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1.1 Introduction

This Final Safety Analysis Report is submitted in support of Duke Power Company's application for Class 103 facility operating licenses for the two-unit Catawba Nuclear Station, located on the shore of Lake Wylie in York County, South Carolina. The station's location is shown on Duke's Service Area Map, Figure 1-1.

Each of the two essentially identical units employs a pressurized water reactor Nuclear Steam Supply System (NSSS) with four coolant loops which is furnished by Westinghouse Electric Corporation. These units are similar to those of the McGuire Nuclear Station. The turbine-generators for each unit are provided by the General Electric Company. Each Containment consists of a free-standing cylindrical steel structure enclosed by a separate reinforced concrete Reactor Building. The Containment and Reactor Buildings are designed by Duke.

Each generating unit is designed to operate at a reactor core full steady state level of 3469 MWt (Unit 1) and 3411 MWt (Unit 2), which corresponds to a net electrical output of about 1165 MWe (Unit 1) and 1145 MWe (Unit 2). All core physics and core-thermal-hydraulic information is based on the reference core design of 3469 MWt (Unit 1) and 3411 MWt (Unit 2). The Containment, Engineered Safety Features, and certain postulated accidents were originally evaluated by Westinghouse for a core rating of 3479 MWt.

Site preparation began under a Limited Work Authorization (LWA) on May 16, 1974, with Construction Permits granted on August 7, 1975. Fuel loading of Unit 1 began in July 1984 and of Unit 2 in February 1986. Unit 1 began commercial operation in June 1985 and Unit 2 in August 1986. A License Renewal Application was applied and accepted by the NRC, which allows an extended period of operation for Catawba Unit 1 and Unit 2 until December 5, 2043.

Duke Power Company is submitting this application for itself and as agent for the North Carolina Municipal Power Agency Number 1, Piedmont Municipal Power Agency, the North Carolina Electric Membership Corporation, and the Saluda River Electric Cooperative, Inc.

Duke is fully responsible for the complete safety and adequacy of the station. Consistent with long-standing practice, Duke personnel design, construct, perform quality assurance for, test, startup and operate the units. Assistance in performing these functions is rendered principally by Westinghouse, along with other consultants and suppliers as may be required.

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1.2 General Station Description

1.2.1 Site Characteristics

Site features are well suited for the location of a nuclear generating plant as evidenced by the 2,500 foot exclusion radius, remoteness from major population centers, sound rock foundation for major structures, abundant supply of cooling water from Lake Wylie, as well as favorable conditions of hydrology, geology, seismology, and meteorology. The Standby Nuclear Service Water Pond, designed for safe shutdown seismic conditions, provides a reliable ultimate heat sink for rejection of decay heat under postulated accident conditions.

1.2.2 Station Description

1.2.2.1 Principal Design Criteria

Catawba Nuclear Station is designed to comply with the intent of “General Design Criteria for Nuclear Power Plants,” Appendix A to 10CFR 50. Specific design criteria for the station are discussed in Chapter 3.

1.2.2.2 General Arrangement

The general arrangement of the major structures including equipment layout is shown on Figure 1-2 through Figure 1-20.

1.2.2.3 Nuclear Steam Supply System

The Nuclear Steam Supply System consists of a reactor and four closed reactor coolant loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump, a steam generator, loop piping, and instrumentation. The Nuclear Steam Supply System also contains an electrically heated pressurizer and certain auxiliary systems.

High pressure water circulates through the reactor core to remove the heat generated by the nuclear chain reaction. The heated water exits the reactor vessel and passes via the coolant loop piping to the steam generators. Here it gives up its heat to the feedwater to generate steam for the turbine generator. The cycle is completed when the water is pumped back to the reactor vessel.

The inherent design of the pressurized water, closed-cycle reactor minimizes the quantities of fission products released to the atmosphere. Three barriers exist between the fission product accumulation and the environment. These are the fuel cladding, the reactor vessel and coolant loops, and the reactor containment. The consequences of a breach of the fuel cladding are greatly reduced by the ability of the uranium dioxide lattice to retain fission products. Escape of fission products through fuel cladding defects would be contained within the pressure vessel, loops, and auxiliary systems. Breach of these systems or equipment would release the fission products to the reactor containment where they would be retained. The reactor containment is designed to adequately retain these fission products under the most severe accident conditions, as analyzed in Chapter 6 and Section 15.1.

As defined in Table 5-1, the 100% NSSS power level is 3488 MWt (Unit 1) and 3430 MWt (Unit 2), which includes 19 MWt from the reactor coolant pumps. Operation at the core design rating of 3469 MWt (Unit 1) and 3411 MWt (Unit 2) yields a steady state core average linear power of 5.53 kW/ft (Unit 1) and 5.44 kW/ft (Unit 2) and a corresponding peak power specified in the

COLR. Reactivity coefficients and other design parameters which are given in this analysis, and which are supported by analysis and experience with other similar plants, provide the basis for concluding that this reactor can be operated safely at the power levels of the application rating.

The reactor core, with its related Control and Protection System, is designed to function throughout its design lifetime without exceeding the acceptable fuel damage limits. The core design, together with process and residual heat removal systems, provides for this capability under all expected conditions of normal operations with appropriate margins for uncertainties and anticipated transient situations, including, as examples, the effects of the loss of reactor coolant flow, turbine trips due to steam and power conversion system malfunctions, and loss of external electrical load. Acceptable fuel damage limits can be found in Section 4.2.

The core is of the multi-enrichment region type. All fuel assemblies are mechanically similar, although the fuel enrichment is not the same in all assemblies.

In the initial core loading, three fuel enrichments are used. Fuel assemblies with the highest enrichments are placed in the core periphery, or outer region, and the two groups of lower enrichment fuel assemblies are arranged in a selected pattern in the central region. In subsequent refuelings, approximately one-third of the fuel is discharged. Fresh and spent fuel is arranged in the core in such a manner as to achieve optimum power distribution.

Rod cluster control assemblies (RCCA's) are used for reactor control and consist of clusters of cylindrical absorber rods. The absorber rods move within guide tubes in certain fuel assemblies. Above the core, each cluster of absorber rods is attached to a spider connector and drive shaff, which is raised and lowered by a drive mechanism mounted on the reactor vessel head. Trip of the rod cluster control assemblies is by gravity.

Pressure in the system is controlled by the pressurizer, where system pressure is maintained through the use of electrical heaters and water sprays. Steam can either be formed by the heaters, or condensed by a pressurizer spray to minimize pressure variations due to contraction and expansion of the coolant. Instrumentation used in the Reactor Coolant System is described in Chapter 7. Spring-loaded steam safety valves and power-operated relief valves for overpressure protection are connected to the pressurizer and discharge to the pressurizer relief tank, where the discharged steam is condensed and cooled by mixing with water.

The reactor coolant pumps are Westinghouse vertical, single-stage, mixed flow pumps of the shaft-seal type. The power supply system to the pumps is designed so that adequate coolant flow is maintained to cool the reactor core under all credible circumstances.

Unit 1 steam generators are BWI vertical U-tube units which contain alloy 690 tubes. Unit 2 steam generators are Westinghouse vertical U-tube units which contain alloy 600 tubes. Integral moisture separation equipment reduces the moisture content of the steam to one-quarter of one percent or less.

The reactor coolant piping and all of the pressure-containing and heat transfer surfaces in contact with reactor water are stainless steel clad except the steam generator tubes and fuel tubes, which are Inconel and Zircaloy, respectively. Reactor core internals, including control rod drive shafts, are stainless steel.

Auxiliary system components are provided to charge the Reactor Coolant System and add makeup water, purify reactor coolant water, provide chemicals for corrosion inhibition and reactor control, cool system components, remove decay heat when the reactor is shutdown, and provide for emergency safety injection.

1.2.2.4 Engineered Safety Features

Engineered safety features are provided to prevent accident propagation or to limit the consequences of postulated accidents. The principal criterion is to limit the potential offsite radiation dose from a Design Basis Accident (DBA) to less than the values of applicable federal regulations.

The Containment consists of a free-standing steel structure within a separate reinforced concrete Reactor Building with an annulus between the two structures. The Containment, including its penetrations, is designed to safely confine the radioactive material that could be released in the event of a loss-of-coolant accident.

The Ice Condenser prevents high post-accident pressures in the Containment by absorbing the energy of the released reactor coolant, thereby reducing the driving potential for escape of fission products from the Containment.

The Containment Spray System further minimizes the possibility of long-term fission product leakage by cooling the post-accident Containment atmosphere. The Annulus Ventilation System collects and filters leakage from the Containment and helps relieve post-accident pressure in the Containment by cooling the annulus.

The Containment Isolation System reduces the number of potential post-accident leakage paths by automatically closing Containment penetrations not required for post-accident functions.

The Hydrogen Control System protects the Containment from the danger of combustible gas accumulation following a loss-of-coolant accident.

The Emergency Core Cooling System delivers borated water to the reactor core, thereby limiting both the post-accident fuel cladding temperature rise and metal-water reaction.

The Air Return and Hydrogen Skimmer System returns air to the lower containment compartment following a loss-of-coolant accident and prevents hydrogen pocketing in Containment subcompartments.

The Nuclear Service Water System normally supplies cooling water from Lake Wylie. The Standby Nuclear Service Water Pond provides an assured source of NSW after simultaneous loss-of-coolant accident, station blackout, and loss of Lake Wylie due to a seismic occurrence.

With the exception of the Nuclear Service Water System, which has two independent sources of cooling water serving both units, the Engineered Safety Features are separate and independent for each unit. Sufficient redundancy for each system is provided to assure proper functioning even with the single-failure criterion.

1.2.2.5 Unit Control

The reactor is controlled by control rod movement and regulation of the boric acid concentration in the reactor coolant. During steady-state operation, the Reactor Control System maintains a programmed average reactor coolant temperature which rises in proportion to the load. The combined actions of the Reactor Control System, steam bypass to the condenser, and steam relief valves are designed to maintain station auxiliary load in the event the plant is electrically separated from the transmission system.

1.2.2.6 Electrical Systems

Each of the two nuclear units is provided with an independent electric power system to supply plant auxiliaries and to provide instrumentation and control power.

Under normal operating conditions, each nuclear unit is supplied electric power from its main generator via the unit auxiliary transformers. If electric power from the unit main generator is unavailable, the generator is isolated by generator breakers, and electric power is supplied to the unit from the offsite power system via two independent 230 kV transmission lines.

Each nuclear unit is provided with two diesel generators as standby power sources in the event that offsite power is unavailable. Each diesel generator will supply power to one of the two redundant and independent Class 1E power trains in each nuclear unit. The capacity of the diesel generators allows one of the two diesel generators per unit to supply the required safe shutdown or accident loads for its nuclear unit.

Instrumentation and control power for Class 1E instrumentation, the Reactor Trip System, and the Engineered Safety Features Actuation System is supplied by a Class 1E Instrumentation and Control Power System.

More detailed information can be found in Chapter 8.

1.2.2.7 Instrumentation and Control

Instrumentation and control is provided to assure safe, efficient operation. The Reactor Trip System and Engineered Safety Features Actuation System automatically initiate appropriate action whenever the parameters monitored by these systems exceed pre-established setpoints. These systems act to trip the reactor, actuate core cooling, close isolation valves, and initiate the operation of standby systems as required for plant safety.

More detailed information can be found in Chapter 7.

1.2.2.8 Steam and Power Conversion System

The Steam and Power Conversion System for each unit removes heat energy from the reactor coolant, delivers it in the form of steam to the turbine-generator, and converts it to electrical energy. The closed feedwater cycle condenses the steam and heats feedwater for return to the steam generators.

1.2.2.9 Fuel Handling and Storage

New fuel assemblies are removed from the rail car or truck and stored dry in the new fuel storage racks located in the New Fuel Storage Building or is stored in the Fuel Pool. Spent fuel is removed from the reactor core and placed in the fuel transfer mechanism by the Reactor Building manipulator crane. This transfer mechanism passes the fuel assembly through the transfer tube into the Fuel Pool Transfer Canal. Once in the Fuel Pool, the Fuel Pool manipulator crane places the spent fuel assembly in a storage rack where it remains for at least a suitable decay period.

Once spent fuel assemblies are removed from the core and placed in an underwater storage rack, a new fuel assembly is placed on the new fuel elevator which lowers the new fuel into the pool, where it is temporarily stored in the underwater racks. Then the fuel pool manipulator crane takes the new fuel to the transfer canal, places it in the fuel transfer mechanism, which takes it through the transfer tube into the Reactor Building.

Separate fuel storage and handling is provided for each of the two units with each system designed to prevent inadvertent criticality and to minimize the possibility of mishandling or faulty operation. A more detailed description of the fuel handling procedure can be found in Section 9.1.

In addition to the spent fuel pools, and Independent Spent Fuel Storage Installation (ISFSI) is available to provide long-term dry storage of irradiated fuel assemblies. Refer to the Catawba ISFSI UFSAR for more details on the Independent Spent Fuel Storage Installation.

1.2.2.10 Cooling Waters

Condenser cooling water is supplied by a closed cycle system using three round, mechanical draft, cross-flow Marley cooling towers per unit. During normal operation, cooling water for the Nuclear Service Water and Low Pressure Service Water Systems is pumped from the Beaver Dam Creek arm of Lake Wylie and returned to Big Allison Creek.

1.2.2.11 Fire Protection

The fire protection system at Catawba is designed to provide adequate detection and suppression systems for station personnel use. Fixed water sprinklers, hose stations, and portable extinguishers are provided to assure fire fighting capability.

More detailed information can be found in Section 9.5.1.

1.2.2.12 Radioactive Waste Management

The Gaseous Waste Disposal System collects, filters, monitors, and stores the gaseous effluent from processed reactor coolant. The Liquid Waste Disposal System includes capability for collection, segregation, storage, treatment, monitoring, disposal, and recording of the liquid wastes. Solid radioactive wastes are stored, packaged, and shipped offsite for ultimate disposition at an NRC licensed storage facility.

1.2.2.13 Shared Facilities and Equipment

Separate and similar systems and equipment are provided for each unit of the two unit Catawba Nuclear Station except as noted in Table 1-1. In those instances where some components of a system are shared by both units, only those components which are shared are shown.

1.2.2.14 Standby Shutdown Facility (SSF)

The Standby Shutdown Facility (SSF) is designed to provide an alternate and independent means to achieve a hot standby condition, and maintain the hot standby mode without recourse to damage control measures. The SSF is intended to mitigate the consequences of a postulated fire or act of sabotage to one or both units at Catawba, in the event that normal and emergency systems are rendered inoperable. The Facility includes a disel generator, control room, electrical distribution equipment, and other support systems. The SSF diesel generator is also credited as the alternate AC (AAC) power source required for safe shutdown during the required station blackout coping duration.

1.2.2.15 Emergency Supplemental Power Source (ESPS) Enclosures

The Emergency Supplemental Power Source enclosures will be permanently installed and contain a Non-Safety Related, commercial grade system, which meet the requirements specified by the NRC Branch Technical Position (BTP) 8-8. The ESPS system consists of the following major componenents:

- Two ESPS Diesel Generator sets (ESPS DGs)
- 6.9kV switchgear

- A 6.9 kV/480 VAC dry transformer
- A 6000 kWe, 6.9 kV resistive load bank.

The major components of the ESPS will be located inside the plant protected area, and outside the existing power block buildings, in the yard area located to the Northwest of the Unit 2 Turbine Building and west of the Monitor Tank Building as shown in Figure 1-19.2, Figure 1-19.3, and Figure 1-20.

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1.3 Comparison Tables

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

1.3.1 Comparisons with Similar Facility Designs

The design of Catawba Nuclear Station is similar in many respects to designs of other nuclear stations which are in operation or under construction. Table 1-2 provides a listing of principal similarities and principal differences between Catawba and McGuire Nuclear Station and Watts Bar Nuclear Power Plant. All of these plants have a Westinghouse four-loop NSSS.

1.3.2 Comparison of Final and Preliminary Information

Table 1-3 provides a listing of significant differences between the final design and preliminary design of the Catawba Nuclear Station. In addition to these changes, numerous items of design development have been incorporated in the design descriptions of the Catawba FSAR, where only criteria or functional requirements were stated in the PSAR.

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1.4 Identification of Agents and Contractors

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

Duke Power is responsible for the design, purchasing, quality assurance, testing, startup, and operation of Catawba Nuclear Station, a practice followed for most of the Company's major generating facilities now in service or planned.

The Nuclear Generation Department has the responsibility for specification of materials and equipment, design of structures and systems and preparation of construction and installation drawings and the responsibility for preoperational testing and initial startup as well as operation and maintenance of the station. The Purchasing Department has the responsibility for the purchase of all materials, equipment and services required to construct, operate and maintain all plants and facilities. Other departments within Duke are available as needed to assist in the design, construction or operation of the station. The organizational structure of Duke Power Company is presented in Section 13.1.

Technical qualifications and nuclear experience of key Duke personnel are presented in Section 13.1.

Duke has contracted with Westinghouse to design, manufacture and deliver to the site two complete Nuclear Steam Supply Systems and fuel. In addition, Westinghouse is supplying technical assistance for erection of the Nuclear Steam Supply equipment and consultation for initial fuel loading, testing and startup of the Nuclear Steam Supply Systems, with coordination, scheduling and administrative direction by Duke. Westinghouse nuclear activity dates from 1936, when a program of research in nuclear physics was begun. Since that time the company has established a broad technological foundation in nuclear power application. Westinghouse experience in commercial nuclear power is evidenced by their participation in projects presently operating, under construction, or planned which exceed 49,000 MWe total generating capacity.

For procurement involving the use of vendors located outside the United States, Duke selects the vendors only after a determination that quality assurance programs are at least equal to similar programs of domestic vendors. Any components supplied to Duke are designed to meet applicable domestic industry code requirements as stated by the equipment specifications.

Duke is utilizing consultants, as necessary, to perform selected design work and to obtain specialized services. The firm of Charles T. Main, Inc. of Boston, Massachusetts was retained to assist in performing flood studies. Engineering Data Systems, Inc. (EDS) of San Francisco, California has been retained to assist in the seismic design of piping. In addition, EDS is used as necessary to review seismic design of components furnished by Westinghouse or purchased by Duke. EDS has participated in the seismic design of more than twenty nuclear stations during recent years and is well qualified in these areas. Law Engineering Testing Company has been retained by Duke to conduct investigations in geology, seismology, subsurface conditions and foundations and groundwater hydrology. This work is conducted under the direction of Professor George F. Sowers, Chief Soils Consultant of Law Engineering, and Regents Professor of Civil Engineering at Georgia Institute of Technology. An independent geologic review panel was retained to review significant geologic features. The panel consisted of Dr. Lynn Glover, III, Professor of Geology at Virginia Polytechnic Institute; Dr. Robert D. Hatcher, Jr., Professor of Geology at Clemson University; and Norman K. Olson, State Geologist, South Carolina State Development Board.

Duke has also utilized the service of McNeary Insurance Consulting Services, Inc. of Charlotte, North Carolina to assist in the area of fire protection at Catawba.

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1.5 Material Incorporated by Reference

This section lists topical reports, which provide information additional to that provided in this FSAR and have been filed separately with the NRC in support of this and similar applications.

1.5.1 Westinghouse Topical Reports

An "A", "P", or "L" at the end of a WCAP number generally have the following meanings:

A NRC review is complete and an SER has been issued.

P Proprietary version; a non-proprietary version of the same WCAP is normally issued also.

L Indicates use for a Licensing issue.

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1.5.2 Duke Reports

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3.	Emergency Plan for Catawba Nuclear Station, Rev. 00-1, May 2000	13.3
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6.	*Final Geologic Report on Brecciated Zones, March 1, 1976.*	*2.5*
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9.	“FSAR Chapter 15 System Transient Analysis Methodology,” DPC-NE-3002-A, Rev 4b, Duke Power Co., SER dated September, 2010	15.2, 15.3, 15.6
10.	"Mass & Energy Release and Containment Response Methodology," DPC-NE-3004-PA, Rev. 1, Duke Power Co., SER dated February 29, 2000	6.2, 6.7
11.	DPC-NE-1004, Duke Power Company, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, November 1992. Amended to include 24 axial level uncertainties and approved by NRC SER, TAC Nos. M94403 and M94404, Docket Nos. 5-413, 50-414, 50-369, 50-370, SER Rev. 1a dated April 1996.	4.3
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	Report	FSAR Section
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15.	*DPC-NE-2007P-A, Duke Power Company Fuel Reconstitution Analysis Methodology, Duke Power Company, October 1995.*	*4.2*
16.	*DPC-NE-2008P-A, Duke Power Company Fuel Mechanical Reload Analysis Methodology Using TACO3, April 1995.*	*4.2, 4.3*
17.	DPC-NE-2010P-A, McGuire Nuclear Station, Catawba Nuclear Station Nuclear Physics Methodology for Reload Design, Duke Power Company, SER Rev. 2a dated June 2003.	4.3
18.	DPC-NE-2011P-A, Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors, Duke Power Company, SER Rev. 1a dated October 2002.	4.3
19.	DPC-NE-3001P-A, McGuire Nuclear Station, Catawba Nuclear Station, "Multidimensional Reactor Transient Analysis and Safety Analysis Physics Parameters Methodology", Duke Power Company, Rev OA, December, 1997.	4.3
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[HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED]		
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22.	DPC-NE-1005-P-A, Duke Power Company, Nuclear Design Methodology using CASMO-4/SIMULATE-3 MOX, SER dated November 12, 2008.	4.3
23.	Deleted Per 2018 Update	

1.5.3 B&W Reports

	Report	FSAR Section
1.	Deleted Per 2001 Update	
2.	Deleted Per 2001 Update	

	Report	FSAR Section
3.	Deleted Per 2001 Update	
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9.	*BAW-10183P-A, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, Lynchburg, Virginia, July 1995.*	*4.2*
10.	*BAW-10172P-A, Mark-BW Mechanical Design Report, Babcock & Wilcox, Lynchburg, Virginia, July 1988.*	*4.2, 4.4*
11.	*BAW-10147P-A, Rev. 1, Fuel Rod Bowing in Babcock & Wilcox Fuel Designs, Babcock & Wilcox, Lynchburg, Virginia, May 1993*	*4.2, 4.4*
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14.	*BAW-10199P-A, The BWU Critical Heat Flux Correlations, Framatome Cogema Fuels, Lynchburg, Virginia, August 1996.*	*4.4*
15.	*BAW-10186P-A, Extended Burnup Evaluation, Framatome Cogema Fuels, SER dated January 25, 1999.*	*4.2*
16.	*BAW10179A, Safety Criteria and Methodology for Acceptable Cycle reload Analyses, August 1, 1993.*	*4.2*

1.5.4 EPRI Reports

	Report	FSAR Section
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1.5.5 Other Reports

	Report	FSAR Section
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1.6 Drawings and Other Detailed Information

1.6.1 Electrical Instrumentation and Control Drawings

All safety related electrical instrumentation and control drawings for the Catawba Nuclear Station were provided under separate cover in the Catawba Electrical Schematics books.

These drawings were submitted for initial review of the FSAR and have not been updated to reflect subsequent changes in plant design.

Figure 1-21 provides the symbols used in the electrical instrumentation and control drawings.

1.6.2 Piping and Instrumentation Diagrams

Two large-scale copies of each piping and instrumentation diagram included in the Catawba FSAR have been provided to the NRC.

Figure 1-22 through Figure 1-24 provide the symbols and abbreviations used in the piping and instrumentation diagrams.

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1.7 Regulatory Guides

The purpose of NRC Regulatory Guides are (1) to describe methods acceptable to the NRC Regulatory Staff for implementing parts of the Commission's regulations and (2) to provide guidance to applicants for permits and licenses.

Section 1.7.1.1 summarizes Duke Power's position on each applicable Division I Regulatory Guide.

1.7.1.1 Regulatory Guides

Regulatory Guide 1.1

Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal (Safety Guide 1, 11/2/70).

