are provided between the bridge and trolley structures and their respective rails to prevent derailing due to the safe shutdown earthquake. The refueling machine is designed to prevent disengagement of a fuel assembly or RCC element from the gripper under the safe shutdown earthquake.

h. The fuel, RCC, and auxiliary hoists are equipped with two independent braking systems. A solenoid release - spring set electric brake is mounted on the motor shaft. This brake operates in the normal manner to release upon application of current to the motor and set when current is interrupted. The second brake is a mechanically actuated load brake internal to the hoist gear box that sets if the load starts to overload the hoist. It is necessary to apply torque from the motor to raise or lower the load. In raising, the motor cams the brake open; in lowering, the motor slips the brake allowing the load to lower. This brake actuates upon loss of torque from the motor for any reason and is not dependent on any electrical circuits. On the fuel and RCC hoists, the motor brake is rated at 350 percent operating load and the mechanical brake at 300 percent.

The fuel hoist system is supplied with redundant paths of load support such that failure of any one component does not result in free fall of the fuel assembly. Two wire ropes are anchored to the winch drum and carried over independent sheaves to a load equalizing mechanism on the top of the gripper tube. In addition, supports for the sheaves and equalizing mechanism are backed up by passive restraints to pick up the load in the event of failure of this primary support. Each cable system is designed to support 13,750 pounds or 27,500 pounds acting together.

The working load of fuel assembly plus gripper is approximately 2500 pounds.

The gripper itself has four fingers gripping the fuel, any two of which will support the fuel assembly weight.

The gripper and hoist system are factory load tested to 6000 pounds and routinely load tested to 3000 pounds.

The RCC hoist system is the same as the fuel hoist except that the working load and routine test load are approximately one half as much.

stringent and conservative acceptance criteria for more probable events such as loss of normal feedwater or station blackout than for more unlikely events such as fire, sabotage, control room evacuation, loss of all A.C. power, feedline break. For standard plant 412, Westinghouse has determined minimum CA flow requirements of 600 GPM @ 120°F for loss of normal feedwater and 470 GPM @ 120°F for more severe events or for plant cooldown following a period of hot standby. Maximum CA temperature at Catawba may reach 138°F based on maximum operating condenser pressure of 24.3 inches Hg vacuum. The Westinghouse requirements based on 120°F are adjusted to the basis of 138°F. Based on a supply temperature of 138°F, minimum CA flow requirements are 613 GPM for loss of normal feedwater and 480 GPM for more severe events or for plant cooldown following a period of hot standby.

Standards for nuclear safety related systems are met for the CA System except for the condensate quality feedwater sources. The nuclear safety related portion of the CA System is designed for seismic and single failure requirements. The CA System will provide the required flow to two or more steam generators regardless of any single active or passive failure in the long term. Safety classifications of the Auxiliary Feedwater System components are presented in Table 3.2.2-2.

The use of redundancy, diversity, and separation has been incorporated into the design of the CA System to ensure its capability to function. Redundancy is provided by using two full capacity motor driven pumps and one full capacity turbine driven pump. Diversity is provided by using several water sources, two types of pump drivers, and adequate valving for source selection, isolation, and cross-connection. Separation is provided with separated power, instrumentation, and control subsystems with appropriate measures precluding interaction between subsystems. Independent piping subsystems are incorporated into the design and protected at interconnection points with appropriate isolation and/or check valves. All of the necessary instrumentation, controls, and valves for the motor driven CA pumps are powered by the train of emergency A.C. electrical power associated with each pump. The controls for the turbine driven pump are also powered by a third emergency D.C. electrical power supply. Separation, diversity, and redundancy are provided throughout the design of the CA System to allow the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of offsite electrical power.

For the postulated non-seismic event of loss of all offsite and all onsite emergency A.C. electrical power, the CA System will perform its safety related function with the limitation that no single failure that would prevent the single A.C. power independent turbine driven pump subsystem from functioning occurs during this limiting event.