Discussion

Paragraph Deleted Per 2001 Update.

Duke follows the guidance contained in Regulatory Guide 1.1 as discussed in Section 6.2.2.3.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.2

Thermal Shock to Reactor Pressure Vessels (Safety Guide 2, 11/1/70).

Discussion

Conformance to Regulatory Guide 1.2 as discussed in Section 5.3.3. NOTE: This Reg Guide was withdrawn by the NRC on 7/31/91 in 56 FR 36175.

Regulatory Guide 1.3

Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors (Revision 2, 6/74).

Discussion

This regulatory guide is not applicable to the Catawba Nuclear Station.

Regulatory Guide 1.4

Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors (Revision 2, 6/74).

Discussion

The discussion of historical short-term (accident) diffusion estimates provided in 2.3.4.2 and the current analysis of radiological consequences of loss-of-coolant accidents presented in Chapter 15 are in compliance with Regulatory Guides 1.145 and 1.194, as well as Regulatory Guide 1.4 with the following clarification:

For site or regional data and time periods of 0-8 hours:

Provide joint frequency distribution of wind direction and speed by atmospheric stability class (derived from Regulatory Guide 1.23 [Safety Guide 23, 2/17/72]) based on appropriate meteorological measurement heights and data reporting periods.

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Provide cumulative frequency distributions of hourly calculated atmospheric dilution factors (X/Q) from joint frequency tabulations using the minimum site boundary distance (exclusion area) and actual site boundary (exclusion area) distances (or distance to maximum X/Q value) from the effluent release point(s). Report to X/Q values from each of these distributions that are exceeded 5% and 50% (median value) of the time.

On September 30, 2005, the NRC staff approved full scope implementation of the method of Alternative Source Terms (AST) at Catawba Nuclear Station (Ref. 19) based on part on their review of the AST analysis of the Loss-of-Coolant Accident at Catawba (Ref. 20-27). With the exceptions noted above, the analysis of radiological consequences of the Loss-of-Coolant Accident (LOCA) no longer conforms to Regulatory Guide 1.4. Rather, this analysis was completed with the method of AST and conforms in general to the NRC regulatory positions of Regulatory Guide 1.183 and Appendix A (see below).

Regulatory Guide 1.5

Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (Safety Guide 5, 3/10/71).

Discussion

This regulatory guide is not applicable to the Catawba Nuclear Station.

Regulatory Guide 1.6

Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Safety Guide 6, 3/10/71).

Discussion

The Catawba design complies with the recommendations of Regulatory Guide 1.6 as discussed in Section 8.3.2.2.3.

Regulatory Guide 1.7

Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident (Revision 2, 11/78).

Discussion

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED.

The design of the electric hydrogen recombines and the Containment Hydrogen Sample and Purge System is in compliance with Regulatory Guide 1.7 as noted in Section 6.2.5.1. With the elimination of the design basis LOCA hydrogen release per 10CFR50.44, the requirements for hydrogen control systems to mitigate such a release during a design basis accident were eliminated.

Regulatory Guide 1.8

Personnel Selection and Training (Revision 1, 9/75).

Discussion

Qualifications of station personnel are in accordance with Regulatory Guide 1.8 with the exception of those for the Radiation Protection Manager as discussed in Section 13.1.3.

Regulatory Guide 1.9

Selection, Design and Qualifications of Diesel Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants. (Revision 2, 12/79)

Discussion

The selection criteria for the diesel generators used as standby power sources complies with the requirements of Regulatory Guide 1.9 as discussed in Section 8.3.1.2.4.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.10

Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures (Revision 1, 1/2/73).

Discussion

Testing of reinforcing bar mechanical splices in Category I concrete structures is performed in compliance with Regulatory Guide 1.10 as discussed in Section 3.8.1.6.1.2. NOTE: This Reg Guide was withdrawn by the NRC on 7/21/81 in 46 FR 37579.

Regulatory Guide 1.11

Instrument Lines Penetrating Primary Reactor Containment (Safety Guide II, 3/10/71).

Discussion

Westinghouse furnished equipment for the nuclear steam supply system meets the recommendations of this regulatory guide as discussed in Section 7.3.1.1.2.

The instrument lines penetrating primary reactor containment comply with the requirements of Regulatory Guide 1.11 as discussed in Section 6.2.4.1.

Regulatory Guide 1.12

Nuclear Power Plant Instrumentation for Earthquakes (Revision 2, 3/97).

Discussion

Seismic instrumentation for Catawba is in conformance with the requirements of Regulatory Guide 1.12, except that: instrumentation locations are only applied to major Seismic Category I structures [as defined per Table 2-90] that contain mechanical and/or electrical equipment important to safe shutdown of the plant (in accordance with the original Catawba commitment to RG 1.12, Revision 1, 4/74); and "free field" sensors (as recommended per ANSI N18.5-1974, Section 4.1.1 and subsequent ANSI/ANS-2.2-1988, Section 4.1.1) are not required (consistent with the original CNS site design characteristics and resultant seismic instrumentation system implementation, given negligible soil interaction of the containment structure foundation). See Section 3.7.4 for further discussion.

Regulatory Guide 1.13

Spent Fuel Storage Facility Design Basis (Revision 1, 12/75).

Discussion

The design of the spent fuel storage facility is in conformance with the requirements of Regulatory Guide 1.13 as discussed in Section 9.1.2.1.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.14

Reactor Coolant Pump Flywheel Integrity (Revision 1, 8/75).

Discussion

The Westinghouse design follows the recommendations of Regulatory Guide 1.14, Revision 1, except for the following:

1. *Post-spin inspection*

   Westinghouse has shown in Reference 1 that the flywheel would not fail at 290 percent of normal speed for a flywheel flaw of 1.15 inches or less in length. Results for a double-ended guillotine break at the pump discharge with full separation of pipe ends assumed, show the maximum overspeed was to be less than 110 percent of normal speed. The maximum overspeed was calculated in Reference 1 to be about 280 percent of normal speed for the same postulated break, and an assumed instantaneous loss of power to the reactor coolant pump. In comparison with the overspeed presented above, the flywheel is tested at 125 percent of normal speed. Thus, the flywheel could withstand a speed up to 2.3 times greater than the flywheel spin test speed of 125 percent provided that no flaws greater than 1.15 inches are present. If the maximum speed were 125 percent of normal speed or less, the critical flaw size for failure would exceed 6 inches in length. Nondestructive tests and critical dimension examinations are all performed before the spin tests. The inspection methods employed (described in Reference 1) provide assurance that flaws significantly smaller than the critical flaw size of 1.15 inches for 290 percent of normal speed would be detected. Flaws in the flywheel will be recorded in the pre-spin inspection program (see Reference 1). Flaw growth attributable to the spin test (i.e., from a single reversal of stress, up to speed and back), under the most adverse conditions, is about three orders of magnitude smaller than that which nondestructive inspection techniques are capable of detecting. For these reasons, Westinghouse performs no post-spin inspections and believes that pre-spin test inspections are adequate.

2. *Interference fit stresses and excessive deformation*

   Much of Revision 1 to Regulatory Guide 1.14 deals with stresses in the flywheel resulting from the interference fit between the flywheel and the shaft. Because Westinghouse's design specifies a light interference fit between the flywheel and the shaft; at zero speed, the hoop stresses and radial stresses at the flywheel bore are negligible. Centering of the flywheel relative to the shaft is accomplished by means of keys and/or centering devices attached to the shaft, and at normal speed, the flywheel is not in contact with the shaft in the *sense intended by Revision 1. Hence, the definition of "Excessive Deformation," as defined in Revision 1 of Regulatory Guide 1.14, is not applicable to the Westinghouse design since the enlargement of the bore and subsequent partial separation of the flywheel from the shaft does not cause unbalance of the flywheel. Extensive Westinghouse experience with reactor coolant pump flywheels installed in this fashion has verified the adequacy of the design.*

Westinghouse's position is that combined primary stress levels, as defined in Revision 0 of Safety Guide 1.14 (Regulatory Positions C.2.a and C.2.c) are both conservative and proven and that no changes to these stress levels are necessary. Westinghouse designs to these stress limits and thus does not have permanent distortion of the flywheel bore at normal or spin test conditions.

3. *Discussion B, cross rolling ratio 1 to 3*

   Westinghouse's position is that specifications of a cross rolling ratio is unnecessary since past evaluations have shown that ASME SA-533, Grade B, Class 1 materials produced without this requirement have suitable toughness for typical flywheel applications. Proper material selection and specification of minimum material properties in the transverse direction adequately ensure flywheel integrity. An attempt to gain isotropy in the flywheel material by means of cross rolling is unnecessary since adequate margins of safety are provided by both flywheel material selection (ASME SA-533, Grade B, Class 1) and by specifying minimum yield and tensile levels and toughness test values taken in the direction perpendicular to the maximum working direction of the material.

4. *Regulatory Position C.1.a, relative to vacuum-melting and degassing process of the electroslag process.*

   The requirements for vacuum melting and degassing process or the electroslag process are not essential in meeting the balance of the regulatory position nor do they, in themselves, ensure compliance with the overall regulatory position. The initial Safety Guide 14 (10/27/71) stated that the "flywheel material should be produced by a process that minimized flaws in the material and improves its fracture toughness properties." This is accomplished by using ASME SA-533 material including vacuum treatment.

5. *Regulatory Position C.2.b*

   Westinghouse suggests that this paragraph be reworded as follows in order to remove the ambiguity of reference to an undefined overspeed transient.

   "Design speed should be 125 percent of normal speed or the speed to which the pump motor might be electrically driven by station turbine generator during anticipated transients, whichever is greater. Normal speed is defined as the synchronous speed of the alternating current drive motor at 60 hertz."

Inservice inspection of reactor coolant pump flywheels is performed in accordance with Regulatory Position C.4 with the exception of C.4.b(4) which has been rewritten as follows:

(4) Acceptance criteria are based upon whether or not the inservice examination indicated an increase in flaw size or growth rate of the flywheel greater than that predicted for its service life as defined in C.4.b(5).

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

Regulatory Guide 1.15

Testing of Reinforcing Bars for Category I Concrete Structures (Revision 1, 12/28/72).

Discussion

Duke complies with the requirements of Regulatory Guide 1.15 which specified conformance with ASTM A-615-72 for testing of reinforcing steel. Duke has used later versions of ASTM A-615 as they were issued. The use of the latest version of ASTM A-615 allows Duke to utilize the current information and data concerning deformed and plain billet-steel bars for concrete reinforcing without compromising the intent of Reg Guide 1.15. ***NOTE:*** *Reg Guide 1.15 was withdrawn by the NRC on July 21, 1981 in 46 FR37579.*

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.16

Reporting of Operating Information - Appendix A Technical Specifications (Revision 4, 8/75).

Discussion

Reporting of operating information will comply with the requirements of Regulatory Guide 1.16 as incorporated into the Technical Specifications. NOTE: This Reg Guide was withdrawn by the NRC on 8/11/09 in 74 FR 40244.

Regulatory Guide 1.17

Protection of Nuclear Power Plants Against Industrial Sabotage (Revision 1, 6/73).

Discussion

The Security Plan (Section 13.6) conforms to the requirements of Regulatory Guide 1.17. ***NOTE:*** *Reg Guide 1.17 was withdrawn by the NRC on July 5, 1991 in FR 30777*

Regulatory Guide 1.18

Structural Acceptance Test for Concrete Primary Reactor Containments (Revision 1, 12/28/72).

Discussion

Catawba has a freestanding steel containment; therefore, Regulatory Guide 1.18 is not applicable. ***NOTE:*** *Reg Guide 1.18 was withdrawn by the NRC on July 21, 1981 in 46FR 37579*

Regulatory Guide 1.19

Nondestructive Examination of Primary Containment Liner Welds (Revision 1, 8/11/72 of Safety Guide 19).

Discussion

Nondestructive testing of the Containment bottom liner is in partial conformance with the requirements of this regulatory guide as discussed in Section 3.8.2.2. ***NOTE:*** *Reg Guide 1.19 was withdrawn by the NRC on July 21, 1981 in 46FR 37579*

Regulatory Guide 1.20

Comprehensive Vibration Assessment Program for Reactor Internals During Pre-operational and Initial Startup Testing (Revision 2, 5/76).

Discussion

The Westinghouse position on Regulatory Guide 1.20 is discussed in Section 3.9.2.4.

Regulatory Guide 1.21

Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants (Revision 1, 6/74).

Discussion

Duke complies with the requirements of Regulatory Guide 1.21 as discussed in Section 11.5.2 with the exception of monitor calibration as discussed in Section 11.5.1.2.5.

Regulatory Guide 1.22

Periodic Testing of Protection System Actuation Functions (Safety Guide 22, 2/17/72).

Discussion

Periodic testing of the Reactor Trip and Engineered Safety Features Actuation Systems and of the Class 1E power systems complies with the requirements of Regulatory Guide 1.22 as discussed in Sections 7.1.2.4.2 and 8.1.5.2.

Regulatory Guide 1.23

Onsite Meteorological Programs (Safety Guide 23, 2/17/72).

Discussion

The meteorological monitoring program for Catawba complies with the recommendations of Safety Guide 23.

Regulatory Guide 1.24

Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Safety Guide 24, 3/23/72).

Discussion

Duke Power complies with the requirements of Regulatory Guide 1.24, with the following exceptions: Organ dose is computed using the dose coefficients provided in Federal Guidance Report 11, and effective and skin dose is computed using the dose coefficients provided in Federal Guidance Report 12.

Regulatory Guide 1.25

Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25, 2/23/72).

Discussion

On April 23, 2002, the NRC approved the implementation of the method of AST for the analysis of radiological consequences of the Fuel Handling Accident and Weir Gate Drop (Ref. 30 & 31). This analysis conforms to Regulatory Guide 1.183 and Appendix B; Regulatory Guide 1.25 no longer applies to Catawba Nuclear Station.

Regulatory Guide 1.26

Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (Revision 3, 2/76).

Discussion

The redefinitions of the safety classes found in ANSI N18.2a-1975 are utilized by Westinghouse.

Duke complies with Regulatory Guide 1.26 except that position C.3 is modified as discussed in Section 3.2.2.

Regulatory Guide 1.27

Ultimate Heat Sink for Nuclear Power Plants (Revision 2, 1/76).

Discussion

The Catawba ultimate heat sink meets the requirements of Regulatory Guide 1.27 as discussed in Section 9.2.5.4.

Regulatory Guide 1.28

Quality Assurance Program Requirements (Design and Construction) (Revision 1, 3/78).

Discussion

The quality assurance program for Catawba complies with the requirements of Regulatory Guide 1.28 as discussed in Chapter 17.

Regulatory Guide 1.29

Seismic Design Classification (Revision 3, 9/78).

Discussion

Westinghouse classifies each component important to safety as Safety Class 1, 2 or 3 and these classes are qualified to remain functional in the event of the Safe Shutdown Earthquake, except where exempted by meeting all of the below requirements. Portions of systems required to perform the same safety function as required of a safety class component which is part of that system shall be likewise qualified or granted exemption. Conditions to be met for exemption are:

1. Failure would not directly cause a Condition III or IV event (as defined in ANSI N18.2-1973),
2. There is no safety function to mitigate, nor could failure prevent mitigation of, the consequences of a Condition III or IV event,

3. Failure during or following any Condition II event would result in consequences no more severe than allowed for a Condition III event, and
4. Routine post-seismic procedures would disclose loss of the safety function.

Structures, systems, and components in Duke's scope comply with the requirements of Regulatory Guide 1.29.

See Section 7.1.2.4.2 and Section 8.1.5.2 for further discussion and Section 3.2 for a tabulation of structures, equipment and systems that are designed for the SSE.

Regulatory Guide 1.30

Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30, 8/11/72).

Discussion

The quality assurance program for Catawba complies with the requirements of Regulatory Guide 1.30 as discussed in Chapter 17.

Regulatory Guide 1.31

Control of Ferrite Content in Stainless Steel Weld Metal (Revision 3, 4/78).

Discussion

The Westinghouse position concerning the control of delta ferrite in stainless steel welding is discussed in Section 5.2.3.4.6. The Westinghouse production weld verification program, as described in Reference 4, was approved as a satisfactory substitute for conformance with the NRC Interim Position on Regulatory Guide 1.31 (April 1974). The results of the verification program have been summarized and documented in Reference 5.

The following requirements are used for austenitic stainless steel welding of nuclear safety related systems at Catawba. These requirements are in lieu of those specified in Regulatory Guide 1.31, Revision 3.

1.0 General

All austenitic stainless steel welding shall conform to the fabrication requirements of the contractual American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, and to the Duke Power Company Quality Assurance Program. All new welding procedures and welding procedure qualifications shall conform to the requirements of the latest edition of the ASME, Boiler and Pressure Vessel Code, Section IX. Requirements other than those stated herein shall not be considered mandatory.

2.0 Welding Filler Material

2.1 All bare welding filler material, including consumable inserts, shall meet the requirements of ASME/SFA 5.9 including applicable addendum and shall contain delta ferrite content of 5 to 20%.

All coated products and submerged arc electrodes shall meet the applicable ASME/SFA requirements and shall contain delta ferrite content of 5 to 20%. Delta ferrite determinations for SFA 5.4 type 16-8-2 weld metal or for filler metal used for weld metal cladding are not required.

2.2 Each heat and lot of filler material shall be verified for delta ferrite content. Delta ferrite determinations for consumable inserts, electrodes, rod or wire filler metal used with the gas tungsten arc welding process, and deposits made with the plasma arc welding process shall be predicted from their chemical composition using an applicable constitutional diagram to demonstrate compliance. Delta ferrite determination shall be made for all other processes using an applicable constitutional diagram with the weld metal chemical composition taken from an undiluted weld deposit. For the submerged arc welding process, a ferrite determination shall be made for each electrode and flux combination. The delta ferrite content of the weld deposit pad shall be in the range of 5 to 20%.

2.3 A certified chemical test report shall accompany each Heat or Lot of material and shall be verified to meet the above requirements prior to issuance of the material for welding. This documentation shall be retained at the job site.

3.0 <u>Traceability</u>

3.1 For ASME Section III Class 1, 2, and 3 welds; each Lot and Heat of filler material shall be readily identifiable and traceable to the specific joint for which it was used, by actual field documentation.

4.0 <u>Welding Procedure</u>

Specific welding procedures shall be identified for each joint to be welded. The interpass temperature shall be restricted to 350°F maximum to control sensitization.

5.0 <u>Inspection Welds</u>

5.1 All welds shall be visually inspected for cracks and other unacceptable defects.

5.2 All ASME Section III Class 1 circumferential butt welds (excluding welds 1 inch NPS and less) shall be inspected by radiography and the liquid penetrant method. All ASME Section III Class 2 circumferential butt welds (greater than 1 inch NPS) shall be inspected by radiography. All ASME Section III Class 3 circumferential butt welds (greater than 4 inches NPS) shall be inspected by the liquid penetrant method. All nondestructive examinations shall be performed in the manner required by the ASME Code. Microfissuring of the magnitude considered to be detrimental to the structural integrity of weldments will be within the sensitivity levels of the NDE methods employed, and shall be rejected and treated as similar types of defects in accordance with the ASME Code's acceptance criteria.

5.3 Other “in process” weld inspections shall be performed (such as verification of welding procedure parameters, welder qualification, joint identification, etc.) in accordance with ASME Section III and Section IX Code requirements and additional requirements of the Duke Power Company Operational Quality Assurance Program.

Regulatory Guide 1.32

Criteria for Safety-Related Electric Power Systems from Nuclear Power Plants (Safety Guide 32, 8/72).

Discussion

The design of the Class 1E onsite power systems complies with the recommendations of Regulatory Guide 1.32 as discussed in Sections 8.3.1.2.5 and 8.3.2.2.4.

Regulatory Guide 1.33

Quality Assurance Program Requirements (Operation) (Revision 2, 2/78).

Discussion

The quality assurance program for Catawba complies with the requirements of Regulatory Guide 1.33 as discussed in Section 13.5.1.1 and Chapter 17.

Regulatory Guide 1.34

Control of Electroslag Weld Properties (12/28/72).

Discussion

Where electroslag welding is used in fabricating nuclear plant components, the Westinghouse procurement practice requires vendors to follow the recommendations of Regulatory Guide 1.34.

Regulatory Guide 1.35

Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures (Revision 2, 1/76).

Discussion

Catawba has a freestanding steel containment; therefore, Regulatory Guide 1.35 is not applicable.

Regulatory Guide 1.36

Nonmetallic Thermal Insulation for Austenitic Stainless Steel (2/23/73).

Discussion - Westinghouse

The Westinghouse practice meets the recommendations of Regulatory Guide 1.36 and is more stringent in several respects as discussed in Section 5.2.3.2.3.

Discussion - Duke

Nonmetallic thermal insulation used for austenitic stainless steel piping and components located both inside and outside of Containment meets the requirements of Regulatory Guide 1.36.

Regulatory Guide 1.37

Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (3/16/73).

Discussion

The quality assurance program for Catawba complies with the requirements of Regulatory Guide 1.37. See Sections 5.2.3.4.1, 5.4.2, 10.3.6.2, and Chapter 17 for additional discussion.

Regulatory Guide 1.38

Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants (Revision 2, 5/77).

Discussion

The quality assurance program for Catawba complies with the requirements of Regulatory Guide 1.38 as noted in Chapter 17.

Regulatory Guide 1.39

Housekeeping Requirements for Water-Cooled Nuclear Power Plants (Revision 2, 9/77).

Discussion

The quality assurance program for Catawba complies with the requirements of Regulatory Guide 1.39 as noted in Chapter 17.

Regulatory Guide 1.40

Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants (3/16/73).

Discussion

Continuous-duty motors installed inside the containment are qualified in accordance with Regulatory Guide 1.40 as noted in Section 3.11.2.1.4.

Regulatory Guide 1.41

Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments (3/16/73).

Discussion

Preoperational testing of the Class 1E ac system is performed in accordance with the recommendations of Regulatory Guide 1.41 as discussed in Section 14.2.

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

Regulatory Guide 1.42

Interim Licensing Policy on As Low As Practicable for Gaseous Radioiodine Releases from Light-Water-Cooled Nuclear Power Reactors (Revision 1, 3/74).

Discussion

This regulatory guide was withdrawn by NRC on March 18, 1976, in 41FR 11891.

Regulatory Guide 1.43

Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (5/73).

Discussion

Conformance with this regulatory guide is discussed in Section 5.3.1.4.

The BWI steam generator design complies with the NRC regulatory guidelines. The ferritic base metals which are clad (SA508, C1.3, equivalent to SA533 Grade B, C1.1, and SA508 C1.1), are procured to fine grain practice and are not considered susceptible to underclad cracking. Weld procedure qualification is performed on material of the same specification (or equivalent) as used in production. BWI performs a 70 degree longitudinal UT examination for underclad cracking on all primary side clad inside RSG surfaces.

Regulatory Guide 1.44

Control of the Use of Sensitized Stainless Steel (5/73).

Discussion

The Westinghouse position on Regulatory Guide 1.44 is discussed in Section 5.2.3.4.

The BWI steam generator design complies with the NRC regulatory position. Sensitized stainless steels are used for cladding in the primary head assembly, for cladding on gasket surfaces or for diaphragms, for the feedwater ring assembly and for the channel divider plate. In each instance the cladding is not a pressure retaining component and is L grade material on all primary surfaces; the feedwater assembly on the secondary side is not subject to post-weld heat treatment temperature and the divider plate is L grade material and is not subject to post-weld heat treatment temperatures.

The following requirements apply to austenitic stainless steels used by Duke in nuclear safety related systems. These requirements are in lieu of those specified in Regulatory Guide 1.44.

1.0 General:

All austenitic stainless steels and the fabrication thereof shall conform to the requirements of the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, 1971 or later Edition and/or later contractual addenda. Requirements other than those stated herein shall not be considered mandatory.