Design features and operational precautions are provided to preclude the possibility of hydraulic instability (water hammer) in both the CA System and the Condensate/Feedwater System during all anticipated operating transients. The conditions necessary to produce water hammer in the main feedwater piping and/or steam generators must occur simultaneously as either low S/G temperature and extremely low S/G level (below the level which initiates the CA System) or low S/G temperature and low S/G pressure. Although piping and

Table 10.4.9-1
Auxiliary Feedwater System
Motor Driven Pump Design Data

Quantity per Unit	2
Type	Centrifugal, Horizontal
Fluid	Water
Design temperature, °F	160
Design flow rate, GPM	500
Design head, ft. H_2O	3210
NPSH requried, at design flow, ft.	15
Rated RPM	3600

Driver:

Type	Direct coupled, electric motor
Rated BHP	600
Rated RPM	3600
Service Factor	1.25
Power Requirements	4000 VAC, 3 Phase, 60 Hz

Table 10.4.9-2
Auxiliary Feedwater System
Turbine Driven Pump Design Data

Quantity per Unit	1
Type	Centrifugal, Horizontal
Fluid	Water
Design temperature, °F	160
Design flow rate, GPM	1000
Design head, ft. H_2O	3217
NPSH required, at design flow, ft.	15
Rated RPM	3600

Driver:

Type	Direct coupled, Single stage turbine
Rated BHP	1160
Rated RPM	3600
Steam inlet pressure, max/min, psig	1210 - 110
Back pressure, psig	3

The auxiliary feedwater pump turbine is qualified to Seismic Category I.

11.2.2.7.2 Faults of Moderate Frequency

11.2.2.7.2.1 Malfunction in the Liquid Radwaste System

Malfunction in this system could include such things as pump or valve failures or evaporator failure. Because of pump redundancy and standardization throughout the system, the backup pump can be started and spare pumps kept in stock can be used to replace most pumps in the system. There is sufficient surge capacity in the system to accommodate waste until repairs can be effected and normal plant operation resumed.

Should any normally non-radioactive sump or tank become contaminated, the radiation monitors provided will either terminate its discharge or divert to another normally radioactive tank for proper treatment.

11.2.2.7.2.2 Excessive Leakage in Reactor Building Equipment

The system is designed to handle a 1 gpm reactor coolant leak in addition to the expected leakage during normal operation. Operation of the system is almost the same as for normal operation except the load on the system is increased. A 1 gpm leak into the reactor coolant drain tank is handled automatically but will increase the load factor of the recycle evaporator. If the 1 gpm leak enters the WEFT, operation is the same as normal except for the increased load on the system. If this situation occurs, it is recommended that the evaporator bottoms be concentrated to 4% boric acid and recycled via the boric acid tanks. It will be necessary to ensure that the evaporator is clean and the boric acid concentrate is not contaminated. It may be necessary to drum the first batch in order to assure the unit's cleanliness. If a 1 gpm leak enters the floor drain tank via the containment floor and equipment sumps, the system will be operated the same as for a 1 gpm leak into the waste evaporator feed tank; i.e. both the evaporator condensate and concentrate will be recycled. If excessive leakage cannot be processed immediately and the WEFT and FDT become full, they may be pumped out to the SGDT Building for temporary holdup. When the evaporator is available, the SGDT contents can be processed directly to the evaporator.

The containment floor and equipment sump pumps as well as the incore instrumentation room sump pump, input to a plant computer program designed to detect one gpm of unidentified leakage inside containment in less than one hour as required by NRC Regulatory Guide 1.45. Since all reactor coolant pump seal leakoffs as well as other normally discharging fluids are routed to the RCDT, all liquid entering the floor sumps during station normal operation is unidentified leakage. In conjunction with the operator aid computer, sump level instrumentation monitors water level between the low and high setpoints and calculates rate of change. These values for both sumps are totaled and yield a computer alarm if the sum is greater than 1 gpm. While any of the sump pumps is running, leakrate is determined as a function of run time. The computer accumulates this time and provides an alarm if the leakrate is greater than 1 gpm. These arrangements will detect unidentified leakage in excess of 1 gpm within an hour. Incore instrumentation room sump pump is located under the reactor in the tunnel area where no leakage is expected. Therefore, an alarm is initiated should this pump ever start. A flow integrator is provided on the combined sump pump discharge for periodic recording.

The quantity and activity of VUCDT contents will also be an indicator of excessive reactor coolant leakage, as this condensate is normally clean, non-contaminated water. A sudden increase in the flow rate of ventilation condensate is an indicator of increased relative humidity in the containment. Such an increase of clean condensate in the absence of containment purge is unusual and is an indicator of leakage from a non-radioactive system, probably a steam leak. Increased radioactivity of ventilation condensate simultaneous with increased flow probably signals a reactor coolant leak. The VUCDT alarms on high level and also alarms if the VUCDT pump discharge is radioactive.