2.0 Definition:

Sensitized stainless steel is defined as those types of unstabilized austenitic stainless steels (typically types 304 and 316) which have been exposed to elevated temperatures (800°F to 1500°F) for extended periods of time. Severe sensitization is defined as the condition of a component of highly sensitized (long term exposure to elevated temperatures) material exposed to high stress in an highly corrosive environment.

3.0 Evaluation of Material:

When highly sensitized stainless steel is to be used in nuclear components, an engineering evaluation of the magnitude of stress, corrosiveness of environment and degree of sensitization shall be performed to determine if the sensitization is “severe”. The degree of sensitization shall be determined by metallurgical evaluation of material exposed to the same time-temperature variables. Components exposed only to the reactor coolant (such as reactor internals) shall not be of concern as the corrosiveness of this environment shall be controlled in accordance with paragraph 5.0 below. Weldments, which will not receive a sensitizing post weld stress relief, shall not be of concern as they shall be controlled in accordance with paragraph 6.0 below.

4.0 Cleanness:

Control shall be exercised to prevent excessive exposure of stainless steel to halogens during manufacturing and construction. Components shall be cleaned in accordance with Duke Power Company cleaning procedures. Pickling of highly sensitized stainless steel shall be avoided. Components shall be protected and stored in accordance with Duke Power Company and/or Vendor specifications.

Weld joints shall be cleaned before and after welding in accordance with Duke Power Company welding specifications.

5.0 Reactor Coolant:

If highly sensitized stainless materials are to be exposed to the reactor coolant, corrosiveness of the coolant shall be controlled to the following requirements:

1. 10ppm max dissolved oxygen at temperatures above 250°F during normal operation.
2. 5.9 pH
3. 15ppm max chlorides

6.0 Weldment Sensitization:

Weldments shall not be post weld heat treated within the temperature range of 800°F to 1500°F unless and evaluation in accordance with paragraph 3.0 above is performed.

Voltage and amperage ranges, size and type of welding electrode shall be specified in welding procedures and monitored to assure compliance. To prevent excessive time of exposure to sensitizing temperatures, a maximum interpass temperature of 350°F shall be specified and monitored on production weldments.

Delta ferrite and other welding controls shall be in accordance with Duke's position on Regulatory Guide 1.31.

Regulatory Guide 1.45

Reactor Coolant Pressure Boundary Leakage Detection Systems (5/73).

Discussion

Adopted with comments and exceptions as shown on Table 5-46.

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

Regulatory Guide 1.46

Protection Against Pipe Whip Inside Containment (5/73).

Discussion

The Westinghouse/Duke practice is consistent with Regulatory Guide 1.46, as discussed in Sections 3.6 and 5.4.11.3. ***NOTE:*** *This Reg Guide was withdrawn by the NRC on 3/11/85 in 50FR 9732*

Regulatory Guide 1.47

Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (5/73).

Discussion

The engineered safety features bypass indication panel is designed in accordance with the recommendations of Regulatory Guide 1.47 as noted in Section 7.1.2.4.2.

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

Regulatory Guide 1.48

Design limits and Loading Combinations for Seismic Category I Fluid System Components (5/73).

Discussion – Westinghouse

NOTE: *This Reg Guide was withdrawn by the NRC on 3/11/85 in 50FR 9732.*

Westinghouse supplied components are designed using the stress limits and loading combinations presented in Sections 3.9.1 and 5.2 for Code Class 1 components and in Section

3.9.3 for Code Class 2 and 3 components. The conservatism in these limits and the associated ASME design requirements preclude any component structural failure.

The operability of active Code Class 1, 2, and 3 valves and active Code Class 2 and 3 pumps (there are no active Class 1 pumps) will be verified by methods detailed in Sections 3.9.1 and 5.2 for Code Class 1 components and in Section 3.9.3 for Code Class 2 and 3 components.

The use of the above stated methods provides an acceptable alternate method to meeting the guidance of this regulatory guide.

Discussion – Duke

***NOTE:** This Reg Guide was withdrawn by the NRC on 3/11/85 in 50FR 9732.*

The design limits and load combinations presented in Section 3.9 comply with the requirements of Regulatory Guide 1.48 except as noted below.

Active ASME Code Class 1 Pumps and Valves (Designed by Analysis)

For active pumps and valves, both pressure integrity and functional operability are ensured. For all portions of the pump or valve contributing to pressure retention, a stress evaluation is performed in accordance with NB-3200. The stress limits are as specified in Regulatory Position 4 for the loading combinations associated with the Normal, Upset, Emergency and Faulted Conditions.

Functional operability of the pump or valve assembly is assured by the operability program outlined in Note 1 below. If the program of Note 1 indicates that operability is not impaired when designing to the higher limits for non-active pumps and valves, those limits may be used. The effects on operability due to secondary deformations of pressure-retaining components of the assembly are accounted for by the program delineated in Note 2.

Active ASME Code Class 1 Valves (Designed by Standard of Alternative Design Rules)

As per Note 1, when analytical techniques are justified to demonstrate operability and when testing is infeasible, analysis may be used to demonstrate functional operability.

ASME Code Class 2 and 3 Vessels (Designed to Division 1 of Section VIII)

Where additional criteria are included in Section III of the Code, these are used in the qualification of Class 2 or 3 vessels.

ASME Code Class 2 Vessels (Designed to Division 2 of Section VIII)

Same as above.

ASME Code Class 2 and 3 Piping

The design limits for Class 2 and 3 piping components of the ASME Code (see Note 3) are used. Where limits for Faulted Conditions are not provided, the criteria of ASME Code Case 1606 are used.

Non-Active ASME Code Class 2 and 3 Pumps and Valves

Where provisions of the ASME Code are given, these are used to qualify nonactive Class 2 and 3 pumps and valves.

Active ASME Code Class 2 and 3 Pumps

The design limits of Regulatory Position 10 are met. Operability is assured by the program delineated in Note 1, which encompasses an appropriate combination of pre-operational testing and analysis. If the results of the operability program demonstrate that functional operability is not impaired when designing to the higher limits for non-active Class 2 and 3 pumps, these limits may be used.

Active ASME Code Class 2 and 3 Valves

The limits on the primary pressure rating of Regulatory Position 12 are met. In addition, operability is assured by the program of Note 1 which encompasses an appropriate combination of testing and analysis. If the results of the operability program demonstrate that operability is not impaired when designing to the higher limits for non-active Class 2 and 3 valves, these limits may be used.

Notes:

1. *The manufacturer is required to provide verification of operability during all design and operating conditions contained in the Design Specification. Testing or analysis or an appropriate combination of both may be acceptable. Testing may be required by the Design Specification in cases when analytical procedures are not adequate to ensure operability. Approval by Duke Power Company of the adequacy of the manufacturer's procedures must be received by the manufacturer prior to shipment of any item.*
2. *To provide assurance against secondary deformation of pressure retaining components that are unacceptable for functional operability, pre-operational and hot functional tests are performed prior to plant startup. All aspects of the operation of the pump or valve assembly are tested at full normal operating conditions. The preclusion of unacceptable secondary deformations are thus verified directly. If secondary effects are expected from other loadings that cannot be applied during these tests, these effects may be conservatively accounted for through modification of the testing and/or analytical techniques.*
3. *Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The applicable Code edition and addenda is determined by the Design Specification for the component and is not limited to Winter 1972.*
4. *Effective with the Winter 1976 Addenda of the ASME Section III Code, the terminology of component design conditions is changed. Under this change the term “Normal Operation” may be replaced with “Service Loading Condition Level A” for those items to which the Winter 1976 Addenda apply.*

   Similarly, the terms “Upset”, “Emergency”, and “Faulted” may be replaced with “Service Loading Condition Level B”, --“C”, --“D”, respectively.

   Since Regulatory Guide 1.48 applies to items subject to the Code terminology both before and after the Winter 1976 Addenda date, the terminology used herein means either terminology of the Code depending upon date under consideration.
5. *Regulatory Guide 1.48 tacitly assumes that Seismic Category I is synonymous with the nuclear-safety-related categories 1, 2, and 3 under the ANS Safety Classification system. If further directly ties its requirements to the ASME Code Section III Code Classes 1, 2, and 3. Therefore, the appropriate categories in the Duke classification system are Classes A, B, and C. Although Duke also has seismic design requirements for other classifications (e.g., Class F), Regulatory Guide 1.48 does not apply to them.*

Regulatory Guide 1.49

Power Levels of Nuclear Power Plants (Revision 1, 12/73).

Discussion

The Catawba Nuclear Station meets the recommendations of Regulatory Guide 1.49, since the projected initial power level is less than 3800 megawatts thermal (MWt) and analyses and evaluation are made at assumed core power levels less than the levels in this regulatory guide. NOTE: This Reg Guide withdrawn by the NRC on 7/5/07 in 72 FR 36737.

Regulatory Guide 1.50

Control of Preheat Temperature for Welding of Low-Alloy Steel (5/73).

Discussion

See Section 5.3.1.4 for a discussion of Westinghouse's position on this regulatory guide.

The BWI steam generator design complies with the NRC regulatory position. This NRC Regulatory Guide applies to weld fabrication for low alloy components specified in Sections III and IX of the ASME Code. For production welds, BWI does not maintain preheat temperature until post-weld heat treatment as required by regulatory position C.2. Instead either the maximum interpass temperature is maintained four hours or the minimum preheat temperature is maintained eight hours after welding. However, as permitted in C.4, this deviation from requirement C.2 is permitted since the soundness of the welds are verified by an acceptable examination procedure appropriate to the weld under consideration.

The following requirements are used by Duke for low alloy preheat temperature control when welding nuclear safety related systems at Catawba. These requirements are in lieu of those specified in Regulatory Guide 1.50.

1.0 General

All low alloy steel pipe welding shall conform to the fabrication requirements of the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 1971 or later edition and later contractual addenda. All new welding procedures and welding procedure qualifications shall conform to the requirements of the ASME, Boiler and Pressure Vessel Code, Section IX, 1971 or later edition and later addenda as specified by the Welding Procedure Qualification. Requirements other than those stated herein shall not be considered mandatory.

2.0 Definition

Low alloy steels shall be defined as those materials listed in ASME, Section IX, P-grouping numbers, P-3, P-4, and P-5.

3.0 Requirements

All requirements of Regulatory Guide 1.50 shall be met with the exception of paragraph C.2, “maintain preheat until stress relief is performed.”

Quenching or rapid cooling shall be prevented. When welding has been completed the weldment may be brought to ambient temperature by the method described in PF1 Standard ES-19 paragraphs 3.5 and 3.6. If the ambient temperature is below 32°F, preheat shall be maintained at 50°F minimum until stress relieving has been done. This will assure no detrimental effects to the metal and at the same time will allow

NDE to be performed and the proper installation of post weld stress relief equipment.

Other requirements of Regulatory Guide 1.50 are as follows:

1. Welding procedure qualification shall specify minimum preheat and maximum interpass temperature.
2. Welding procedures shall be qualified at the minimum preheat temperature.
3. Monitoring of preheat, interpass and postheat treatment temperature shall be performed.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.51

Inservice Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components (5/73).

Discussion

This regulatory guide was withdrawn by the NRC on July 15, 1975, in 40FR 30510.

Regulatory Guide 1.52

Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants (Revision 2, 3/78).

Discussion

A comparison of the design of each nuclear safety-related filter system to this regulatory guide is provided in Section 12.3.3.4.

Regulatory Guide 1.53

Application of the Single-Failure Criteria to Nuclear Power Plant Protection Systems (6/73).

Discussion - Westinghouse

Westinghouse furnished systems meet the recommendations of this regulatory guide in accordance with the comments of Section 7.1.2.4.2.

Discussion - Duke

As noted in Section 8.1.5.2, the Class 1E onsite power system, both ac and dc, have sufficient independence and redundancy to perform their safety function assuming a single failure.

Control system design for balance of plant safety and support systems includes implementation of single failure criteria. These systems are discussed in detail in Sections 7.4, 7.5, and 7.6. Westinghouse equipment is addressed in Section 7.1.

Regulatory Guide 1.54

Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants (6/73).

<u>Discussion</u>

Conformance to the requirements of this regulatory guide are discussed in Sections 6.1.2.1 and 6.1.2.2.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

<u>Regulatory Guide 1.55</u>

Concrete Placement in Category I Structures (6/73).

<u>Discussion</u>

The Catawba design and construction requirements and procedures for concrete placements in Category I structures are in conformance with the requirements of Regulatory Guide 1.55. ***NOTE:*** *This Reg Guide was withdrawn by the NCR on 7/21/81 in 46FR 37579*

<u>Regulatory Guide 1.56</u>

Maintenance of Water Purity in Boiling Water Reactors (6/73).

<u>Discussion</u>

This regulatory guide is not applicable to the Catawba Nuclear Station.

<u>Regulatory Guide 1.57</u>

Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (6/73).

<u>Discussion</u>

The design limits and load combinations given in Table 3-32 are used rather than those given in Regulatory Guide 1.57. Table 3-32 delineates acceptable design limits and appropriate combinations of loadings, which is the intent of Regulatory Guide 1.57.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

<u>Regulatory Guide 1.58</u>

Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel (8/73).

<u>Discussion</u>

Regulatory Guide 1.58 incorporates ANSI N45.2.6-1978 for both construction and operations personnel. The qualifications of station personnel are in accordance with ANSI N18.1-1971 as discussed in Section 13.1.3, rather than ANSI N45.2.6. NOTE: This Reg Guide was withdrawn by the NRC on 7/31/91 in 56 FR 36175.

<u>Regulatory Guide 1.59</u>

Design Basis Floods for Nuclear Power Plants (Revision 2, 8/77).

Discussion

Duke's position generally follows the Regulatory Position in that the flood study covers the PMF, PMF due to seismic event, and the effect of wind action. However, a PMF was not selected to meet the Corps of Engineers definition but was selected on basis of worst known area flood plus 40%. The study does not include the effects of coincident smaller event floods nor does it consider seismic event floods occurring at the recommended water levels. The water levels for the seismic event were not as high as recommended by the Regulatory Position but were at a reasonable and realistic level due to Duke Power's total control of the watershed. The flood study report does not include extensive historical flood data for use as a backup data.

Regulatory Guide 1.60

Design Response Spectra for Seismic Design of Nuclear Power Plants (Revision 1, 12/73).

Discussion - Westinghouse

The design response spectra of Regulatory Guide 1.60 are acceptable to Westinghouse as long as the damping values recommended and approved by the NRC in Reference 6 can be used in dynamic analysis of Westinghouse supplied equipment.

Discussion - BWI

BWI complies with the NRC regulatory position.

Discussion - Duke

The development of design response spectra for Catawba does not follow the procedure outlined in Regulatory Guide 1.60. The method of development for Catawba is described in Section 2.5.2.8.

Regulatory Guide 1.61

Damping Values for Seismic Design of Nuclear Power Plants (10/73).

Discussion - Westinghouse

The damping values listed in Regulatory Guide 1.61 are acceptable to Westinghouse with the single exception of the large piping systems faulted condition value of 3 percent critical. Higher damping values when justified by documented test data have been provided for in Regulatory Position C.2. A conservative value of 4 percent critical has therefore been justified by testing for the Westinghouse reactor coolant loop configuration in Reference 6 and has been approved by the NRC.

Discussion - BWI

BWI complies with the NRC regulatory position.

Discussion - Duke

The damping values given in Section 3.7.1.3 are in conformance with Regulatory Guide 1.61. The alternative values provided in Section 3.7.1.3 are in conformance with ASME Code Case N-411.

<u>*Regulatory Guide 1.62*</u>

Manual Initiation of Protective Actions (10/73).

<u>*Discussion - Westinghouse*</u>

Westinghouse furnished systems meet the recommendations of this regulatory guide in accordance with the comments of Section 7.3.2.2.7.

<u>*Discussion - Duke*</u>

Means for manual initiation of protective systems and Class 1E power systems required for safety are provided in the control room in accordance with the recommendations of Regulatory Guide 1.62.

<u>*Regulatory Guide 1.63*</u>

Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants (Revision 1, 5/77; Supplemented 2/78).

<u>*Discussion*</u>

Catawba electrical penetrations have been qualified in accordance with environmental qualification requirements of Regulatory Guide 1.63.

The mechanical, electrical, and test requirements of Regulatory Guide 1.63 for the design, construction, and installation of electric penetration assemblies in the containment structure are met subject to the clarification to Regulatory Position C.1 noted in Section 8.1.5.2.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

<u>*Regulatory Guide 1.64*</u>

Quality Assurance Requirements for the Design of Nuclear Power Plants (Revision 2, 6/76).

<u>*Discussion*</u>

The initial quality assurance program implemented by Westinghouse for the Catawba Nuclear Station was described in RESAR-3, Amendment 6, as supplemented by PSAR Chapter 17. The Westinghouse Quality Assurance Program discussed in Reference 2 was applicable to activities within Westinghouse scope performed for the Catawba Nuclear Station which were initiated between January 1, 1975 and October 1, 1977. Subsequently, the present Westinghouse Quality Assurance Program, which is described in Reference 3, is applicable to activities within Westinghouse scope which were initiated after October 1, 1977.

Duke Power conforms to Regulatory Guide 1.64 (Rev. 2) except that the following two sentences are added to Paragraph C.2:

The use of the originator's immediate supervisor for design verification shall be restricted to special situations where the immediate supervisor is the only individual within the design organization competent to perform the verification. Justification for such use shall also be documented along with the extent of the supervisor's input into the design as specified in the Design Engineering Department Quality Assurance Program Manual. NOTE: This Reg Guide was withdrawn by the NRC on 7/31/91 in 56 FR 36175.

Regulatory Guide 1.65

Materials and Inspections for Reactor Vessel Closure Studs (Revision 1, 4/2010).

Discussion - Westinghouse

Westinghouse follows the recommendations of Regulatory Guide 1.65 subject to the exceptions discussed in Section 5.3.1.7.

Discussion - Duke

Conformance of the optional Nova HydraNuts (Engineering Change EC104610) to the requirements of Regulatory Guide 1.65 (Revision 1, 4/20/2010) is discussed in Section 5.3.1.7.1.

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

Regulatory Guide 1.66

Nondestructive Examination of Tubular Products (10/73).

Discussion

This regulatory guide was withdrawn by the NRC on September 28, 1977, in 42FR 54478

Regulatory Guide 1.67

Installation of Overpressure Protection Devices (10/73).

Discussion

The Catawba design conforms to the requirements of Regulatory Guide 1.67 as discussed in Section 3.9.3.3. ***NOTE:*** *This Reg Guide was withdrawn by the NRC on 4/27/83 in 48FR 19101*

Regulatory Guide 1.68

Initial Test Programs for Water-Cooled Reactor Power Plants (Revision 2, 8/78).

Discussion

The testing activities conducted as part of the initial test program comply with the intent of Regulatory Guide 1.68 with exceptions as noted in Section 14.2.7.

Regulatory Guide 1.68.1

Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants (Revision 1, 1/77).

Discussion

This regulatory guide is not applicable to the Catawba Nuclear Station.

Regulatory Guide 1.68.2

Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water Cooled Nuclear Power Plants (Revision 1, 7/78).

Discussion

The initial startup test program for Catawba complies with the requirements of Regulatory Guide 1.68.2.

Regulatory Guide 1.69

Concrete Radiation Shields for Nuclear Power Plants (12/73).

Discussion

The design of the Catawba concrete radiation shields conforms to the requirements of Regulatory Guide subject to the following exceptions to Sections 5.1.1, 6.2.2, 7.1.5, 8.1.8, 8.7.1, 8.7.2, 11.2, 11.3, 11.4.1, and 11.4.2 of ANSI N101.6-1972 “Concrete Radiation Shields.”

C.9 ANSI N101.6, Section 6.2.2 - Strength of Forms - The design and engineering of formwork shall be the responsibility of the contractor. When necessary, the Construction Department shall require that design calculations and shop drawings for forms be prepared for their approval prior to the start of formwork construction.

C.10 ANSI N101.6, Section 8.1.8 - General Requirements for Placement of Concrete. Tests are performed on aggregates at pre-determined intervals but samples are not bagged and retained for permanent records.

C.11 ANSI N101.6, Section 8.7.1 - Construction joints. Construction joints shall be located as shown on the design drawings. Any variation of construction joint location from the drawings shall be approved by Design Engineering. Reinforcing steel details and keys at joints shall be in accordance with the design drawings. The surface of the concrete at joints shall be thoroughly cleaned and all laitance removed. Joints shall not be bonded with grout except as specified on drawings.

C.12 ANSI N101.6, Section 11.3 - Preconstruction Concrete Test. Trial proportioning tests to determine the effects of variables which may affect the hardened concrete shall be performed by the Construction personnel in the QC laboratory.

C.13 ANSI N101.6, Section 5.1.5 - Aggregate for Normal Concrete. Fine aggregates shall conform to the requirements of ASTM C33-74a. Coarse aggregates shall conform to one of the following:

1. ASTM C33-74a or
2. “General Requirements for Aggregate” and “Aggregate for Portland Cement Concrete,” Section 905 (Revised July 29, 1975) and Section 914-2 (Revised October 5, 1976), respectively, of the 1972 North Carolina Department of Transportation's “Standard Specification for Roads and Structures,” or
3. “Coarse Aggregate for Portland Cement Concrete for Structures” Section 701.07 and page 588 of the 1973 South Carolina State Highway Department's “Standard Specifications for Highway Construction.”

C.14 ANSI N101.6, Section 7.1.5 - Design Requirements for Embedments. Project Specifications require that embedments conform to design drawings, purchase order specifications and 6.0 of ACI 301-72, "Specifications for Structural Concrete for Buildings."

C.15 ANSI N101.6, Section 8.7.2 - Curing. Concrete shall be cured and protected in accordance with Chapter 12 of "Specifications for Structural Concrete for Buildings," ACI 301-72.

As an alternative, the protection and moisture retention and requirements of Sections 12.2.3 and 12.3.1 may be waived when the following procedure is followed.

When the mean daily outdoor temperature is less than 40°F additional cylinders shall be cast and cured at the form under the same protective and curing conditions as the placed concrete. After placement, the temperature of the concrete shall be maintained above 50°F for a minimum period of 3 days. At the end of this 3-day period 2 cylinders shall be tested and if their average compressive strength equals or exceeds 500 psi, cold weather temperature protective measures may be discontinued. Moisture retention measures, however, shall continue until an additional 4 days of curing is achieved where the average mean daily outdoor temperature is 40°F or above or until tests of remaining cylinders show that the average compressive strength of the concrete has reached 70% of the specified 28-day compressive strength.

The form wetting requirements of Section 12.2.2 of ACI 301-72 may be waived when watertight forms are used.

Unless otherwise indicated on design drawings, formwork may be removed as follows: The temperature or ranges of temperature are the average mean daily temperature of the ambient air. For temperatures less than 40°F, the curing and protection requirements stated above shall govern the removal of side forms. Shoring and bottom forms shall not be removed at temperatures less than 40°F.

Type of Formwork	Time Framework to Remain in Place (Average mean daily temperature of ambient air)		
	60F or greater	**50 to 60F**	**40 or 50F**
Side Forms[1]	18 hours	36 hours	2 days
Side Forms[2]	36 hours	3 days	5 days
Bottom Forms and shoring[3]	14 days	18 days	24 days

Notes:

1. For walls, columns, beams and girders whose least lateral dimension is greater than 12"
2. For walls, columns, beams and girders whose least lateral dimension is less than 12".
3. In addition, the compressive strength of the concrete must be equal to or greater than 70% of the 28-day compressive strength.