11.2.2.7.2.3 Excessive Leakage in Auxiliary Building Equipment

Excessive leakage from components and flanges in the Auxiliary Building enters the floor drain tank subsystem (see Section 11.2.2.1.5). Leakage in "radiation areas" flows directly to the FDT or indirectly to the FDT via floor drain sumps pumps A and B. The FDT overflows to the WEFT which overflows to the floor drain sumps, by which time the leaking component should be isolated. Sump pump and tank status help the operator detect the location of the leaking component and magnitude of the leakage.

Leakage in "non-radiation areas" flows to floor drain sumps C and D which automatically start on high level and discharge through a radiation monitor to the Turbine Building sump. If the leakage contains radioactivity, air operated valves divert the flow to the FDT. Sump pump status helps the operator detect the location and magnitude of the leakage.

Pipe trenches entering the Auxiliary Building are designed to prevent externally-caused flooding from entering the Auxiliary Building, as in the case of earthquake. Seismically designed piping and isolation valves serve to isolate drains and non-safety class lines and prevent them from becoming a leak path into the Auxiliary Building after such an earthquake.

Refer to Section 3.4.1 for a discussion of the safe water-holding capacities of various areas inside the Auxiliary Building.

11.2.2.7.3 Station Blackout

The system will not normally operate during a blackout. Only the containment spray and residual heat removal pump room sump pumps and auxiliary feedwater pump pit sump pumps are supplied with diesel power; these pump any leakage to

Table 12.3.3-1 (Page 3)

Comparison Of Control Room Area Pressurizing Air Filtration System With Regulatory Guide 1.52 Revision 2, March 1978

Paragraph	Compliance Status
C-4-c	In compliance
C-4-d	In compliance
C-4-e	In compliance
C-5-a	In compliance
C-5-b	In compliance
C-5-c	In compliance
C-5-d	In compliance, except that $\leq$ 0.01 PPM residual refrigerant will not be verified in the air stream following the challenge of the carbon adsorber bed. Instead, the system will be run with the preheaters energized for at least 10 hours following the test, to drive off any excess refrigerant.
C-6-a	In compliance
C-6-b	In compliance

Table 12.3.3-2 (Page 3)

Comparison Of Auxiliary Building Filtered Exhaust System With Regulatory Guide 1.52 Revision 2, March 1978

Paragraph	Compliance Status
C-4-c	In compliance
C-4-d	In compliance
C-4-e	In compliance
C-4-f	In compliance
C-5-a	In compliance
C-5-b	In compliance
C-5-c	In compliance
C-5-d	In compliance, except as noted on Table 12.3.3-1 (Page 3)
C-6-a	In compliance
C-6-b	In compliance

Table 12.3.3-3 (Page 3)

Comparison Of Fuel Handling Area
Exhaust Filtration System With Regulatory Guide 1.52
Revision 2, March 1978

Paragraph	Compliance Status
C-5-a	In compliance
C-5-b	In compliance
C-5-c	In compliance
C-5-d	In compliance, except as noted on Table 12.3.3-1 (Page 3)
C-6-a	In compliance
C-6-b	In compliance

Table 12.3.3-4 (Page 3)

Comparison Of Annulus Ventilation Filtration
System With Regulatory Guide 1.52
Revision 2, March 1978

Paragraph	Compliance Status
C-4-e	In compliance
C-5-a	In compliance
C-5-b	In compliance
C-5-c	In compliance
C-5-d	In compliance, except as noted on Table 12.3.3-1 (Page 3)
C-6-a	In compliance
C-6-b	In compliance

Table 12.3.3-6 (Page 3)

Comparison of Containment Purge Filter System
With Regulatory Guide 1.52
Revision 2, March 1978

Paragraph	Compliance Status
C-3-k	Adsorber section design includes a manual water spray system. Single-failure criterion is not considered in its design.
C-3-l	In Compliance
C-3-m	In Compliance
C-3-n	1980 version of ANSI N509 is followed.
C-3-o	In Compliance
C-3-p	In compliance with the following clarification: 1980 version of ANSI N509 is followed.
C-4-a	In Compliance
C-4-b	Since all filter banks are arranged for external servicing, three feet of spearation between filter banks is not necessary. Two and one half feet are provided for inspection purposes.
C-4-c	In Compliance
C-4-d	Containment purge system operation is Tech Spec limited.
C-4-e	In Compliance
C-5-a	In Compliance
C-5-b	In Compliance
C-5-c	In Compliance
C-5-d	In Compliance, except as noted on Table 12.3.3-1 (Page 3)
C-6-a	In Compliance
C-6-b	In Compliance