C.16 ANSI N101.6, Section 11.2 - Materials Test and Inspection, Table 2-Mixing Water. Mixing water and mixing water used for ice shall be evaluated by performing the following tests:

1. Soundness, in accordance with "Autoclave Expansion of Portland Cement," ASTM C151-74a. The results obtained for the proposed mixing water shall not be increased by more than +0.10 of those obtained for distilled water.
2. Time of setting, in accordance with "Time of Setting Hydraulic Cement by Vicat Neddle," ASTM C191-74. The results obtained for the proposed mixing water shall be not less than 45 minutes for initial setting time and not more than 8 hours for final setting time.
3. Compressive strength, in accordance with "Compressive Strength of Hydraulic Cement Mortars (using 2 in. Cube Specimens)," ASTM C109-73. The 7 and 28-day strengths of mortar cubes made with the proposed mixing water shall not be lower by more than 10 percent of the strengths obtained for mortar cubes made with distilled water.

Tests shall be performed semi-annually.

C.17 ANSI N101.6, Section 11.4.1 - Concrete Control Test During Construction. Samples of fresh concrete shall be obtained in accordance with "Method of Sampling Fresh Concrete," ASTM C172-71. The compressive strength of the concrete shall be determined in accordance with the following:

Six compressive strength test specimens shall be made from a sample of fresh concrete obtained for each 100 cubic yards, or fraction thereof, of concrete placed in any one day for each mix designation.
The test specimens shall be made and cured in accordance with "Method of Making and Curing Concrete Compression and Flexure Test Specimens in the Field," ASTM C31-69.
The test specimens shall be tested in accordance with "Method of Test for Compressive Strength of Molded Concrete Cylinders," ASTM C39-72, except that paragraph 4.3 and 6.1.6 shall apply only when the test specimens fail to meet the specified 28-day compressive strength requirements. Two test specimens shall be tested at the age of 7 days and 28 days. The remaining two test specimens are spares and may be tested if deemed necessary.
The strength level of the concrete will be considered satisfactory so long as the averages of all sets of three consecutive strength test results equal or exceed the specified compressive strength, and no individual strength test result falls below the specified compressive strength by more than 500 psi. If concrete should fail to meet these strength requirements, Design Engineering shall be notified of the test results for design review.
Standard deviation data shall be developed for each mix in accordance with "Recommended practice for Evaluation of Compression Test Results of Field Concrete," ACI 214-65 and maintained by the Construction Department.

C.18 ANSI N101.6, Section 11.4.2 - Unit Weight Tests. The unit weight and yield of the freshly mixed concrete shall be determined monthly during production.

Regulatory Guide 1.70

Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition (Revision 3, 11/78).

Discussion

The Catawba FSAR conforms to Revision 3 of Regulatory Guide 1.70, with exceptions as noted in Section 3.9.1.1.

Regulatory Guide 1.71

Welder Qualification for Areas for Limited Accessibility (12/73).

Discussion

Compliance with the requirements of this regulatory guide are discussed in Section 5.3.1.4 and Section 10.3.6.2.

Regulatory Guide 1.72

Spray Pond Piping made from Fiberglass-Reinforced Thermosetting Resin.

Discussion

Catawba does not have a spray pond cooling system; therefore, this regulatory guide is not applicable.

Regulatory Guide 1.73

Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (1/74).

Discussion

For safety-related motor operated valves located inside containment, environmental qualification is performed in accordance with IEEE Standard 382-1972. Qualification conditions (temperature, pressure, radiation, and chemistry) are those specified in Part III of IEEE Standard 382-1972 for pressurized water reactor applications. Since there are no exposed organic materials, consideration of beta radiation is not required.

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

Regulatory Guide 1.74

Quality Assurance Terms and Definitions (2/74).

Discussion

This regulatory guide incorporated ANSI N45.2.10-1973. Some definitions used by Duke are worded differently than those in this standard; however, the general meanings are the same. ***NOTE:*** *This Reg Guide was withdrawn by the NRC on 9/21/81 in 54FR 38919*

Regulatory Guide 1.75

Physical Independence of Electric Systems (Revision 2, 9/78).

Discussion

Westinghouse furnished systems meet the recommendations of this regulatory guide in accordance with the comments of Section 7.1.2.2.

Regulatory Guide 1.76

Design Basis Tornado for Nuclear Power Plants (4/74).

Discussion

The Catawba design conforms to the requirements of this regulatory guide with exceptions as noted in Section 3.3.2.1. Also see Sections 3.5.1.4 and 3.5.3.1.

Regulatory Guide 1.77

Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (5/74).

Discussion

Duke analyses the Control Rod Ejection Accident in Section 15.4.8.3.

On September 30, 2005, the NRC staff approved full scope implementation of the method Alternative Source (AST) at Catawba Nuclear Station (Ref. 19) based on part on their review of the AST analysis of the Rod Ejection Accident at Catawba (Ref. 24-29). This analysis no longer conforms to Regulatory Guide 1.77. Instead, it conforms in general to Regulatory Guide 1.183 and Appendix H. Any exceptions are noted in the submittal of the AST analysis of the Rod Ejection Accident (Ref. 24).

Regulatory Guide 1.78

Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (6/74).

Discussion

Conformance to this regulatory guide is discussed in Table 6-101

Regulatory Guide 1.79

Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors (Revision 1, 9/75).

Discussion

The initial test program for Catawba is in compliance with this regulatory guide except as noted in Section 14.2.7.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.80

Preoperational Testing of Instrument Air Systems (6/74).

Discussion

The initial test program for Catawba is in compliance with this regulatory guide except as noted in Section 14.2.7. ***Note:*** *This Reg Guide was withdrawn by the NRC on 5/4/82 and reissued as Reg Guide 1.68.3.*

Regulatory Guide 1.81

Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants (Revision 1, 1/75).

Discussion

The recommendations of Regulatory Guide 1.81 are not applicable to Catawba based on the implementation date of the guide. The intent of this guide is met in that the Class 1E ac and dc power systems are not shared between units. Capability is provided to connect a source of preferred power from either unit to the Class 1E ac power system of the remaining unit as discussed in Section 8.3.1.1.2.1.

Regulatory Guide 1.82

Sumps for Emergency Core Cooling and Containment Spray Systems (6/74).

Discussion

The Containment Recirculation Sump at Catawba is designed to fully meet the regulatory positions of the regulatory guide with modifications to positions C.4, 6, and 7 as stated below.

C.4 The floor level in the vicinity of the coolant sump location should not slope down toward the sump.

C.6 An outer trash rack should be provided to prevent large debris from reaching the fine inner screen. The strength of the trash rack should be considered in protecting the inner screen from missiles and large debris.

C.7 The design coolant velocity at the fine inner screen should be approximately 2.0 ft/sec. The available surface area used in determining the design coolant velocity should be based on one-half of the free surface area of the fine inner screen to conservatively account for partial blockage. No horizontal screen should be considered in determining available surface area.

Upon completion of the ECCS sump strainer assembly modifications during outage 2EOC15 for Unit 2 and 1EOC17 for Unit 1, the following Discussion section will apply:

Discussion

The Containment Recirculation Sump at Catawba is designed to fully meet the regulatory positions of the regulatory guide with modifications to positions as shown below:

C.1 Catawba will utilize the containment side structure and floor as the intake structure boundary since an acceptable post-LOCA water level in the containment is achievable. Thus, making additional sump depressions in the floor is non-productive. Redundance will be provided by two separate suction pipes.

C.2 The containment recirculation intake structure and suction piping are protected from high energy piping systems to the extent practical to preclude damage by whipping pipes or high-velocity jets of water or steam. ECCS redundancy begins at the sump suction pipes, and the need to provide ECCS/CSS train separation within the common sump strainer is not required in the absence of any credible loads which could fail the sump strainer.

C.3 The sump is located on the lowest floor elevation in the containment exclusive of the reactor vessel cavity. A substantial strainer is provided to filter debris from recirculated coolant. The polar crane wall acts as a primary filter to prevent large debris from reaching the sump strainer assembly.

C.4 Exception is taken to this position.

C.6 The location of the sump strainer assembly provides a protection from missiles and large debris. The polar crane wall acts as a primary filter to prevent large debris from reaching the sump strainer.

C.7 The sump strainer design (i.e., size and shape) will preclude the loss of NPSH to ECCS and CSS pumps from debris blockage during the period that the ECCS is required to operate and maintain long-term cooling.

C.8 Vortex suppression is provided to preclude air entrainment in the recirculated coolant.

C.9 The sump strainer is designed to withstand the vibratory motion of seismic events without loss of structural integrity.

C.10 The size of openings in the sump strainer are based on the minimum restrictions found in systems served by the sump. The minimum restriction takes into account the overall operability of the system served.

C.12 Materials for the sump strainers were selected to avoid degradation during periods of inactivity and operation and have a low sensitivity to adverse effects such as stress assisted corrosion that may be induced by chemically reactive spray during LOCA conditions.

C.13 The sump strainer includes access openings to facilitate inspection to the extent practical. Physical restrictions in the area the strainer is located limits access for inspection without disassembly.

C.14 Inservice inspection requirements for coolant sump components (the strainer assembly) include the following:

a) Coolant sump components shall be inspected during every refueling period downtime, and

b) The inspection shall be a visual examination of the components for evidence of structural distress or corrosion to the extent practical. Physical restrictions in the area the strainer is located limits access for inspection without disassembly.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.83

Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes (Revision 1, 7/75).

Discussion

Westinghouse and BWI steam generators are designed to permit access to tubes for inspection and/or plugging. The inservice inspection program is discussed in the Technical Specifications.

The BWI steam generator design complies with the regulatory position with the following clarifications:

The Regulatory Guide addresses both new and in-service components. The RSGs are new components and as such comply with the appropriate sections of this regulatory guide. Specifically C.1.a, C.1.b, C.2, C.3.a, and C.4.a. A 100 percent baseline inspection of the RSG is performed prior to the unit being put into service. BWI acceptance criteria exceeds the NRC guidelines for wall thickness reductions in that BWI limits wall thickness reductions to no more than 15% versus 20% allowed in the NRC guidelines. NOTE: This Reg Guide was withdrawn by the NRC on 11/12/09 in 74 FR 58324.

Regulatory Guide 1.84

Code Case Acceptability - ASME Section III Design and Fabrication (Revision 16, 5/80).

Discussion - Westinghouse

1. Westinghouse controls its suppliers to:

   a. Limit the use of code cases to those listed in Regulatory Position C.1 of the Regulatory Guide 1.84 and 1.85 revision in effect at the time the equipment is ordered, except as allowed in item b. below.

   b. Identify and request permission for use of any code cases not listed in Regulatory Position C.1 of the Regulatory Guide 1.84 and 1.85 revision in effect at the time the equipment is ordered, where use of such code cases is needed by the supplier.

   c. Permit continued use of a code case considered acceptable at the time of equipment order, where such code case was subsequently annulled or amended.

2. Westinghouse seeks NRC permission for the use of Class 1 code cases needed by suppliers and not yet endorsed in Regulatory Position C.1 of the Regulatory Guide 1.84 and 1.85 revision in effect at the time the equipment is ordered and permits supplier use of these code cases only if NRC permission is obtained or is otherwise assured (e.g., a later version of the regulatory guide includes endorsement).

Discussion - Duke

Same as Westinghouse position.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.85

Code Case Acceptability - ASME Section III Materials (Revision 17, 12/80).

Discussion

Refer to the discussion of Regulatory Guide 1.84. *NOTE: This Reg Guide was withdrawn 6/1/03 by the NRC and incorporated into Reg Guide 1.84.*

Regulatory Guide 1.86

Termination of Operating Licenses for Nuclear Reactors (6/74).

Discussion

The termination of the operating license and subsequent decommissioning of the Catawba Nuclear Station will address the regulations in effect at that time.

Regulatory Guide 1.87

Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596) (Revision 1, 6/75).

Discussion

This regulatory guide is not applicable to the Catawba Nuclear Station.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.88

Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records (Revision 2, 10/76).

Discussion – Westinghouse

NOTE: *This Reg Guide was withdrawn by the NRC on 7/31/91 in 56FR 36175.*

This regulatory guide recognizes ANSI N45.2.9-1974. For the Catawba Nuclear Station, the Westinghouse Quality Assurance Program follows the guidance of this standard. Records are identified, indexed, stored, and protected in a manner consistent with Regulatory Guide 1.88.

For active files, Westinghouse maintains duplicate records in separate geographical locations as an alternative to the protection construction provisions of the standard.

Discussion – Duke

NOTE: *This Reg Guide was withdrawn by the NRC on 7/31/91 in 56FR 36175.*

The quality assurance program for Catawba complies with the requirements of Regulatory Guide 1.88 as noted in Chapter 17.

Regulatory Guide 1.89

Qualification of Class 1E Equipment for Nuclear Power Plants (11/74).

Discussion

By virtue of its implementation date this Regulatory Guide is not applicable to the Catawba Nuclear Station. Seismic and environmental qualification of electrical equipment is discussed in Sections 3.10 and 3.11, respectively.

Regulatory Guide 1.90

Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons (Revision 1, 8/77).

Discussion

Catawba has a freestanding steel containment; therefore, Regulatory Guide 1.90 is not applicable.

Regulatory Guide 1.91

Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants (Revision 1, 2/78).

Discussion

The evaluation of postulated explosions on transportation routes near the Catawba Nuclear Station are in conformance with the recommendations of Regulatory Guide 1.91.

Regulatory Guide 1.92

Combining Modal Responses and Spatial Components in Seismic Response Analysis (Revision 1, 2/76).

Discussion

The Westinghouse procedure for combining modal response is presented in Sections 3.7.2.6 and 3.7.2.7. This procedure is considered as an alternate acceptable solution which provides a basis for findings requisite to issuance of an Operating License by the NRC.

The BWI steam generator design complies with the regulatory position. Modal responses which are not closely spaced (greater than 10 percent) are combined by using the square root of the sum of the squares (SRSS) method to determine the maximum response for the seismic direction being considered. Dynamic systems that exhibit closely spaced modes (if any) are analyzed in accordance with the Regulatory Guide position.

The provisions of Regulatory Guide 1.92 were not utilized in the design of the structures at Catawba. The Catawba design combines the responses of one of the two horizontal directions simultaneously with motion in the vertical direction; whereas the method described in Regulatory Guide 1.92 combines all three directions (two horizontal and one vertical) simultaneously. The methods used by Duke for combining modal responses are explained in detail in Section 3.7.2.

Regulatory Guide 1.93

Availability of Electric Power Sources (12/74).

Discussion

The recommendations of Regulatory Guide 1.93 are not applicable to Catawba based on the implementation date of the guide. The availability of electric power sources with respect to limiting conditions for operation is presented in the Technical Specifications.

Regulatory Guide 1.94

Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (Revision 1, 4/76).

Discussion

The quality assurance program for Catawba complies with the requirements of Regulatory Guide 1.94 as noted in Chapter 17.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.95

Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release (Revision 1, 1/77).

Discussion

Conformance to this regulatory guide is discussed in Table 6-100. Regulatory Guide 1.96 Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants (Revision 1, 6/76). NOTE: This Reg Guide was withdrawn by the NRC on 1/1/02. See Reg Guide 1.78 for guidance.

Regulatory Guide 1.96

Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants (Revision 1, 6/76).

Discussion

This regulatory guide is not applicable to the Catawba Nuclear Station.

Regulatory Guide 1.97

Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident (Revision 2, 12/80).

Discussion

A discussion of this regulatory guide is provided in the Duke Power Company Response to Supplement 1 to NUREG-0737.

Regulatory Guide 1.98

Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor (3/76).

Discussion

This regulatory guide is not applicable to the Catawba Nuclear Station.

Regulatory Guide 1.99

Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials (Revision 1, 4/77).

Discussion

Deleted per 2004 update.

The NRC issued Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Materials and Its Impact on Plant Operations," on July 12, 1988. The purpose of Generic Letter 88-11 was to inform licensees of the promulgation of Revision 2 to R.G. 1.99 and to request applicable revisions to plant operating basis documents in accordance with R.G. 1.99, Revision 2 methodologies. Duke Power Co. responded to this request by a letter from H.B. Tucker to the NRC, dated November 28, 1988. Since that time, Catawba Station has fully complied with the requirements of R.G. 1.99, Revision 2 with regard to determination of beltline radiation embrittlement and application of such data to plant operating limits. UFSAR Section 5.3.2 contains further descriptions of Reactor Vessel Pressure-Temperature Limits and their implementation.

Regulatory Guide 1.100

Seismic Qualification of Electric Equipment for Nuclear Power Plants (Revision 1, 8/77).

Discussion

The seismic qualification of Category I instrumentation and electrical equipment is discussed in Section 3.10.1.1 and Reference 8.

Regulatory Guide 1.101

Emergency Planning for Nuclear Power Plants (Revision 1, 3/77).

Discussion

This Regulatory Guide has been superseded by NUREG-0654, Revision 1.

Regulatory Guide 1.102

Flood Protection for Nuclear Power Plants (Revision 1, 9/76).

Discussion

Duke's position generally follows the requirements of this regulatory guide. Referring to Regulatory Position C.1.c, there are other types of sealing arrangements that may be used to provide acceptable flood protection at pipe penetrations in addition to the two examples mentioned.

Flood protection is discussed in Section 3.4.1.

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

Regulatory Guide 1.103

Post-Tensioned Prestressing Systems for Concrete Reactor Vessels and Containments (Revision 1, 10/76).

Discussion

Catawba has a freestanding steel containment; therefore, Regulatory Guide 1.103 is not applicable. ***Note:*** *This Reg Guide was withdrawn by the NRC on 7/21/81 in 46FR 37579.*

Regulatory Guide 1.104

Overhead Crane Handling Systems for Nuclear Power Plants (2/76).

Discussion

This regulatory guide was withdrawn by the NRC on 8/22/79 in 44FR 49321.

Regulatory Guide 1.105

Instrument Setpoints (Revision 1, 11/76).

Discussion

The Technical Specifications provide the margin from the nominal setpoint to the Technical Specification limit to account for drift when measured at the rack during periodic testing. The allowances between the Technical Specification limit and the safety limit include the following items: a) the inaccuracy of the instrument, b) process measurement accuracy, c) uncertainties in the calibration, d) the potential transient overshoot determined in the accident analyses (this may include compensation for the dynamic effect), and e) environmental effects on equipment accuracy caused by postulated or limiting postulated events (only for those systems required to mitigate consequences of an accident). Westinghouse designers choose setpoints such that the accuracy of the instrument is adequate to meet the assumptions of the safety analysis.

The range of instruments is chosen based on the span necessary for the instrument's function. Narrow range instruments will be used where necessary. Instruments will be selected based on expected environmental and accident conditions. The need for qualification testing will be evaluated and justified on a case-by-case basis.

Administrative procedures coupled with the present cabinet alarms and/or locks provide sufficient control over the setpoint adjustment mechanism such that no integral setpoint securing device is required. Integral setpoint locking devices will not be supplied.

The assumptions used in selecting the setpoint values in Regulatory Position C.1 and the minimum margin with respect to the technical specification limit and calibration uncertainty will be documented by Westinghouse. Drift rates and their relationship to testing intervals will not be documented by Westinghouse.

Regulatory Guide 1.106

Thermal Overload Protection for Electric Motors on Motor-Operated Valves (Revision 1, 3/77).

Discussion

Compliance with Regulatory Guide 1.106 is discussed in Section 8.1.5.2.

Regulatory Guide 1.107

Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures (Revision 1, 2/77).

Discussion

Catawba has a freestanding steel containment; therefore, Regulatory Guide 1.107 is not applicable.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.108

Periodic Testing of Diesel Generators Used as Onsite Electric Power Systems at Nuclear Power Plants (Revision 1, 8/77; Supplemented 9/77).

Discussion

The periodic testing requirements for the safety-related diesel generators are presented in Technical Specifications. ***Note:*** *This Reg Guide was withdrawn by the NCR on 8/5/93 in 58FR 41813.*

Regulatory Guide 1.109

Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I (Revision 1, 10/77).

Discussion

Compliance with Appendix I was evaluated in accordance with this regulatory guide as discussed in Sections 11.2.3.3 and 11.3.3.4.

Regulatory Guide 1.110

Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (3/76).

Discussion

Since the Catawba construction permit application was docketed on October 27, 1972, dose limits are taken from the annex to Appendix I, 10 CFR 50, and a radwaste cost-benefit analysis is not performed per the option available in Paragraph II.D.

Regulatory Guide 1.111

Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors (Revision 1, 7/77).

Discussion

Duke utilizes the XOQDOQ air dispersion model to implement Reulatory Guide 1.111, as discussed in NUREG/CR-2919 (1982) for long term (routine annual) impacts.

Regulatory Guide 1.112

Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors (Revision O-R, 4/76; Reissued 5/77).

Discussion

Regulatory Guide 1.112 incorporates, NUREG-0017 by reference. Duke complies with the requirements of this regulatory guide with the following exceptions:

a. To increase the usefulness of the PWR-GALE code, the mixing efficiency (Section 1.5.2.21, NUREG-0017) and gas stripping fraction on the letdown stream (Section 1.5.2.16, NUREG-0017) will become actual code inputs.
b. Tritium production and release (Section 2.2.19, NUREG-0017) will be calculated by Duke. The model used will include sources from boron, lithium, tertiary fission, and deuterium. Loss terms will be leakage, decay, and feed and bleed operations.

Regulatory Guide 1.113

Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I (Revision 1, 4/77).

Discussion

Dispersion of accidental releases of liquid effluents in surface waters is discussed in Section 2.4.12.

The methodology for calculating the potential annual average radiation doses to the public that may result from radioactive material in liquid effluents routinely released to surface water bodies is discussed in Selected Licensee Commitment (SLC) 16.11-3.

Regulatory Guide 1.114

Guidance on Being Operator at the Controls of a Nuclear Power Plant (Revision 1, 11/76).

Discussion

Duke complies with the requirements of this regulatory guide. Figure 13-6 defines the area designated “at the controls.”

Regulatory Guide 1.115

Protection Against Low-Trajectory Turbine Missiles (Revision 1, 6/77).

Discussion

Low-trajectory turbine missiles are discussed in Section 3.5.1.3.4.2.

Regulatory Guide 1.116

Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (Revision O-R, 6/76; Reissued 5/77).

Discussion

The quality assurance program for Catawba complies with the requirements of Regulatory Guide 1.116 as discussed in Chapter 17.

Regulatory Guide 1.117

Tornado Design Classification (Revision 1, 4/78).

Discussion

The implementation date for this regulatory guide (construction permit applications docketed after May 30, 1978) is after the October 27, 1972 construction permit docket date for the Catawba Nuclear Station and, therefore, this regulatory guide will not be addressed.

Regulatory Guide 1.118

Periodic Testing of Electric Power and Protection Systems (Revision 2, 6/78).

Discussion

The periodic testing requirements of the electric power and protection systems are presented in the Technical Specifications. These testing requirements are also discussed in section 7.1.2.4.2.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.119

Surveillance Program for New Fuel Assembly Designs (6/76).

Discussion

This regulatory guide was withdrawn by the NRC on June 30, 1977, in 42FR 33387.

Regulatory Guide 1.120

Fire Protection Guidelines for Nuclear Power Plants (Revision 1, 11/77).

Discussion

As discussed in Section 9.5.1.3, the fire protection evaluation for Catawba was performed in response to BTP APCBS 9.5-1. NOTE: This Reg Guide withdrawn by the NRC on 8/1/01.

<u>*Regulatory Guide 1.121*</u>

Bases for Plugging Degraded PWR Steam Generator Tubes (8/76).

<u>*Discussion*</u>

Exceptions to this regulatory guide are discussed in Section 5.4.2.5.4.

<u>*Regulatory Guide 1.122*</u>

Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components (Revision 1, 2/78).

<u>*Discussion*</u>

The implementation date for this regulatory guide is after the October 27, 1972 construction permit docket date for the Catawba Nuclear Station and, therefore, this regulatory guide will not be addressed.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

<u>*Regulatory Guide 1.123*</u>

Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants (Revision 1, 7/77).

<u>*Discussion*</u>

The initial quality assurance program implemented by Westinghouse for the Catawba Nuclear Station was described in RESAR-3, Amendment 6, as supplemented by PSAR Chapter 17. The Westinghouse Quality Assurance Program discussed in Reference 2 was applicable to activities within Westinghouse scope performed for the Catawba Nuclear Station which were initiated between January 1, 1975 and October 1, 1977. Subsequently, the present Westinghouse Quality Assurance Program, which is described in Reference 3, is applicable to activities within Westinghouse scope which were initiated after October 1, 1977.

The quality assurance program for Catawba complies with the requirements of Regulatory Guide 1.123 as discussed in Chapter 17. ***Note:*** *This Reg Guide was withdrawn by the NRC on 7/31/91 in 56FR 36175.*

<u>*Regulatory Guide 1.124*</u>

Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports (Revision 1, 1/78).

<u>*Discussion*</u>

The implementation date for this regulatory guide (construction permit applications docketed after January 10, 1978) is after the October 27, 1972 construction permit docket date for the Catawba Nuclear Station and, therefore, this regulatory guide will not be addressed.

<u>*Regulatory Guide 1.125*</u>

Physical Models for Design Operation of Hydraulic Structures and Systems for Nuclear Power Plants (Revision 1, 10/78).

Discussion

The implementation date for this regulatory guide (construction permit applications docketed after October 1978) is after the October 27, 1972 construction permit docket date for the Catawba Nuclear Station and, therefore, this regulatory guide will not be addressed.

Regulatory Guide 1.126

An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification (Revision 1, 3/78).

Discussion

Westinghouse uses the fuel densification model presented in Reference 11 which has been approved by the NRC.

Regulatory Guide 1.127

Inspection of Water-Control Structures Associated with Nuclear Power Plants (Revision 1, 3/78).

Discussion

Duke complies with the requirements of the regulatory guide with the following exceptions:

C.2 - Onsite Inspection Program

a. Concrete Structures in General

(5) Drains-Foundation, Joint, Face. All drains should be examined to the extent possible for the purpose of ensuring that they are capable of performing their design function.

(8) Monolithic Joints-Construction Joints. All monolithic construction joints where practical should be examined to determine the condition of the joint and filler material, any movement of joints, or any indication of distress or leakage.

b. Embankment Structures

These procedures do not apply to structures which are submerged under water most of the time. Such structures may be inspected to the extent possible, by divers at about 5 year intervals.

d. Reservoirs

(4) Watershed Runoff Potential. The drainage basin should be examined where practical for any extensive recent alternations to the surface of the drainage basin such as changed agricultural practices, timber clearing, railroad or highway construction, or real estate developments that might adversely affect the runoff characteristics. Upstream projects that could have an impact on the safety of the dam should be identified.

C.5 Inspection Report

b. Subsequent Reports (Second Paragraph)

> The inspection should be conducted under the direction of engineers experienced in the investigation, design, construction, and operation of these types of facilities. The field inspection team should include engineers, engineering geologists, or other specialists able to recognize and assess signs of possible distress (e.g., structural joint movement, piezometric fluctuations, seepage variations, settlement and horizontal misalignments, slope movement, cracking of concrete, erosion, and corrosion of equipment and conduits) and able to recommend appropriate mitigating measures.

Regulatory Guide 1.128

Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants (Revision 1, 10/78).

Discussion

Compliance with Regulatory Guide 1.128 is discussed in Section 8.1.5.2.

Regulatory Guide 1.129

Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants (Revision 1, 2/78).

Discussion

The Technical Specifications surveillance requirement for demonstrating the operability of the 125 VDC vital instrumentation and control power batteries is in accordance with the recommendations of Regulatory Guide 1.129.

Regulatory Guide 1.130

Design Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports (Revision 1, 10/78).

Discussion

The implementation date for this regulatory guide (construction permit applications docketed after April 1, 1978) is after the October 27, 1972 construction permit docket date for the Catawba Nuclear Station and, therefore, this regulatory guide will not be addressed.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.131

Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants (8/77).

Discussion

The qualification testing of electric cables, field splices, and connections is discussed in Section 3.11. NOTE: This Reg Guide was withdrawn by the NRC on 4/20/09 in 74 FR 18000.

Regulatory Guide 1.132

Site Investigations for Foundations of Nuclear Power Plants (Revision 1, 3/79).

Discussion

The implementation date for this regulatory guide (construction permit applications docketed after March 30, 1979) is after the October 27, 1972 construction permit docket date for the Catawba Nuclear Station and, therefore, this regulatory guide will not be addressed.

Regulatory Guide 1.133

Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors (Revision 1, 5/81).

Discussion

The Loose Parts Detection System for Catawba is in compliance with the requirements of Regulatory Guide 1.133, Revision 1. This system is described in Section 7.8.8.

Regulatory Guide 1.134

Medical Certification and Monitoring of Personnel Requiring Operator Licenses (Revision 1, 3/79).

Discussion

The medical certification and monitoring requirements of licensed personnel comply with the requirements of this regulatory guide.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.135

Normal Water Level and Discharge at Nuclear Power Plants (9/77).

Discussion

The implementation date for this regulatory guide (construction permit applications docketed after May 1, 1978) is after the October 27, 1972 construction permit docket date for the Catawba Nuclear Station and, therefore, this regulatory guide will not be addressed. NOTE: This Reg Guide was withdrawn by the NRC on 8/6/2009 in 74 FR 39349.

Regulatory Guide 1.136

Material for Concrete Containments (Revision 1, 10/78).

Discussion

The implementation date for this regulatory guide (construction permit applications docketed after November 30, 1978) is after the October 27, 1972 construction permit docket date for the Catawba Nuclear Station and, therefore, this regulatory guide will not be addressed.

Regulatory Guide 1.137

Fuel-Oil Systems for Standby Diesel Generators (Revision 1, 10/79).

Discussion

A description of the Diesel Generator Engine Fuel Oil System is provided in Section 9.5.4.

The Technical Specifications surveillance requirement for demonstrating the operability of the diesel generators is in accordance with the recommendations of this regulatory guide. Technical Specification Task Force (TSTF) 2, Rev.1 exceptions for the 10-year tank sediment cleaning and inspections of the diesel fuel tanks were added per license amendments 206/200. This allowed the tank sediment cleaning and inspection requirements to be moved to licensee-controlled documents.

Regulatory Guide 1.138

Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants (4/78).

Discussion

Laboratory investigations and testing practices for soils at Catawba comply with the requirements of Regulatory Guide 1.138 as discussed in Section 2.5.4.2.2.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.139

Guidance for Residual Heat Removal (5/78).

Discussion

The Catawba position on Branch Technical Position RSB 5-1 is provided in Section 5.4.7.2.6. NOTE: The Reg Guide was withdrawn by the NRC on 6/10/08 in 73 FR 32750.

Regulatory Guide 1.140

Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Absorption of Light-Water-Cooled Nuclear Power Plants (Revision 1, 10/79).

Discussion

The Catawba design, maintenance and testing of the Normal Ventilation Exhaust System Air Filtration Units are in compliance with the requirements of the regulatory guide with the following clarification:

C.3.1 System damper construction follows the guidelines of ANSI N509-1980 Table 5-2.

Table 14-1 discusses exceptions to Regulatory Positions C.3, C.5, and C.6.

Regulatory Guide 1.141

Containment Isolation Provisions for Fluid Systems (4/78).

Discussion

Exception is taken to Section 3.6.4 of ANSI N271-1976 as referenced in Regulatory Guide 1.141, Position C.1 as follows:

It is Duke Power Company's position that the leak tight enclosure is not required for penetrations which are subject to low stress levels. The containment sump suction lines meet this criteria and the valve and pipe are considered an extension of the containment. The welds in the unguarded portions outside containment will be subjected to a program of augmented inservice inspection to insure integrity.

The Westinghouse position on this regulatory guide is provided in Section 6.2.4.1.

Regulatory Guide 1.142

Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments) (4/78).

Discussion

The implementation date for this regulatory guide (construction permit applications docketed after December 15, 1978) is after the October 27, 1972 construction permit docket date for the Catawba Nuclear Station and, therefore, this regulatory guide is not applicable to Catawba. Nevertheless, a comparison of the requirements of ACI 318 and ACI 349 is provided in Table 3-35.

Regulatory Guide 1.143

Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants (Revision 1, 10/79).

Discussion

The design of radioactive waste management systems, structures, and components at Catawba is in compliance with the requirements of this regulatory guide with the following exceptions or alternatives:

Section	Compliance Status
C.1.1	In compliance, except that the Steam Generator Blowdown System is not considered a radioactive waste treatment system and is, therefore, outside the scope of Regulatory Guide 1.143.
C.1.1.1	In compliance.
C.1.1.2	In compliance, except that deviations from material restrictions are allowed when use of non-steel material is warranted by documented and approved engineering evaluations.
C.1.1.3	In compliance.
C.1.1.4	In compliance.
C.1.2	In compliance.

C.1.2.1 In compliance, except that the secondary condensate storage facilities are not considered part of the radioactive waste treatment system and are, therefore, outside the scope of Regulatory Guide 1.143. Also, tank liquid level alarms will sound at the main control panel associated with the particular tank affected.

C.1.2.2 In compliance.

C.1.2.3 In compliance, except that curbs are added where deemed necessary from an ALARA standpoint.

C.1.2.4 In compliance.

C.1.2.5 In compliance.

C.2.1 In compliance.

C.2.1.1 In compliance.

C.2.1.2 In compliance, except that atmospheric and 0-15 psig tanks can also be designed, fabricated, inspected, and tested to ASME Section VIII, Division 1. Also, an exception to the materials restrictions may be allowed when non-steel material is justified by documented and approved engineering evaluations.

C.2.1.3 In compliance.

C.3.1 In compliance.

C.3.1.1 In compliance.

C.3.1.2 In compliance, except that deviations from material restrictions are allowed when use of non-steel material is warranted by documented and approved engineering evaluations.

C.3.1.3 In compliance.

C.3.1.4 In compliance.

C.4.1 In compliance.

C.4.2 In compliance, except that atmospheric and 0-15 psig tanks can also be designed, fabricated, inspected, and tested to ASME Section VIII, Division 1.

C.4.3 In compliance.

C.4.4 In compliance.

C.4.5 In compliance.

C.5 In compliance.

C.6 In compliance, except that atmospheric and 0-15 psig tanks can also be designed, fabricated, inspected, and tested to ASME Section VIII, Division 1.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.144

Auditing of Quality Assurance Programs for Nuclear Power Plants (Revision 1, 9/80).

<u>*Discussion*</u>

Duke Power conforms to Regulatory Guide 1.144 for internal audits. Duke Power's method of external audit is in compliance with ANSI N45.2.13-1976 and is fully described in the Duke Power Company Topical Report, DUKE-1A. ***NOTE:*** *The NRC withdrew this Reg Guide on 7/31/91 in FR 36175.*

<u>*Regulatory Guide 1.145*</u>

Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (Revision 1, 2/83).

<u>*Discussion*</u>

Current values of the design basis accident Offsite X/Q values for the EAB and LPZ are provided in UFSAR Chapter 15 for each accident scenario. Historical values used in plant licensing remain in UFSAR Table 2-36 and Table 2-37 for information only.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

<u>*Regulatory Guide 1.146*</u>

Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plant (8/80).

<u>*Discussion*</u>

The quality assurance program for Catawba complies with the requirements of Regulatory Guide 1.146. ***NOTE:*** *The NRC withdrew this Reg Guide on 7/31/91 in 56FR 36175.*

<u>*Regulatory Guide 1.147*</u>

In-service Inspection Code Case Acceptability, ASME Section XI, Division 1 (Rev 5, 8/86)

<u>*Discussion*</u>

The BWI steam generator design complies with the regulatory position. Code Case N401 is used on the RSG to permit digitized collection and storage of data for permanently recording eddy current examination of pre-service exam.

<u>*Regulatory Guide 1.147*</u>

In-service Inspection Code Case Acceptability, ASME Section XI, Division 1 (Rev 13, 6/2003)

<u>*Discussion*</u>

This regulatory guide identifies the Code Cases that have been determined by the NRC to be acceptable alternatives to applicable parts of Section XI. These Code Cases may be used by licensees without a request for authorization from the NRC, provided that they are used with any identified limitations or modifications. Code Cases N-640 and N-641 are both utilized under the provisions of Regulatory Guide 1.147. Code Case N-640 was used to develop the 34 EFPY operating limit curves for heatup and cooldown of the reactor vessels for CNS Units 1 and 2. Code Case N-641 was also used to develop the low temperature overpressure (LTOP) requirements for the reactor vessels at both units. The implementation of these code cases is described in Sections 5.3.1 and 5.2.2, respectively.

Regulatory Guide 1.149

Nuclear Power Simulation Facilities for Use in Operator Licensing Examinations (October 2001)

Discussion

This Regulatory Guide provides guidance in implementing the requirements of 10CFR 55, "Operators' Licenses".

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Guide 1.150

Ultrasonic Testing of Reactor Vessel Welds during Preservice and Inservice Examinations (6/81) Rev. 1.

Discussion

Duke follows the guidance of Reg Guide 1.150, including appendix A with the exception of Regulatory Position C.7.a on best estimate of the error band in sizing the flaws. NOTE: This Reg Guide was withdrawn by the NRC on 2/11/03 in 73 FR 7766.

Regulatory Guide 1.155

Station Blackout (August 1988)

Discussion

This Regulatory Guide provides guidance in implementing the requirements of the Station Blackout Rule, 10CFR50.63. Discussion of Station Blackout is provided in section 8.4.

Regulatory Guide 1.160

Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (May 2012) Revision 3.

Discussion

This Regulatory Guide provides guidance in implementing the requirements of the Maintenance Rule 10CFR50.65.

Regulatory Guide 1.163

Performance-Based Containment Leak-Test Program dated 9/1/1995.

Discussion

In lieu of Regulatory Guide 1.163, this program is implemented in accordance with NEI 94-01, “Industry Guidance for Implementing Performance-Based Option of 10CFR Part 50, Appendix J” Revision 3-A, dated July 2012, with the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008.

Regulatory Guide 1.194

Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants (June 2003).

Discussion

This Regulatory Guide provides guidance on air dispersion modeling to evaluate control room habitability, with potential impacts at the control room ambient air intakes. Duke uses the ARCON96 model (NUREG/CR-6331 Revision 1, 1997), which implements Regulatory Guide 1.194.

Regulatory Guide 1.183

Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors (07/2000)

Discussion

The analyses of radiological consequences of the LOCA, Fuel Handling Accident and Weir Gate Drop, and the Rod Ejection Accident now conforms to Regulatory Guide 1.183 (all) and Appendices A (LOCA), B (Fuel Handling Accident and Weir Gate Drop), and H (Rod Ejection Accident). Exceptions to Regulatory Guide 1.183 and Appendix A taken in the AST analysis of the LOCA have been identified (Ref. 20 and 22). Exceptions to Regulatory Guide 1.183 and Appendix H taken in the AST analysis of the Rod Ejection Accident have also been identified (Ref. 24). The AST analysis of the Fuel Handling Accident and Weir Gate Drop conforms to Regulatory Guide 1.183 and Appendix B without exceptions.

Regulatory Guide 1.205

Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants (December 2009)

Discussion

This Regulatory Guide endorses the requirements for a fire protection program that meets the requirements of 10 CFR 50.48(c). The fire protection program is discussed in Section 9.5.1.

Regulatory Guide 4.15

Quality Assurance for Radiological Monitoring Program (Normal Operations) - Effluent Streams and the Environment (Rev. 1, 2/79)

Discussion

Catawba complies with the Regulatory Guide as described in fleet Count Room procedures.

Regulatory Guide 7.10

Establishing QAPs for Packaging Used in the Transport of Radioactive Material (Rev. 1, 6/86)

Discussion

Duke complies with the requirements of Regulatory Guide 7.10, except as identified in Duke Nuclear Guide 7.10, revision 1, dated 6/30/89.

Regulatory Guide 8.2

Guide for Administrative Practices in Radiation Monitoring (Rev. 0, 2/73)

Discussion

Conformance with Regulatory Guide 8.2 is described in UFSAR Chapter 12.

Regulatory Guide 8.3

Film Badge Performance Criteria (Rev. 0, 2/73)

Discussion

Conformance with Regulatory Guide 8.3 is described in UFSAR Chapter 12.

Regulatory Guide 8.4

Direct and Indirect Reading Pocket Dosimeters (Rev. 0, 2/73)

Discussion

Conformance with Regulatory Guide 8.4 is described in UFSAR Chapter 12.

Regulatory Guide 8.7

Occupational Radiation Exposure Records System (Rev. 0, 5/73)

Discussion

Conformance with Regulatory Guide 8.7 is described in UFSAR Chapter 12.

Regulatory Guide 8.8

Information Relevant to Ensuring that Occupational Radiation Exposures and Nuclear Power Stations will be ALARA (Rev. 3, 6/78)

Discussion

Conformance with Regulatory Guide 8.8 is described in UFSAR Chapters 12 and 13.

Regulatory Guide 8.9

Acceptable Concepts, Models, Equations and Assumptions for Bioassay Program (Rev. 0, 9/73)

Discussion

Conformance with Regulatory Guide is described in UFSAR Chapter 12.

Regulatory Guide 8.10

Operating Philosophy for Maintaining Occupational Radiation Exposure ALARA (Nuclear Power Reactors) (Rev. 1, 9/75)

Discussion

Conformance with Regulatory Guide 8.10 is described in UFSAR Chapter 12. Regulatory Guide was adopted with modifications as identified in Chapter 12.

Regulatory Guide 8.12

Criticality Accident Alarm Systems (Rev. 2, 10/88)

Discussion

Conformance with Regulatory Guide 8.12 is described in UFSAR Chapter 12.

Regulatory Guide 8.13

Instruction Concerning Prenatal Radiation Exposure (Rev. 2, 12/87)

Discussion

Conformance with Regulatory 8.13 is described in UFSAR Chapters 12 and 13.

Regulatory Guide 8.15

Acceptable Programs for Respiratory Protection (Rev. 0, 10/76)

Discussion

Conformance with Regulatory Guide 8.15 is described in UFSAR Chapter 12.

Regulatory Guide 8.19

Occupational Radiation Dose Assessment in Light Water Reactor Plants Design Stage Man-Rem Estimates (Rev.1, 6/79)

Discussion

Conformance with Regulatory Guide 8.19 is described in UFSAR Chapter 12.

Deleted per 2016 Update

1.7.2 References

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

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THIS IS THE LAST PAGE OF THE TEXT SECTION 1.7.

1.8 Response to TMI Concerns

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

Section 1.8.1 lists the TMI-related requirements and actions that are contained in "NUREG-0694, TMI-Related Requirements for New Operating Licenses," and "NUREG-0737, Clarification of TMI Action Plan Requirements," and describes the actions taken or planned in response to each item.

1.8.1 Response to TMI Concerns

1.8.1.1 Shift Technical Advisor (I.A.1.1)

A Shift Technical Advisor is included in the on-shift organization as discussed in Section 13.1 and the Technical Specifications.

1.8.1.2 Shift Supervisor Administrative Duties (I.A.1.2)

See Section 13.1.

1.8.1.3 Shift Manning (I.A.1.3)

The shift crew composition for operation of Catawba Units 1 and 2 will be in accordance with Chapter 5 of the Technical Specifications.

Provisions governing the amount of overtime worked by licensed operators is addressed in an administrative procedure. This procedure states that licensed operators shall (1) not work more than 16 hours straight, (2) not work more than 24 hours in any 48-hour period, (3) not work more than 72 hours in any 7-day period, and (4) a break of at least 8 hours should be allowed between work periods. These limits on working time do not include shift turnover time. Deviation from these limits on working time will be authorized and documented by either the Superintendent of Operations or the Station Manager.

1.8.1.4 Immediate Upgrading of Operator and Senior Operator Training and Qualification (I.A.2.1)

See Section 13.2.

1.8.1.5 Administration of Training Programs for Licensed Operators (I.A.2.3)

See Section 13.2.

1.8.1.6 Revise Scope and Criteria for Licensing Exams (I.A.3.1)

See Section 13.2.

1.8.1.7 Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants (I.B.1.2)

See Sections 13.1 and 13.4.

1.8.1.8 Short-Term Accident Analysis and Procedure Revision (I.C.1)

Duke has implemented new procedures and training guidelines for controlling and mitigating small break LOCAs, incidents of inadequate core cooling, and certain anticipated transients. Duke's effort is in conjunction with analysis and research being performed by Westinghouse.

The Westinghouse analysis of small break LOCAs in upper head injection plants, WCAP 9600 and WCAP 9639, have been submitted to the NRC for their review. Duke has reviewed these reports and made the necessary modifications to the Catawba emergency procedures and training program.

Duke Power Company has provided a response to supplement 1 to NUREG-0737, which included a description of the Emergency Procedure Upgrading Program for Catawba. Section 6.2.1 of that response described Duke's plans for incorporating the Westinghouse Emergency Response Guidelines (ERG'S) into the Catawba emergency procedures.

On June 1, 1994 Duke redefined the Catawba Plant Specific Technical Guidelines (PSTGs) to be the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs) and associated deviation documents.

1.8.1.9 Shift Relief and Turnover Procedures (I.C.2)

See Section 13.5.

1.8.1.10 Operations Shift Manager Responsibilities (I.C.3)

The Vice President of Nuclear Production has issued a corporate management directive that clearly establishes the command duties of the operations shift manager and emphasizes the operations shift manager's primary management responsibility for safe operation of the plant. This directive will be reissued annually.

The operations shift manager has been provided with administrative assistance to relieve him from administrative duties which detract from or are subordinate to his management responsibility for safe operation of the plant. The duties, responsibilities, and authority of the operations shift manager and control room operators will be defined in an administrative procedure.

1.8.1.11 Control Room Access (I.C.4)

Administrative procedures have been written to limit personnel access to the control room and to establish a clear line of authority for coping with operational transients and accidents. The Catawba Security Plan controls access to all vital areas of the plant including the control area.

1.8.1.12 Procedures for Feedback of Operating Experience to Plant Staff (I.C.5)

See procedures AD-PI-ALL-0400, Operating Experience Program, AD-PI-ALL-0401, Significant Operating Experience Program, and AD-PI-ALL-0402, INPO Consolidated Event System (ICES) Reporting.

1.8.1.13 Procedures for Verifying Correct Performance of Operating Activities (I.C.6)

Permanent station procedures that require valve movement in safety-related systems include provisions to provide assurance that these valves are returned to their correct position. These procedures require verification of the operability of a redundant system prior to the removal of any safety-related system from service, verification of the operability of all safety-related *systems when they are returned to service, and notification of and action by the Operations Shift Manager and reactor operators whenever any safety-related system is removed from or returned to service. Formal checklists are used to provide assurance that all valves in these safety-related systems are properly aligned. These procedures also require independent verification of proper valve alignment and source of power to those valves that are important to safety in safety-related systems.*

A removal and restoration procedure governs the repositioning of valves in safety-related systems following maintenance activities or other non-normal activities which require valve movement. A formal checklist provides assurance that the repositioned safety-related valves are properly aligned following these activities. This procedure also requires independent verification of proper valve alignment and source of power to those valves that are important to safety in safety-related systems.

Notification of and action by the Operations Shift Manager and reactor operators whenever any safety-related system is removed from or returned to service is accomplished by the use of red tags and the red tag logbook, white tags and the white tag logbook, out of service stickers, and the 1.47 bypass panel. Log entries denoting the removal and restoration are made in the Reactor Operator's Log. All of the above documents are reviewed during shift turnovers as required by the turnover procedure.

1.8.1.14 NSSS Vendor Review of Procedures (I.C.7)

See Sections 13.5 and 14.2.3.2.

1.8.1.15 Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants (I.C.8)

Station emergency procedures will be available for review by the NRC as described in Section 13.5.

1.8.1.16 Control Room Design (I.D.1)

A description and results of the Unit 1 control room review were provided in the document “Response to Supplement 1 to NUREG-0737” which was submitted on April 14, 1983 by letter from H. B. Tucker to H. R. Denton. A supplemental report for Unit 2, including environmental survey results for both units, was submitted on March 28, 1984 by letter from H. B. Tucker to H. R. Denton.

1.8.1.17 Training During Low-Power Testing (I.G.1)

See Section 14.5.30 abstract.

1.8.1.18 Reactor Coolant System Vents (II.B.1)

A reactor vessel head vent has been incorporated into the reactor coolant system design and appears on Figure 5-1, Figure 5-3 and Figure 5-4. The former vessel vent line, which was intended for use only during reactor vessel fill and drain operation, has been suitably modified to permit reactor vessel venting during the course of an accident. If the requirement for venting is indicated, the operator will open valves 1NC250A and 1NC251B which will vent the vessel head to the pressurizer relief tank via flow restricting orifice 1NCFE6330. When venting is completed, the operator will close these two valves, but the closure of either will isolate the reactor vessel. If the A (B) train of emergency power is lost prior to venting initiation, the operator would have to *restore power to valve 1NC252B (1NC253A) (in the parallel flow path to the valve in the failed train) in order to open the vent line. Valves NC252B and NC253A are in the closed position with power removed during normal operation. A single failure would affect only one of the two powered valves. Isolation of the vessel at the conclusion of the vent would be assured by the closure of one of these valves.*

The four one-inch, class A, active, EMO vessel vent isolation valves are located in lower containment with controls and valve position indication in the control room. These valves receive power from diesel backed, emergency power sources and are environmentally qualified for postulated post accident environmental conditions. These valves may be individually stroke tested during power operation.

Vent system piping valves, components and supports are classified seismic Category I and: 1) Safety Class A up to and including the second normally closed valve; and 2) Safety Class B up to and including the flow-restricting orifice. The design temperature and pressure of piping, valves and components are 650°F and 2,500 psia, respectively.

A LOCA caused by a break in the present vent pipe would be the same as one in the initial vent pipe so is effectively included in the original LOCA analyses. In addition, flow restricting orifice 1NCFE6330 will limit flow during venting to less than the lower limit of a LOCA. Potential for inadvertent operation of the vent path is reduced by requiring two valves be opened to initiate venting.

Criteria for use of the RCS Vent System were submitted to the NRC for review per the July 7, 1981 letter from R. W. Turgensen (Chairman, Westinghouse Owner Group) to Stephen H. Hanauer (NRC) in response to NUREG-0737, Item 1.C.1.

1.8.1.19 Plant Shielding (II.B.2)

A radiation and design review of systems that, as a result of an accident, contain highly radioactive materials was performed using the same methods and codes used in Section 12.3.3. The systems and volumes assumed to contain the radiation sources were containment, residual heat removal system, safety injection systems, CVCS, containment spray system, sample lines and gaseous radwaste.

The radioactive source term for the direct radiation to the control room complex form external sources were equivalent to the terms recommended in Regulatory Guide 1.183 (ref. Section 15.0). The source terms used to calculate the radiation doses to areas other than the control room complex were equivalent to the terms recommended in Regulatory Guide 1.3 and 1.4.

1. *Liquid Containing Systems: 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium radioiodine inventory and 1% of all others were assumed to be mixed in the largest sump volume.*
2. *Gas Containing Systems: 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium radioiodine inventory and 1% of all other assumed to be mixed in the containment net free volume.*

The liquid and gaseous source terms vs. hours after release are shown in Table 12-22.

Areas of vital access are the control room complex, the post-accident sample room, the normal sample room, the count room, and the technical support center. Occupancy for the control room and technical support center is based on occupancy factors in Standard Review Plan 6.4 (i.e., 100% for the first 24 hours, 60% for 24 to 96 hours, and 40% for occupancy after 96 hours). The radiation dose in the control room complex from sources external to it are 750 mRem (0.75 Rem). This is added to the radioactive dose in the control room complex from exposure to an *inhalation of fission products following a design basis accident. This addition is shown in Table 15-40. The post accident dose in the technical support center is 200 mRem. Occupancy times for the normal sample room, the radiochemical laboratory, and the health physics laboratory are 50% of the time an individual is working, or 12% of the total one year integrated dose in those areas. Doses for these areas are below 100 mRem for the duration of the accident. Access exposure to these areas are below 1 mRem per trip.*

For the post-accident sample area it is assumed that one sample is taken each shift or eight hour period, and that it takes five minutes to remove the quick-disconnect sample cylinder from the panel and leave the area. The occupancy time for the individual is 0.3% of the total integrated dose for the year or 66 mRem. Access exposure to this area is below 1.5 mRem per trip. For operation of the post-accident sample panel, refer to 1.8.1.20.

The radwaste control panels and the hydrogen recombiner control panel are located in the corridor area on elevation 543 and are not affected by a LOCA. The normal operation dose rate in this area is less than 5 mR/hr. Access exposure to these areas is below 1.5 mRem per trip.

The radiation review showed that doses for personnel in vital areas of the plant did not exceed GDC19.

The zone limits are listed below:

Zone I - Greater than 100.0 rads/hr.

Zone II - 10.0 - 100.0 rads/hr.

Zone III - 1.0 - 10.0 rads/hr.

The radiation zones are shown on Figure 1-25 through Figure 1-29. Gradients are expected in radiation levels within each zone. Dose rates are determined at the time of maximum activity. No decay time is taken for the containment and recirculation fluid. For the Annulus and Auxiliary Building ventilation filters, the activity builds as a function of time and reaches a maximum at 326 hours for the annulus filters and 23 hours for the Auxiliary Building filters. In both cases a plot of dose rate (normalized to the maximum) versus time are determined for the one year following an accident.

Additional shielding is not necessary in order to meet the requirements of GDC19.

1.8.1.20 Post Accident Sampling (II.B.3)

A new sampling panel has been designed to allow analysis of reactor coolant samples under accident conditions. The sample line is routed from containment directly to the sampling panel. In order to minimize personnel access limitations, the routing is in accordance with the findings of the plant shielding review. This system will allow collection of reactor coolant samples under pressurized conditions and under low pressure conditions when the reactor coolant pumps are running.

The design of this system significantly reduces radiation exposures during sample collection under accident conditions.

In addition to the reactor coolant sample line, a containment atmosphere sample line will be routed to a new accident air sampling panel. The containment atmosphere sample will be obtained from the hydrogen analyzer sample lines. Sample line length is minimized and large radius bends are used to minimize plate out and provide an accurate effluent sample.

Procedures for collection and transport of reactor coolant, sump water, and containment air samples under post-accident conditions have been revised to incorporate actions to be taken to *minimize radiation exposures. These procedures specify the preplanning to be performed as well as modifications and approvals required prior to sample collection. The post-accident sampling systems will be tested per periodic test procedures to ensure, to a high degree of reliability, that they will be available, if required. Samples can be collected and analyzed within three hours in all instances. Post-accident dose assessment for sampling indicates that personnel exposures would be well below GDC-19 criteria.*

License Amendments No. 193 and 185 were made in 2001 to eliminate licensing requirement to have and maintain the Post Accident Sampling System.

See Section 9.3.2.1.2 for more information.

1.8.1.21 Training for Mitigating Core Damage (II.B.3)

Duke has modified the Catawba training program in order to place increased emphasis on the operation and significance of any Catawba systems or instrumentation which could be used to monitor and control accidents in which the core may be severely damaged. This additional training identified the vital instrumentation which supplies the operator with needed information in a degraded core situation. The training also identifies alternate methods of obtaining this information as well as specific instruction in the interpretation of instrument readings in degraded core situations.

Operating personnel from the station manager through the operating chain (including the Shift Work Managers (STA's) to the licensed operators receive training for mitigating core damage.

1.8.1.22 Relief and Safety Valve Test Requirements (II.D.1)

EPRI PWR Safety and Relief Valve Test Program will be used by Duke to respond to NRC recommendations in NUREG-0737. The Catawba valves covered by the EPRI program are pressurizer safety valves (Dresser type 6-31749A), pressurizer PORV (Control Components, Inc.), and PORV block valves (Rockwell Equiwedge gate valve).

Plant specific information on the Catawba pressurizer safety valves, PORV's and PORV block valves was transmitted by H.B. Tucker's letters of October 26, 1983 and February 3, 1984 to H. R. Denton.

1.8.1.23 Relief and Safety Valve Position Indication (II.D.3)

<u>PORV</u>

The position of the pressurizer power-operated relief valves is detected by seismically and environmentally qualified stem-mounted limit switches. The limit switches actuate indicator lights on the main control board. The entire circuit including power supply is safety-related. Additionally, a control room computer alarm is activated upon the opening of a PORV.

<u>Safety Valve</u>

Flow through the safety valves is detected by an acoustic flow detection system. This system senses vibrations caused by flow through the valve which is an indication that the valve is not fully closed.

Two accelerometers are strapped to the discharge piping of each safety valve. One of these is an installed spare and is wired to the electronics cabinet but not monitored. A charge converter processes the accelerometer output and provides the voltage input to the monitor. The RMS value of this signal is related to the flow through the valve. This signal is filtered and amplified and is available on a front panel BNC connector. An RMS to DC converter provides an output to *drive a bar graph on the front panel. The bar graph is a set of ten vertically arranged indicator lights which are labeled to give valve position as a fraction of full open. The charge converter is located in containment and the electronics cabinet is in the electrical penetration room.*

The alarm output of the monitor is used to provide indication and alarm when flow exists through any of the three safety valves. A safety grade indicator light and a non-safety annunciator are provided. The bar graphs on the monitor can be used to determine which valve is open.

The system, with the exception of the annunciator alarms, is safety-grade and meets the appropriate seismic and environmental qualification requirements.

1.8.1.24 Auxiliary Feedwater System Reliability Evaluation (II.E.1.1)

This section of the UFSAR is a post-TMI discussion related to CA System reliability. A modification was made to the CA System to delete the flow optimization and pump runout protection circuitry under NSMs CN-11371/0 and CN-21371/0. CN-11371/0 was implemented along with the Replacement Steam Generator (RSG) Project for Catawba Unit 1. These changes and all associated analyses were not applied and were not valid prior to the start of Unit 1, Cycle 10. CN-21371/0 was implemented during the outage EOC8. These changes and all associated analyses were not applied and were not valid prior to the start of Unit 2, Cycle 9. These NSMs aligned the turbine driven CA pump to all four S/Gs which differed from the existing design by additionally aligning the A and D S/Gs. ***Flow optimization and runout protection*** *are provided in a "passive" way by positioning the travel stops on the flow control valves. This action results in adequate flow being provided to an acceptable number of "intact S/Gs" while simultaneously providing adequate flow resistance via the CA system piping, valves and components to avoid runout concerns.*

These modifications result in a CA System that is less complex due to the removal of flow optimization and pump runout circuitry. The resulting design eliminates the failure modes of valves CA-46B and CA-58A, failing to close when required as dictated by the previous circuitry. Also eliminated, are the failure modes associated with the circuitry that was removed. The CA pumps will still be operating within manufacturer limitations as related to runout.

It is evaluated that the reliability of the CA System as discussed in this section of the UFSAR is not degraded as a result of these modifications. These NSMs make the CA System more passive, less complex, and more reliable while providing adequate flow for those UFSAR-evaluated events which credit the CA System.

The Auxiliary Feedwater System is described in Section 10.4.9. A reliability analysis was submitted on October 12, 1981.

The Auxiliary Feedwater System has been reviewed and found to meet all the requirements of Standard Review Plan 10.4.9 and Branch Technical Position ASB 10-1.

A reliability evaluation (WCAP-9946) based on the method described in Enclosure 1 of the March 10, 1980 letter has been performed and was submitted by an October 12, 1981 letter from W. O. Parker, Jr. to Harold R. Denton. This evaluation was designed to allow a reliability comparison to other studied auxiliary feedwater systems and to identify any dominant component failures or other faults affecting system availability.

The results of the analysis indicate that the reliability ranking of the Catawba Auxiliary Feedwater System, compared to the reliabilities as defined and reported for Westinghouse plants in NUREG-0611, is high for a loss of main feedwater transient, medium (high end) for a loss of main feedwater coupled with a loss of offsite power transient, and medium (low end) for *the unlikely transient of a loss of main feedwater in coincidence with a total loss of both onsite and offsite AC power.*

Dominant contributors to system unavailability which were identified in this analysis include: system outages resulting from unscheduled maintenance of components and component hardware failures.

A review of the Westinghouse evaluation described above was conducted by Duke Power Company. It was determined that no changes are required in the auxiliary feedwater system design for Catawba Nuclear Station.

Response to the generic short-term and long-term recommendations discussed in Section 5 of Enclosure 1 to the March 10, 1980 letter follows:

1.8.1.24.1 Short-Term Recommendations

Recommendation GS-1

The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumentation can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

Response

Catawba Nuclear Station has Standard Technical Specifications and as such already has these requirements included in the Technical Specifications.

Recommendation GS-2

The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See "Recommendation GL-2" for the longer-term resolution of this concern.

Response

The Catawba auxiliary feedwater system design has redundant flow paths via redundant pumps, valves and piping. The valves in this system will be inspected per the Standard Technical Specifications.

Recommendation GS-3

The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system flow will not result in plant damage due to water hammer.

<u>Response</u>

Auxiliary feedwater flow is not throttled initially to prevent water hammer. The required flow rate is available within 60 seconds following the initiating event.

<u>Recommendation GS-4</u>

Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place.

<u>Response</u>

Transfer of the auxiliary feedwater supply from the normal to the safety grade assured supply occurs automatically. The instrumentation and controls utilized in the switchover logic are safety grade.

<u>Recommendation GS-5</u>

The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train, independent of any AC power source.

<u>Response</u>

The auxiliary feedwater system at Catawba is capable of automatic initiation and of providing the required flow for 2 hours independent of AC power source. This is accomplished by means of the turbine-driven auxiliary feedwater pump and DC powered instrumentation and controls.

<u>Recommendation GS-6</u>

The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

1. *Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.*
2. *The licensee should propose Technical Specifications to assure that, prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.*

<u>Response</u>

1. *Procedures used to manipulate valves which are important to safety of the auxiliary feedwater system require that such valves be manipulated by one operator and independently verified by another operator. (See response to Section 1.8.1.13).*
2. *Prior to unit startup following any cold shutdown of 30 days or longer and at least once per refueling cycle, the CA System is given either a manual or an automatic initiation signal in order to verify the normal flow path (Technical Specifications 4.7.1.2.2).*

Recommendation GS-7

The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade.

Response

The Catawba auxiliary feedwater system employs safety-grade automatic initiation signals and circuits. Automatic initiation of the system is discussed in Section 10.4.9.2.

Recommendation GS-8

The licensee should install a system to automatically initiate AFW system flow.

Response

See response to "Recommendation GS-7.".

1.8.1.24.2 Additional Short-Term Recommendations

Recommendation

The licensee should provide redundant level indication and low level alarms in the control room for the AFW system primary water supply, to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW is operating.

Response

As noted in the response in "Recommendation GS-4", the Catawba design utilizes an automatic transfer of the auxiliary feedwater supply to the assured supply, the Nuclear Service Water System. Redundant level instrumentation is provided as discussed in Section 10.4.9.2. In addition to this, single channel, non-safety-grade level indication and low level alarms are provided in the control room for each of the normal auxiliary feedwater sources (Auxiliary Feedwater Condensate Storage Tank, (for Unit 1 only), upper surge tank, and condenser hotwell). For Unit 2 the auxiliary feedwate condensate storage tank is also an auxiliary feedwater source but is normally isolated when the CA System is required to be operable.

Recommendation

The licensee should perform a 72 hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72 hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

Response

It is our understanding that the NRC Staff has modified this recommendation to perform a 48 hour endurance test, rather than a 72-hour test. Forty-eight hour endurance tests have been *performed on each motor driven auxiliary feedwater pump and the turbine driven auxiliary feedwater pump prior to initial fuel loading.*

Recommendation

The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

1. *Safety-grade indication of AFW flow to each steam generator should be provided in the control room.*
2. *The AFW flow instrument channels should be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the AFW system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.*

Response

1. *Indication of AFW flow to each steam generator will be provided in the control room per the requirements specified in NUREG-0737.*
2. *AFW flow instrumentation channels are powered from emergency buses consistent with satisfying the emergency power diversity requirements set forth in the Auxiliary Systems Branch Technical Position 10-1.*

Recommendation

Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system from the test mode to its operational alignment.

Response

The Catawba AFW system consist of three separate trains. Two trains would remain available for operation during testing of any one train.

1.8.1.24.3 Long-Term Recommendations

Recommendation GL-1

For plants with a manual starting AFW system, the licensee should install a system to automatically initiate the AFW system flow. This system and associated automatic initiation signals should be designed and installed to meet safety-grade requirements. Manual AFW system start and control capability should be retained with manual start serving as backup to automatic AFW system initiation.

Response

See response to “Recommendation GS-7”.

Recommendation GL-2

Licensees with plant design in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

Response

The Catawba auxiliary feedwater system design has redundant flow paths via redundant pumps, valves and piping.

Recommendation GL-3

At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any AC power source for at least two hours. Conversion of DC power to AC power is acceptable.

Response

See response to "Recommendation GS-5".

Recommendation GL-4

Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW system to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure, or upgrading the normal source of water to meet seismic Category 1 and tornado protection requirements.

Response

Auxiliary feedwater system pumps are protected by automatic switchover to the safety-grade source of water following any loss of normal source resulting from natural phenomena or other causes.

Recommendation GL-5

The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

Response

See response to "Recommendation GS-7".

Response to the questions identified in Enclosure 2 of the March 10, 1980, letter follows:

Item 1a

Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

1. Loss of Main Feed (LMFW)
2. LMFW w/loss of offsite AC power
3. LMFW w/loss of onsite and offsite AC power
4. Plant cooldown
5. Turbine trip with and without bypass
6. Main steam isolation valve closure
7. Main feed line break
8. Main steam line break
9. Small break LOCA
10. Other transient or accident conditions not listed above.

Response

The Auxiliary Feedwater System serves as a backup system for supplying feedwater to the secondary side of the steam generators at times when the feedwater system is not available, thereby maintaining the heat sink capabilities of the steam generator. As an Engineered Safeguards System, the Auxiliary Feedwater System is directly relied upon to prevent core damage and system overpressurization in the event of transients such as a loss of normal feedwater or a secondary system pipe rupture, and to provide a means for plant cooldown following any plant transient.

Following a reactor trip, decay heat is dissipated by evaporating water in the steam generators and venting the generated steam either to the condensers through the steam dump or to the atmosphere through the steam generator safety valves or the power-operated relief valves. Steam generator water inventory must be maintained at a level sufficient to ensure adequate heat transfer and continuation of the decay heat removal process. The water level is maintained under these circumstances by the main Feedwater System, or if the main Feedwater System is not operable, by the Auxiliary Feedwater System which delivers an emergency water supply to the steam generators. The Auxiliary Feedwater System must be capable of functioning for extended periods, allowing time either to restore normal feedwater flow or to proceed with an orderly cooldown of the plant to conditions where the Residual Heat Removal System can be placed into operation for continued decay heat removal. The Auxiliary Feedwater System flow and the emergency water supply capacity are sufficient to remove core decay heat, reactor coolant pump heat, and sensible heat during the plant cooldown. The Auxiliary Feedwater System can also be used to maintain the steam generator water levels above the tubes following a LOCA. In the latter function, the water head in the steam generators serves as a barrier to prevent leakage of fission products from the Reactor Coolant System into the secondary plant.

Design Conditions

The reactor plant conditions which impose safety-related performance requirements on the design of the Auxiliary Feedwater System are as follows for the Catawba Units.

1. Loss of Main Feedwater Transient
   a. Loss of main feedwater with offsite power available
   b. Loss of main feedwater with offsite power not available
2. Secondary System Pipe Ruptures
   a. Feedline rupture
   b. Steamline rupture

3. *Loss of all AC Power*
4. *Loss of Coolant Accident (LOCA)*
5. *Cooldown*

Loss of Main Feedwater Transients

The design loss of main feedwater transients are those caused by:

1. *Interruptions of the Main Feedwater System flow due to a malfunction in the feedwater or condensate system.*
2. *Loss of non-emergency AC power with the consequential shutdown of the system pumps, auxiliaries, and controls.*

Loss of main feedwater transients are characterized by a reduction in steam generator water levels which results in a reactor trip, a turbine trip, and auxiliary feedwater actuation by the protection system logic. Following reactor trip from a high initial power level, the power quickly falls to decay heat levels. The water levels continue to decrease, progressively uncovering the steam generator tubes as decay heat is transferred and discharged in the form of steam either through the steam dump valves to the condenser or through the steam generator safety or power-operated relief valves to the atmosphere. The reactor coolant temperature increases as the residual heat increased temperature, the volume of reactor coolant expands and begins filling the pressurizer. Without the addition of sufficient auxiliary feedwater, further expansion will result in water being discharged through the pressurizer safety and/or relief valves. If the temperature rise and the resulting volumetric expansion of the primary coolant are permitted to continue, then (1) pressurizer safety valve capacities may be exceeded causing overpressurization of the Reactor Coolant System and/or (2) the continuing loss of fluid from the primary coolant system may result in bulk boiling in the Reactor Coolant System and eventually in core uncovering, loss of natural circulation, and core damage. Hence, the timely introduction of sufficient auxiliary feedwater is necessary to arrest the decrease in the steam generator water levels, to reverse the rise in reactor coolant temperature, to prevent the pressurizer from filling to a water solid condition, and eventually to establish stable hot standby conditions. Subsequently, a decision may be made to proceed with plant cooldown if the problem cannot be satisfactorily corrected.

The loss of non-emergency AC power transient differs from a simple loss of main feedwater in that emergency power sources must be relied upon to operate vital equipment. The loss of power to the electric driven condenser circulating water pumps results in a loss of condenser vacuum and condenser dump valves. Hence, steam formed by decay heat is relieved through the steam generator safety valves or the power-operated relief valves. This transient is similar to that discussed in the previous paragraph except that reactor coolant pump heat input is not a consideration following loss of power to the reactor coolant pump bus.

Secondary System Pipe Ruptures

The feedwater line rupture accident is postulated to result in the loss of feedwater flow to the steam generators but also results in the complete blowdown of one steam generator within a short time if the rupture should occur downstream of the last check valve in the main or auxiliary feedwater piping to an individual steam generator. Another significant result of a feedline rupture may be the spilling of auxiliary feedwater out the break as a consequence of the fact that the auxiliary feedwater branch line may be connected to the main feedwater line in the region of the postulated break. Such of the total auxiliary feedwater flow (the system preferentially pumps water to the lowest pressure region) to the faulted loop rather than to the intact steam *generators which are at relatively high pressure. The concerns are similar for the main feedwater line rupture as those explained for the loss of main feedwater transients.*

Main steamline rupture accident conditions are characterized initially by plant cooldown and, for breaks inside containment, by increasing containment pressure and temperature. Auxiliary feedwater is not needed during the early phase of the transient, but flow to the faulted loop will contribute to an excessive release of mass and energy to containment. Thus, steamline rupture conditions establish the upper limit on auxiliary feedwater flow delivered to a faulted loop. Eventually, however, the Reactor Coolant System will heat up again and auxiliary feedwater flow will be required to be delivered to the non-faulted loops, but at somewhat lower rates than for the loss of feedwater transients described previously. Provisions are made in the design of the Auxiliary Feedwater System to limit, control, or terminate the auxiliary feedwater flow to the faulted loop as necessary in order to prevent containment overpressurization following a steamline break inside containment, and to ensure the minimum flow to the remaining intact loops.

Loss of All AC Power

The loss of all AC power is postulated as resulting from accident conditions wherein not only onsite and offsite AC power is lost but also AC emergency power is lost as an assumed common mode failure. Battery power for operation of protection circuits is assumed available. The impact on the Auxiliary Feedwater System is the necessity for providing both an auxiliary feedwater pump power and control source which are not dependent on AC power and which are capable of maintaining the plant at hot shutdown until AC power is restored.

Loss-of-Coolant Accident (LOCA)

The loss of coolant accidents do not impose on the Auxiliary Feedwater System any flow requirements in addition to those required by the other accidents addressed in this response. The following description of the small LOCA is provided here for the sake of completeness to explain the role of the Auxiliary Feedwater System in this transient.

Small LOCA's are characterized by relatively slow rates of decrease in reactor coolant system pressure and liquid volume. The principal contribution from the Auxiliary Feedwater System following such small LOCA's is basically the same as the system's function during hot shutdown or following spurious safety injection signal which trips the reactor. Maintaining a water level inventory in the secondary side of the steam generators provides a heat sink for removing decay heat and establishes the capability for providing a buoyancy head for natural circulation. The Auxiliary Feedwater System may be utilized to assist in a system cooldown and depressurization following a small LOCA while bringing the reactor to a cold shutdown condition.

Cooldown

The cooldown function performed by the Auxiliary Feedwater System is a practical one since the reactor coolant system is reduced from normal zero load temperatures to a hot leg temperature of approximately 350°F. The latter is the maximum temperature recommended for placing the Residual Heat Removal System (RHRS) into service. The RHR system completes the cooldown to cold shutdown conditions.

Cooldown may be required following expected transients, following an accident such as a main feedline break, or during a normal cooldown prior to refueling or performing reactor plant maintenance. If the reactor is tripped following extended operation at rated power level, the

Auxiliary Feedwater System is capable of delivering sufficient auxiliary feedwater to remove decay heat and reactor coolant pump heat following reactor trip while maintaining the steam generator water level. Following transients or accidents, the recommended cooldown rate is consistent with expected needs and at the same time does not impose additional requirements on the capacities of the auxiliary feedwater pumps, considering a single failure. In any event, the process consists of being able to dissipate plant sensible heat in addition to the decay heat produced by the reactor core.

Item 1b

Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

1. *Maximum RCS pressure (PORV or safety valve actuation)*
2. *Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)*
3. *RCS cooling rate limit to avoid excessive coolant shrinkage*
4. *Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.*

Response

Table 1-7 summarizes the criteria which are the general design bases for each event, discussed in the response to "Item 1a" above. Specific assumptions used in the analyses to verify that the design bases are met are discussed in response to "Item 2".

The primary function of the Auxiliary Feedwater System is to provide sufficient heat removal capability for heatup following reactor trip and to remove the decay heat generated by the core and prevent system overpressurization. Other plant protection systems are designed to meet short term or pre-trip fuel failure criteria. The effects of excessive coolant shrinkage are evaluated by the analysis of the rupture of a main steam pipe transient. The maximum flow requirements determined by other bases are incorporated into this analysis, resulting in no additional flow requirements.

Item 2

Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in "Item 1a" above including:

1. *Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.*
2. *Time delay from initiating event to reactor trip.*
3. *Plant parameter(s) which initiates AFWS to flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).*
4. *Minimum steam generator water level when initiating event occurs.*
5. *Initial steam generator water inventory and depletion rate before and after AFWS flow commences -- identify reactor decay heat rate used.*
6. *Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.*

7. Minimum number of steam generators that must receive AFW flow; e.g., 1 out of 2? 2 out 4?

8. RC flow condition -- continued operation of RC pumps or natural circulation.

9. Maximum AFW inlet temperature.

10. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain.

11. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

12. Operating condition of steam generator normal blowdown following initiating event.

13. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

14. Time at hot standby and time to cooldown RCS to RHR system initiation temperature to size AFW water source inventory.

Response

Analyses have been performed for the limiting transients which define the AFWS performance requirements. These analyses have been provided in Chapter 15 of the FSAR. Specifically, they include:

Section 15.2.6
Section 15.2.7
Section 15.2.8
Section 15.1.5

In addition to the above analyses, calculations have been performed specifically for Catawba Units to determine the plant cooldown flow (storage capacity) requirements. The Loss of All AC Power is evaluated via a comparison to the transient results of Section 15.2.6, assuming an available auxiliary pump having a diverse (non-AC) power supply. The LOCA analysis, as discussed above, incorporates the system flows requirements as defined by other transients, and therefore is not performed for the purpose of specifying AFWS flow requirements. Each of the analyses listed above are explained in further detail in the following sections of this response.

Loss of Non-Emergency Power to the Station Auxiliaries

Loss of Normal Feedwater

A loss of feedwater, assuming a loss of power to the reactor coolant pumps, is described in FSAR Section 15.2.6. It is shown for this transient that the peak RCS pressure remains below the criterion for Condition II transients and no fuel failures occur (refer to Table 1-7). Table 1-8 summarizes the assumptions used in this analysis. The analysis assumes that the plant is initially operating at 102% (calorimetric error) of the Engineered Safeguards Design (ESD) rating shown on the table, a very conservative assumption in defining decay heat and stored energy in the RCS. The reactor is assumed to be tripped on low-low steam generator water level, allowing for level uncertainty. As shown in the FSAR, there is a considerable margin with respect to filling the pressurizer for a loss of normal feedwater transient with or without power to the reactor coolant pumps.

Rupture of Main Feedwater Pipe

The double ended rupture of a main feedwater pipe downstream of the main feedwater line check valve is analyzed in FSAR, Section 15.2.8. Table 1-7 summarizes the assumptions used in this analysis. Reactor trip is assumed to occur when the faulted generator is at the low-low level setpoint (adjusted for errors). This conservative assumption maximizes the stored heat prior to reactor trip and minimizes the ability of the steam generator to remove heat. As in the loss of normal feedwater analysis, the initial power rating was assumed to be delivered to the 2 non-faulted steam generators 1 minute after reactor trip. The criteria listed Table 1-7.

This analysis established requirements for layout to preclude indefinite loss of auxiliary feedwater to the postulated break, and establishes train association requirements for equipment so that the AFWS can deliver the minimum flow required in 1 minute assuming the worst single failure.

Rupture of Main Steam Pipe Inside Containment

Because the steamline break transient is a cooldown, the AFWS is not needed to remove heat in the short term. Furthermore, addition or excessive auxiliary feedwater to the faulted steam generator will affect the peak containment pressure following a steamline break inside containment. This transient is performed at four power levels for several break sizes. Auxiliary feedwater is assumed to be initiated at the time of the break, independent of system actuation signals. The maximum flow is used for this analysis. Table 1-8 summarizes the assumptions used in this analysis. At 10 minutes after the break, it is assumed that the operator has isolated the AFWS from the faulted steam generator which subsequently blows down to ambient pressure. The criteria stated in Table 1-7 are met.

This transient establishes the maximum allowable auxiliary feedwater flow rate to a single faulted steam generator assuming all pumps operating, and establishes layout requirements so that the flow requirements may be met considering the worst single failure.

Plant Cooldown

Maximum and minimum flow requirements from the previously discussed transients meet the flow requirements of plant cooldown. This operation, however, defines the basis for minimum required condensate storage tank level, based on the required cooldown duration, maximum decay heat input and maximum stored heat in the system. As previously discussed in the response to "Item 1a," the Auxiliary Feedwater System cools the system to the point where the RHRS may complete the cooldown, i.e., 350°F in the RCS. Table 1-8 shows the assumptions used to determine the cooldown heat capacity of the Auxiliary Feedwater System.

The cooldown is assumed to commence at the maximum rated power, and maximum trip delays and decay heat source terms are assumed when the reactor is tripped. Primary metal, primary water, secondary system metal and secondary system water are all included in the stored heat to be removed by the AFWS. See Table 1-9 for the items constituting the sensible heat stored in the NSSS.

This operation is analyzed to establish minimum tank size requirements for auxiliary feedwater fluid sources which are normally aligned.

Item 3

Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the *margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.*

<u>Response</u>

1. *The AFW pumps will supply the necessary flow to the steam generators considering a single failure.*
2. *The Catawba AFW pumps are provided minimum flow protection by automatic recirculation control valves. These valves do not recirculate any flow during normal operation of the AFW pumps unless the flow rate through the pumps approaches the minimum flow value. This is not a continuous recirculation system. No margin is allowed for pump recirculation flow.*
3. *A margin of 10 gpm is available to account for system leakage.*
4. *A margin of 3% is available to account for pump wear.*

1.8.1.25 Auxiliary Feedwater Initiation and Indication (II.E.1.2)

See Sections 7.4.1 and 10.4.9.

1.8.1.26 Emergency Power for Pressurizer Heaters (II.E.3.1)

<u>Pressurizer Heaters</u>

For each Catawba unit, two groups of pressurizer heaters with a capacity of 416 Kw each are supplied from the 600 VAC Blackout Auxiliary Power System, one heater group per power train. Power is available to each heater group from the offsite power system or from the onsite emergency power system (reference FSAR Chapter 8). Each heater group has the capability to maintain natural circulation under hot standby conditions.

The pressurizer heaters are normally powered from the unit auxiliary power system and are not affected by the occurrence of a safety injection actuation signal (SIAS). In the event of a loss of offsite power, the pressurizer heaters can be manually loaded on the emergency power sources. If the pressurizer heaters are already being powered from the emergency power sources and an SIAS occurs, the heaters will be automatically load shed from the emergency power sources by the diesel generator load sequencers. The SIAS and diesel generator load sequencer must both be reset before the operator can manually reload the pressurizer heaters onto the emergency power sources. These resets, and the manual controls for the pressurizer heater feeder breakers are located in the control room. Procedures addressing manual loading of the pressurizer heaters onto the emergency power sources following an SIAS are available to the operator.

A study was performed by Westinghouse to determine the heater capacity required to maintain Reactor Coolant System pressure with a loss of offsite power and the time frame when emergency power supplies must be available to the heaters.

Pressurizer heat losses can be divided into two basic components: 1) losses through the pressurizer walls, insulation, supports, connections, etc., and 2) losses due to continuous spray flow. Spray flow is driven to the top of the pressurizer by reactor coolant pump head; without offsite power, the pumps will coast down and no spray flow will be supplied. Thus, without offsite power, only heat losses through insulation, supports, etc., must be offset by heaters.

A review of heat loss calculations for a typical 1800 ft^3 pressurizer, such as that installed at Catawba resulted in the determination that a heater capacity of 150 Kw will conservatively *compensate for heat losses from the pressurizer at or below normal operating pressure with no allowance for continuous spray.*

A transient analysis of the loss of offsite power event established that the ability to supply emergency power to heaters at 150 Kw capacity within four hours will prevent loss of subcooling in the primary. Conservative assumptions resulting in least margin to subcooling and rapid decrease in pressure were utilized in this analysis.

1.8.1.27 Containment-Dedicated Penetrations (II.E.4.1)

See Section 6.2.5.

1.8.1.28 Containment Isolation Dependability (II.E.4.2)

See Sections 6.2.4 and 7.3.

1.8.1.29 Additional Accident Monitoring Instrumentation (II.F.1)

Applicability of Regulatory Guide 1.97, Revision 2, to Catawba Nuclear Station has been addressed as an integral part of the overall response to NRC Generic Letter 82-33. Generic Letter 82-33 transmitted Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability". The following items 1.8.1.29.1 and 1.8.1.29.2 evaluate compliance of Catawba Nuclear Station with NUREG 0737 Supplement 1 and NRC Regulatory Guide 1.97, Revision 2. Table 1-11 contains a comparison of plant specific accident monitoring variables with the recommendations of the Regulatory Guide.

1.8.1.29.1 Catawba Nuclear Station Position of Accident Monitoring Instrumentation (A)

1.8.1.29.1.1 Accident-Monitoring Instrumentation (A.1)

The criteria and requirements contained in ANSI/ANS-4.5-180, "Criteria for Accident Monitoring Functions in Light-Water Cooled Reactors," are considered by Duke Power to be generally acceptable for providing instrumentation to monitor variables for accident conditions subject to the clarifications defined below.

1.8.1.29.1.1.1 TYPE A Variables (A.1.a)

Type A variables are defined as those variables which are monitored to provide the primary information required to permit the Control Room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accidents. Primary information is defined as that which is essential for the direct accomplishment of the specified safety functions; it does not include those variables associated with contingency actions which may also be identified in written procedures.

The following variables are those determined to be Type A for Catawba Nuclear Station, as defined above:

1. *Reactor Coolant System Pressure*
2. *Core Exit Temperature*
3. *Reactor Coolant System Hot Leg Water Temperature*
4. *Reactor Coolant System Cold Leg Water Temperature*
5. *Pressurizer Level*

6. *Degrees of Subcooling*
7. *Steam Generator Narrow Range Level*
8. *Steam Line Pressure*
9. *Refueling Water Storage Tank Level*

1.8.1.29.1.1.2 TYPE B and C Variables (A.1.b)

Type B and C variable selection is based on the SPDS Critical Safety Functions (CSFs) described in Section 7.5. The SPDS is provided as an aid to the Control Room operating crew in monitoring the status of the Critical Safety Functions. The Critical Safety Functions monitored are those defined in the Westinghouse Owners Group Critical Safety Function Status Trees. The SPDS provides continuous status updated at regular intervals of the Critical Safety Functions as defined in the Emergency Response Guidelines (ERG).

Since these Critical Safety Functions constitute the basis of the Catawba SPDS and the emergency operating procedures, it is Duke Power's position that they should also be identified as the plant safety functions for accident monitoring (i.e., the basis for Type B and C variable selection).

Using the SPDS Critical Safety Functions as the basis for defining the accident monitoring instrumentation incorporates the concept of monitoring the multiple barriers to the release of radioactive material. The Critical Safety Functions monitored are those which assure the integrity of these barriers. The Status Tree provides an explicit, systematic mechanism for organizing the plant data required to evaluate a Critical Safety Function. The prioritization of the Critical Safety Functions is consistent with the concept of multiple barriers to radiation release.

The Critical Safety Functions are:

1. *Subcriticality*

   The subcriticality status tree monitors the reactor core to assure that it is maintained in a subcritical condition following a successful reactor trip.

2. *Core Cooling*

   The core cooling status tree monitors those variables necessary to evaluate the status of fuel clad heat removal.

3. *Reactor Coolant System Integrity*

   The Reactor Coolant System integrity status tree monitors the pressure-temperature relationship of the Reactor Coolant System with respect to various regions of the pressure-temperature curves.

4. *Heat Sink*

   The heat sink status tree monitors the ability to transfer energy from the reactor coolant to an ultimate heat sink.

5. *Containment*

   The containment status tree monitors those variables which would indicate a threat to containment integrity or other undesirable conditions within containment.

6. *Inventory*

   The inventory status tree monitors for indications of off-normal quantities of reactor coolant in the primary system.

1.8.1.29.1.1.3 Design and Qualification Criteria (A.1.c)

Design and qualification criteria used by Duke Power Company for plant instrumentation are provided below. The category designations are provided for reference to the Regulatory Guide 1.97 (Revision 2) document.

1.8.1.29.1.1.4 Design and Qualification Criteria - Category 1 (A.1.c.i)

Accident monitoring instrumentation which comprise this design and qualification category are considered by Duke Power to be Nuclear Safety Related and thus are classified as Quality Assurance Condition 1 (QA1).

1. *QA1 instrumentation is environmentally qualified in accordance with IEEE 323-1971 as described in the FSAR Section 3.11 and in the Duke Power Company NUREG 0588 submittal. Seismic qualification is in accordance with IEEE 344-1971 as described in the FSAR Section 3.10. Instrumentation is qualified to read within the required accuracy following, but not necessarily during a safe shutdown earthquake.*

   Most of the QA1 instrumentation loops are provided as part of the Process Control System (PCS) Protection Cabinets. These cabinets are supplied as nuclear safety related equipment. Each qualified instrument loop has output isolation devices which provide signals to a control board display and/or a strip chart recorder, and the plant operator-aid computer. The location of the output isolation device is such that it is accessible for maintenance during accident conditions.

2. *No single failure within either the accident monitoring instrumentation, its auxiliary supporting features, or its power sources, con-current with the failures that are a condition or result of a specific accident, will prevent the operators from being presented the information necessary to determine the safety status of the plant and to bring the plant to and maintain it in a safe condition following that accident. Where failure of one accident-monitoring channel results in information ambiguity (i.e., the redundant displays disagree) that could lead operators to defeat or fail to accomplish a required safety function, additional information is provided to allow the operators to deduce the actual conditions in the plant. This is accomplished by providing additional independent channels of information of the same variable (an identical channel) or by providing an independent channel to monitor a different variable that bears a known relationship to the multiple channels (a diverse channel). The information provided to the operator to eliminate ambiguity between redundant channels is needed only during a failure of one of the instrument loops. Therefore, it is considered acceptable to use installed instrumentation of equal design and qualification category, installed instrumentation of a lesser design and qualification category, temporary or portable instrumentation, or sampling to allow the operators to deduce the actual conditions in the plant. Redundant QA1 channels are electrically independent and physically separated from each other.*

   At least one channel of QA1 instrumentation is displayed on a direct-indicating or recording device. (Note: Within each redundant division of a safety system, redundant monitoring channels are not needed.)

3. *The instrumentation is energized from Class 1E Power sources (as described in Chapter 8 of the FSAR) as provided in Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," and are backed by batteries where momentary interruption is not tolerable.*

4. *The instrumentation channel will be available prior to an accident except as provided in Paragraph 4.11, "Exception," as defined in IEEE Standard 279-1971 or as specified in Technical Specifications.*
5. *The following documents pertaining to quality assurance are applicable:*

   Duke 1 Duke Power Company Topical Report, "Quality Assurance Program"

   Catawba Nuclear Station Final Safety Analysis Report

   Chapter 17, "Quality Assurance"

6. *Continuous indication display is provided. Where two or more instruments are needed to cover a particular range, overlapping of instrument span is provided.*
7. *Recording of instrumentation readout information is provided for at least one of the redundant channels. Where direct and immediate trend or transient information is essential for operator information or action, the recording is continuously available on dedicated recorders. Otherwise, it may be continuously updated, stored in computer memory, and displayed on demand. Intermittent displays such as data loggers and scanning recorders may be used if no significant transient response information is likely to be lost by such devices.*

1.8.1.29.1.1.5 Design and Qualification Criteria - Category 2 (A.1.c.ii)

1. *Class 1E (QA1) Category 2 Instrumentation*

   For instrumentation loops that are installed as Class 1E (QA1), environmental qualification is provided per the methodology described in the Catawba NUREG 0588 submittal and the FSAR Section 3.11. Seismic qualification is in accordance with IEEE 344-1971 as described in Section 3.1 of the Catawba FSAR. Quality Assurance in the design, procurement, and installation of these QA Condition 1 Instrumentation Systems is provided as described in the Duke Power Company Topical Report "Duke 1" and FSAR Chapter 17. These instruments are powered from Class 1E Power sources (as described in Chapter 8 of the FSAR) and are backed by batteries where a momentary power interruption is not tolerable.

2. *Non-Class 1E (Non-QA1) Category 2 Instrumentation*

   For instrumentation loops of lesser importance which are not Class 1E (QA1), appropriate qualification is provided. The intent is to provide reasonable assurance that this non-Class 1E instrumentation can be expected to be operable for accident monitoring and analysis. It is Duke's position that such assurance does not need to be as rigorous as for Category 1 and, thus, practical approaches employing equipment design ratings, similarity to qualified equipment, or other engineering judgments are acceptable. Full harsh environment withstand rating may not be provided where an engineering evaluation determines that other environmentally qualified diverse or alternate instrumentation provides an adequate backup reading. Additionally, Category 2 instrumentation which is of primary use during one phase of an accident need not be qualified for all phases of the event. For example, an instrument of primary importance prior to attaining the recirculation mode need not be demonstrated to withstand post-recirculation radiation.

   For non-Class 1E (non-QA1) Category 2 instrumentation, seismic qualification is not required unless seismic induced failure of the instrumentation would unacceptably degrade a safety system.

These instrumentation systems are designed, procured, and installed per Duke Power Company standard practices. Duke Power considers that this is adequate to assure the quality of the subject instrumentation.

Isolation devices are provided to interface between Class 1E and non-Class 1E portions of any of the subject instrumentation loops.

The instrumentation is energized from a high-reliability power source, not necessarily Class 1E Power, and is backed by batteries where momentary interruption is not tolerable.

3. *All Category 2 Instrumentation*

   For both Class 1E and non-Class 1E Category 2 instrumentation:

   The out-of-service interval should be based on normal Technical Specification requirements for the system it serves where applicable or where specified by other requirements.

   The instrumentation signal may be displayed on an individual instrument or it may be processed for display on demand by a video monitor or by other appropriate means.

   The method of display may be by dial, digital, video monitor, or stripchart recorder indication. Effluent radioactivity monitors and meteorology monitors will be recorded. Where direct and immediate trend or transient information is essential for operator information or action, the recording is continuously available on dedicated recorders. Otherwise, it may be continuously updated, stored in computer memory, and displayed on demand.

1.8.1.29.1.1.6 Design and Qualification Criteria - Category 3 (A.1.c.iii)

These instruments do not play a key role in the management of an accident but they do add depth to the Category 1 and 2 instrumentation to the extent that they remain operable. The instrumentation is of high quality commercial grade and is selected to withstand the normal power plant service environment.

The method of display may be by dial, digital, video monitor, or stripchart recorder indication. Effluent radioactivity monitors and meteorology monitors will be recorded. Where direct and immediate trend or transient information is essential for operator information or action, the recording is continuously available on dedicated recorders. Otherwise, it may be continuously updated, stored in computer memory, and displayed on demand.

1.8.1.29.1.1.7 Additional Criteria for Categories 1 and 2 (A.1.d)

In addition to the criteria in A.1.c, the following criteria apply to Categories 1 and 2:

1. *For Nuclear safety related (Class 1E) signals which are transmitted to non-safety related (non-Class 1E) equipment, isolation devices are utilized.*
2. *Dedicated Control board displays for the instruments designated as Types A, B, and C, Category 1 or 2 and qualified for use throughout all phases of an accident will be specifically identified on the control panels so that the operator can discern that they are available for use under accident conditions.*

1.8.1.29.1.1.8 Additional Criteria for All Categories (A.1.e)

1. *In addition to the above criteria, the following criteria apply to all instruments identified in this document:*
2. *Servicing, testing, and calibration programs are specified to maintain the capability of the monitoring instrumentation. For those instruments where the required interval between tests*

will be less than the normal time interval between generating station shutdowns, the capability for testing during power operation is provided.

3. *Whenever means for removing channels from service are included in the design, the design facilitates administrative control of the access to such removal means.*
4. *The design facilitates administrative control of the access to all setpoint adjustments, module calibration adjustments, and test points.*
5. *The monitoring instrumentation design minimizes the development of conditions that would cause meters, annunciators, recorders, alarms, etc., to give anomalous indications which are potentially confusing to the operator. Human factors guidelines are used in determining type and location of displays. The Duke Control Room Review Team will make recommendations as to the type and location of displays for added instrumentation.*
6. *To the extent practicable, the instrumentation is designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.*
7. *To the extent practicable, monitoring instrumentation inputs are from sensors that directly measure the desired variables.*
8. *To the extent practicable, the same instruments are used for accident monitoring as are used for the normal operations of the plant to enable the operators to use, during accident situations, instruments with which they are most familiar. However, where the required range of monitoring instrumentation results in a loss of necessary sensitivity in the normal operating range, separate instruments are used.*
9. *Periodic checking, testing, calibration, and calibration verification are in accordance with the applicable portions of the Catawba FSAR Chapter 7.*

1.8.1.29.1.2 System Operation Monitoring (Type D) and Effluent Release Monitoring (Type E) Instrumentation (A.2)

1.8.1.29.1.2.1 Definitions (A.2.a)

Type D: *those variables that provide information to indicate the operation of individual safety systems.*

Type E: *those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and in continually assessing such releases.*

1.8.1.29.1.2.2 Operator Usage (A.2.b)

The plant design has included variables and information display channels required to enable the Control Room operating personnel to:

1. *Ascertain the operating status of each individual safety system to the extent necessary to determine if each system is operating or can be placed in operation to help mitigate the consequences of an accident.*
2. *Monitor the effluent discharge paths to ascertain if there have been significant releases (planned or unplanned) of radioactive materials and to continually assess such releases.*
3. *Obtain required information through a backup or diagnosis channel where a single channel may be likely to give ambiguous indication.*

1.8.1.29.1.2.3 Design and Qualification Criteria - Types D and E (A.2.c)

The design and qualification criteria for safety system operation monitoring (Type D) and effluent release monitoring (Type E) instrumentation are provided in Section 1.8.1.29.1.1.3 through Section 1.8.1.29.1.1.8.

1.8.1.29.2 Regulatory Guide 1.97 Comparison (B)

Table 1-11 contains a comparison of plant specific accident monitoring variables with the recommendations of NRC Regulatory Guide 1.97, Revision 2. Instrument Ranges, design, environmental qualifications, type of display, and position statements are provided for each variable named in Table 2, "PWR Variables," of Regulatory Guide 1.97, Revision 2.

1.8.1.29.3 Other Instrumentation (C)

1.8.1.29.3.1 Noble Gas Monitors (C.1)

Vent monitors for noble gases are provided with a range adequate to cover both normal and postulated accident conditions. The high and low range noble gas monitors cover the range of 10^{-7} μCi/cc to 10^{+3} μCi/cc. A gross gamma detector is also provided to extend the monitoring range up to 10^{+5} μCi/cc (refer to Table 1-11). The gross gamma monitor is mounted on the unit vent with the detector protruding into the vent stack. If an event were to occur to cause the activity being released to be in the range of 10^{+2} μCi/cc to 10^{+5} μCi/cc, the noble gas monitor sample will be isolated. This action prevents the noble gas monitor from becoming contaminated and rendering erroneous indications when activity starts decreasing.

In addition, procedures for quantifying radioactive releases through all of the atmospheric steam release valves have been developed. These procedures involve the use of dedicated, area radiation monitors. The containment hydrogen purge exhaust discharges through the unit vent and is monitored by the unit vent radiation monitors. These procedures provide Duke with the capability to quantify the noble gas releases.

1.8.1.29.3.2 Containment Pressure (C.2)

Continuous indication of containment pressure is provided in the control room. Measurement and indication range extends from -5 psig to 60 psig. Two redundant differential pressure transmitters are located in separate electrical penetration rooms and are equipped with separate one-half inch tubing impulse lines.

Each impulse line has a fail-open isolation valve located in the annulus. These valves are normally open and have position indication and manual control in the control room. Continuous indication from each transmitter pressure is recorded. These instruments are functionally independent of the existing containment pressure transmitters.

This instrumentation is provided in response to the requirements of Appendix B of NUREG-0737. Further design information is presented in Table 1-11.

1.8.1.29.3.3 Containment Water Level (C.3)

Two containment floor and equipment sumps are provided on the floor of the lower containment (El 550'-6") to collect floor drains and equipment drains.

However, these sumps and their associated pumps and instrumentation serve no safety function.

The containment emergency recirculation sump at Catawba encompasses the entire floor of the lower containment. The two ECCS recirculation lines take suction just inside the Containment wall at elevation 552' and are oriented horizontally. They are not located in the bottom of a recess or sump in the floor. Redundant safety grade level instrumentation is provided to measure emergency recirculation sump level. The range of this instrumentation is .5 to 20.5 feet (El 552'-8 3/4" to 572'-8 3/4") which is equivalent to a lower containment volume of approximately 1,000,000 gallons. The accuracy of this instrumentation is +10.0, -15.0% over the full range.

This instrumentation utilizes five float type level sensors mounted in an overlapping configuration and connected in series to provide a 20.0 foot span. A receiver module located in the electrical penetration room measures the resistance produced by movememt of the floats and generates a standard electrical signal proportional to the calibrated span. Continuous indication from each transmitter is provided in the control room. In addition, both channels of containment water level are recorded.

This instrumentation is provided in response to the requirements of Appendix B of NUREG-0737. Further design information is presented in Table 1-11.

1.8.1.29.3.4 Containment Hydrogen Monitoring (C.4)

Continuous indication of hydrogen concentration in the containment atmosphere is provided in the control room after appropriate solenoid valves are energized. This hydrogen monitoring system consists of two redundant Teledyne analyzer systems with a dual range of 0 to 10%/0 to 30% hydrogen by volume. These analyzers operate independent of the recombiner system and are powered from redundant Class 1E power supplies. Each analyzer has its own containment sample and return lines, and is able to monitor either of two identical containment sampling headers or the calibration gases. Each analyzer has a local control panel indicator and alarm and a separate control room indicator and alarm. In addition, two channels of containment hydrogen concentration are recorded.

Each containment sample header has three inlet samples available for monitoring.

1. *Top of containment*
2. *Operating level*
3. *Steam Generator Cavity.*

All sample selection and switching is accomplished manually by the operator from the local analyzer control panel.

This instrumentation is provided in response to the requirements of Appendix B of NUREG-0737. Further design information is presented in Table 1-11.

1.8.1.29.3.5 Containment High Range Radiation Monitoring (C.5)

Two physically and electrically separated radiation monitors are installed inside the containment. Monitors 1&2EMF53A are located at elevation 580', 200°, and monitors 1&2EMF53B are located at elevation 580', 340°. These monitors are supplied by General Atomics and feature GA detector model number RD23. Each monitor utilizes an ionization chamber to measure gamma radiation and cover the range from 10^0 to 10^8 R/hr. No overlapping of ranges is required. Monitor sensitivity to 62 Kev is $9.8X10^{-12}$ Amps/Rad/hr and the sensivity to 52 Kev is $9.0X10^{-12}$ Amps/Rad/hr. Seismic qualification and environmental qualification of these monitors are discussed in FSAR Sections 3.10 and 3.11, respectively.

One monitor for each unit (1&2EMF53A) is powered from the Train A vital instrument bus, and the other monitor for each unit (1&2EMF53B) is powered from the Train B vital instrument bus. Analog meters (one per train per unit) continuously indicate monitor output in the control room. A continuous strip chart recorder (one train per unit) is also located in the control room.

An electronic calibration of the monitors is performed every refueling outage. In addition, a radiation source is used to perform an in-situ calibration of the monitor range below 10 R/hr.

1.8.1.30 Inadequate Core Cooling Instruments (II.F.2)

Subcooling Monitor

The margin to saturation is calculated from Reactor Coolant System (RCS) pressure and temperature measurements. When RCS pressure is sufficiently less than 800 psig to ensure the low range pressure sensor is within its measurement span, the low range input is used. The wide range pressure inputs are used for the remaining conditions. The average of the five highest value incore thermocouples (from 40 EQ T/C's) are used to represent core exit conditions. The wide range hot leg RTD's are used to measure the loop hot leg temperatures. The plant computer performs averaging and auctioneering functions and a comparison to adjusted saturation curves (adjusted for possible measurement uncertainties) to compute margins and initiate alarms if appropriate.

The computer output consists of a video monitor graphic display which plots plant pressure and temperature in relation to the computer generated adjusted saturation curve. In addition, numerical values are provided for parameters of interest such as pressure, temperatures, and subcooling margins. Alarm status is indicated by messages on the video monitor graphic display, the Alarm video monitor, and by printout on the Alarm Typer. Alarms are provided at a selected margin from the adjusted saturation curve to warn of the approach to loss of adequate subcooling and again upon reaching the adjusted saturation curve to warn of the loss of adequate subcooling. Further details on this subcooling monitor are provided in Table 1-5.

Normal control board instrumentation for RCS temperature and pressure will be used in conjunction with a control room paper copy of the adjusted saturation curves and a written procedure to determine margin to saturation as a backup to the computer calculations.

Reactor Vessel Level Measurement

The reactor vessel level instrumentation system (RVLIS) is of Westinghouse design. The RVLIS is of standard Westinghouse design for upper head injection (UHI) reactor systems and utilizes a microprocessor for data processing. The RVLIS uses differential pressure (DP) transmitters to measure the pressure drops from the bottom of the reactor vessel to the hot legs for UHI plants and from the hot legs to the top of the reactor vessel. Under natural circulation or no-circulation conditions, these pressure drops will provide indication of the collapsed liquid level or relative void content in the reactor vessel above and below the hot legs. Under forced-flow conditions, the pressure drops will provide indication of the vessel void content above the hot legs and the relative void content of the circulating primary coolant system fluid. Automatic compensation for changes in the temperature of the impulse lines leading from the reactor vessel and hot legs to the DP transmitters is incorporated in the system. Strap-on RTD's are mounted on the vertical runs of the impluse lines for measuring impulse-line temperatures. Automatic compensation for changes in the reactor coolant system fluid densities is also incorporated in the system. Following a hypothetical accident which causes a loss of primary coolant, the RVLIS will be used by the plant operators to assist in detecting a gas bubble or void in the reactor vessel and assist in detecting the approach to a condition of inadequate core *cooling. If forced-flow conditions are maintained after the accident, the RVLIS will also be used to assist in detecting the formation of void in the circulating primary coolant system fluid. The equipment which comprises the RVLIS includes the DP transmitters, impulse lines, impluse-line RTD's, in-containment sensor bellows units, out-of-containment hydraulic isolators, and all the necessary electronic signal conditioning, processing and display equipment. A technical description of the system appears in Westinghouse's manual entitled, "RVLIS - Summary Report, December, 1980."*

An item-by-item discussion of NUREG-0737, Item 1.8.1.30, is provided in Table 1-6.

<u>Incore Thermocouple System</u>

The present incore thermocouple system has 65 T/C's (thermocouples) positioned to sense exit flow temperature of selected fuel assemblies. The T/C's penetrate the reactor vessel head in 5 locations known as instrument ports. Each instrument port has 13 T/C's. Electrical connection to the T/C's is made at the instrument ports by qualified connectors. The class 1E thermocouples are cabled to qualified thermocouple penetrations. Forty of the thermocouple channels have been upgraded to insure a minimum of four per core quadrant are always available. The system design accounts for attrition. The remaining non 1 E T/C's are cabled to junction boxes inside containment to allow transition to copper for the remainder of the cabling including the run to an instrument penetration. Outside containment, the class 1 E T/C's are cabled to a class 1 E backup display directly from the T/C penetrations. These T/C's were cabled to the primary display using qualified isolation devices. The back up displays are alphanumeric/graphic plasma flat panel displays located on the main control boards in the Control Room. The Incore Thermocouple System configuration is shown on Figure 1-31 and Figure 1-35.

1. *System Capabilities (NUREG 0737 II.F.2 Attachment 1 format)*
   a. *Core inlet temperature data is used with core exit temperature to give radial distribution of coolant temperature rise across the core. This is available to the operator via video monitor or hard copy in the control room.*
   b. *The plant computer via video monitor is the operator's primary display having the following capabilities:*
      1) *A spatially oriented core map is available on demand indicating temperature at each core exit thermocouple location.*
      2) *The incore thermocouples are an input into the saturation monitor program to assist operator actions for inadequate core cooling procedures.*
      3) *Direct readout via video monitor and hard copy print out capability is provided for all thermocouple temperatures. This readout range extends from 0 degrees F. to 2300 degrees F. for safety related T/C's and 32 degrees F. to 2,300 degrees F. for non-safety related T/C's.*
      4) *Trending of selected thermocouples to show temperature - time history is available on demand.*
      5) *Alarm capabilities are provided thru the saturation monitor program.*
      6) *Addressed in the Control Room Design Review.*
   c. *A qualified plasma backup display is provided in the control room to read any of the thermocouples. The range of this backup display extends from 0 degrees F to 2,300 degrees F for safety related T/C's and 32 degrees F to 2,300 degrees F for non-safety related T/C's.*

A non-1E backup display is provided in the control room to read any of the thermocouples. With push-to-read switches, readings can be taken well within the six minute time guidance. The range of this backup display extends from 0 degrees F. to 2300 degrees F. for safety related T/C's and 32 degrees F. to 2,300 degrees F. for non-safety related T/C's.

d. *See Item 1b6 above.*

e. *The following consists of an evaluation of the Catawba Nuclear Station Incore Thermocouple System compliance with Appendix B to Section 1.8.1.30. The paragraph numbers relate directly to Appendix B paragraph numbering.*

   1) *The Class 1E instrumentation is environmentally qualified in accordance with Catawba FSAR Section 3.11 and the Duke Power Company NUREG-0588 submittal. The qualification applies from the sensor (qualification assumed per Item 1.8.1.30 guidance) to the final display device. For the primary display via the plant computer, qualification applies from the sensor to the isolation device. The isolation device would be accessible for maintenance during accident conditions.*

      The Class 1E instrumentation is seismically qualified in accordance with Catawba FSAR Section 3.10. This instrumentation would operate with the required accuracy after, but not necessarily during, a safe shutdown earthquake.

      Seismic qualification is not required for the primary display and associated hardware beyond the isolator.

   2) *No single failure within the incore thermocouple system or its supporting systems would prevent the operator from being presented with the information he would need in order to determine the safety status of the station and to bring the reactor to a safe, stable condition following an accident. This is feasible because the incore thermocouple system has two reliable portions: (a) the non-safety primary display portion which has a battery-backed power source, (b) the redundant Class 1E portion which has separate battery-backed power sources.*

      Additional diverse Class 1E indications of reactor coolant system pressures and temperatures are provided to assist the operator in the case of discrepancies in redundant read-outs.

      Redundant Class 1E channels are electrically independent, energized from separate power supplies, and are physically separated per FSAR Sections 7.1.2.2 and 8.3.1.4 up to and including the isolation device. Direct recording and trending capabilities are provided for any of the 65 incore thermocouple channels.

   3) *The Class 1E Incore Thermocouple Instrumentation and backup display is powered by Class 1E power sources.*

      The primary incore displays are powered by batterybacked power sources.

   4) *Incore instrumentation channel availability is addressed in the Catawba Technical Specifications.*

   5) *The provisions of Duke Power Company's Quality Assurance Program as described in FSAR Chapter 17 and Topical Report Duke 1A have been applied to the Class 1E portion of the Incore Thermocouple System. The primary display and other non-safety hardware beyond the isolation device are not required to be governed by this QA program. For further information on the Class 1E quality assurance provided, please consult the cited references.*

6) Indication and recording capabilities have been provided as specified in Duke's response to Supplement 1 to NUREG 0737, Regulatory Guide 1.97 Rev. 2 Section.

7) Same answer as 1e6.

8) Identification of the appropriate post-accident channels is performed as described in the response to Supplement 1 to NUREG 0737, Regulatory Guide 1.97 Rev. 2 Section.

9) Qualified isolation devices are utilized to isolate the Class 1E portions of the system from the non-safety portions as specified in RG 1.75.

10) Test capabilities are provided to check channel operational availability during reactor operation.

11) Servicing, testing, and calibration programs are specified to maintain the capabilities of the system.

12) Means for the removal of channels for maintenance are included in the design, and those means are under administrative control.

13) The Catawba incore design facilities the administrative control of the access to setpoint adjustments, calibration adjustments, and test points.

14) The Catawba design minimizes the existence of conditions which could lead to anomalous indications and confusion of the operator.

15) The Catawba design facilities the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.

16) All incore instrumentation system inputs are from sensors which directly measure the desired variables (core exit temperatures).

17) To the extent practical, the same instruments are utilized for accident monitoring as are used for normal operations of the station.

18) Periodic testing of the incore instrumentation channels is in accordance with position on "Regulatory Guide 1.118".

f. The primary and backup displays are energized from independent battery-backed power sources. The back display and associated hardware are supplied with Class 1E power. Due to physical constraints in the reactor vessel head area configuration, full separation as defined in FSAR Sections 7.1.2.2 and 8.3.1.4 cannot be attained. The maximum practical separation is provided in this area and mineral insulated cabling is used to enhance separation and integrity. Once the cabling leaves the refueling canal area separation as specified in FSAR Sections 7.1.2.2 and 8.3.1.4 is maintained for the entire remainder of the system cabling.

g. The Class 1E T/C instrumentation (T/C qualification assumed) is seismically and environmentally qualified up to and including the isolation device. Seismic qualification is consistent with the methodologies described in Section 3.10. Instrumentation subject to a harsh environment is environmentally qualified consistent with the Duke Power Company position on the Category II Guidelines of NUREG 0588 as detailed in the Duke submittal of June 30, 1982. The isolation device would be in an accessible area following an accident.

h. The availability of the Class 1E back-up displays will be addressed in the Technical Specifications.

i. The provisions of Duke Power Comany's Quality Assurance Program as described in FSAR Chapter 17 and Topical Report Duke 1A was applied to the Class 1E portion of the Incore Thermocouple System. The primary display and other non-safety hardware beyond the isolation device are not required to be governed by this QA program. For further information on the Class 1E quality assurance provided, please consult the cited references.

Procedures

See Section 13.5.

1.8.1.31 Emergency Power for Pressurizer Equipment (II.G)

Pressurizer PORV

The pressurizer power-operated relief valves are air-operated with DC control solenoids. Power for the solenoid valves is supplied from the 125VDC Vital Instrumentation and Control Power System (See Section 8.3.2). The solenoid operators and their controls are safety-related.

Pressurizer PORV Block Valves

The pressurizer PORV block valves are motor-operated valves with both motive and control power supplied from the 600VAC Essential Auxiliary Power System (See Section 8.3.1). The block valves including their power and control circuits are safety-related.

Pressurizer Level Indication

Three redundant channels of pressurizer level instrumentation are provided. These channels are part of the safety-related portion of the Process Control System which receives its power from the Vital Instrumentation and Control Power System (See Section 8.3.2).

1.8.1.32 IE Bulletins on Measures to Mitigate Small-Break LOCAs and Loss of Feedwater Accidents (II.K.1)

1.8.1.32.1 (C.1.5)

During the planning and procedure development stage of the integrated Engineered Safety Features (ESF) test a complete review of all valves receiving a safety injection actuation signal and containment isolation signal is conducted. This review primarily evaluates the response time requirements for each of these valves. However, in order to verify the response times, valve positioning requirements are also reviewed. As a result of this review and the successful performance of the integrated ESF test, direct verification of correct valve positioning requirements and valve positions under ESF conditions are obtained. Specific valve requirements and closure times are given in Table 3-104.

Correct valve positioning requirements, valve positions, and valve response times are verified during the ESF test via the Operator Aid Computer (OAC). The correctness of the OAC indication is verified through the use of operating procedures which require visual verification that the valve position indication in the control room and on the OAC is identical to the actual valve position. These operating procedures are required to be performed on all ESF valves after any maintenance activities which could affect proper operation of the valve.

1.8.1.32.2 (C.1.10)

Procedures for repositioning valves following maintenance or test activities provide assurance that these valves are returned to their correct position. These procedures require verification of the operability of a redundant system prior to the removal of any safety-related system from service, verification of the operability of all safety-related systems when they are returned to service, and notification of the reactor operators whenever a safety-related system is removed from and returned to service.

The operability of redundant systems and safety-related systems is verified by performing an initial functional test and subsequent periodic tests. A Removal and Restoration procedure governs the repositioning of valves following these tests and following any maintenance activities performed on these valves. This procedure utilizes a formal checklist to provide assurance that these systems are properly aligned.

Notification of operators when safety-related systems are removed from, or returned to, service is accomplished by the use of Removal and Restoration checksheets, red tags and red tag logbook, white tags and white tag logbook, out of service stickers, and the 1.47 bypass panel. Log entries denoting the removal and restoration are made in the Reactor Operator's Log. All of the above documents are reviewed during shift turnovers.

The Catawba Work Request Program governs all maintenance activities performed at Catawba. These work requests describe the maintenance to be performed and the procedures for performing it. Upon completion of the maintenance all work requests are entered into the corporate computer. This program provides for portable historical records of all maintenance performed on safety-related systems.

1.8.1.32.3 (C.1.17)

The design of Catawba Nuclear Station does not feature safety injection initiation on coincident pressurizer level and pressure signals. Safety injection is initiated whenever the low pressurizer pressure trip setpoint is reached independent of pressurizer level (See Section 7.3).

1.8.1.33 Commission Orders on B&W Plants (II.K.2)

1.8.1.33.1 THERMAL MECHANICAL REPORT - Effect of High-Pressure Injection on Vessel Integrity for Small-Break Loss-of-Coolant Accident with No Auxiliary Feedwater (II.K.2.13)

WCAP-10019 which addresses the NRC requirements of detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater was submitted to the NRC on December 30, 1981 (OG-66). This WCAP was developed under the sponsorship of the Westinghouse Owners Group (WOG). On March 23, 1982 WOG letter OG-68 was submitted to the NRC which described the additional effort underway to resolve NRC comments and questions concerning WCAP-10019. Results of the program to date show that operating plants can withstand the limiting transients for the expected life of their vessels.

1.8.1.33.2 Potential for Voiding in the Reactor Coolant System During Transients (II.K.2.17)

Westinghouse (in support of the Westinghouse Owners Group) has performed a study which addresses the potential for void formation in Westinghouse designed nuclear steam supply systems during natural circulation cooldown/depressurization transients, consistent with Generic

Letter 81-21. This study has been submitted to the NRC by the Westinghouse Owners Group (Letter OG-57, dated April 20, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners Group) to P.S. Check (NRC)) and is applicable to Catawba Nuclear Station.

In addition, the Westinghouse Owners Group has developed a natural circulation cooldown guideline that takes the results of the study into account so as to preclude void formation in the upper head region during natural circulation cooldown/depressurization transients, and specifies those conditions under which upper head voiding may occur. These Westinghouse Owners Group generic guidelines have been submitted to the NRC (Letter OG-64, dated November 30, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners Group) to D. G. Eisenhut (NRC)). The generic guidance developed by the Westinghouse Owners Group (augmented as appropriate with plant specific considerations) will be utilized in the implementation of Catawba plant specific operating procedures.

1.8.1.33.3 Sequential Auxiliary Feedwater Flow Analysis (II.K.2.19)

Subsequent to the issuance of NUREG-0737 and as documented in Reference 3, the NRC has completed a generic review on this subject and concluded that the concerns expressed in Item 1.8.1.33.3 are not applicable to NSSSs with inverted U-tube steam generators such as those designed by Westinghouse. Therefore, this item is not applicable to Catawba and no further action is necessary.

1.8.1.34 Final Recommendations of B&O Task Force (II.K.3)

1.8.1.34.1 Installation and Testing of Automatic Power-Operated Relief Valve Isolation System (II.K.3.1)

Based on the reduction in PORV LOCA frequency due to post-TMI modifications already implemented, an automatic PORV block valve closure system is unnecessary and therefore not incorporated. See the response to Item 1.8.1.34.2 below and WCAP-9804 (Reference 8) for further discussion.

1.8.1.34.2 Report on Overall Safety Effect of Power-Operated Relief Valve Isolation System (II.K.3.2)

The Westinghouse Owners Group submitted WCAP-9804 (Reference 8), "Probabilistic Analysis and Operational Data in response to Item 1.8.1.34.2 for Westinghouse NSSS Plants" to the NRC on March 13, 1981 (Letter OG-52). This report describes various modifications to Westinghouse plants since TMI and, using probabilistic analysis via event trees, estimates the effect of the post-TMI changes, including an automatic PORV isolation concept identified in NUREG-0737 Item 1.8.1.34.1. The requested safety valve operational data is included in this report.

The following modifications have been made to the Catawba PORV/Safety Valves:

1. *PID controller modification (derivative time constant set to zero).*
2. *Safety valve position indication/flow installed (acoustic monitoring system).*
3. *PORV position-indicating limit switches replaced with environmentally qualified or sealed limit switches.*
4. *Circuitry modified to provide control room annunciation from PORV limit switch or to provide alarm on PORV open position.*

Therefore, the improvements in reactor safety demonstrated by WCAP-9804 are applicable to Catawba.

1.8.1.34.3 Reporting Safety Valve and Relief Valve Failures and Challenges (II.K.3.3)

Duke Power Company will promptly report to the NRC any failure of a Catawba PORV or safety valve to close. In addition, all challenges to the PORV's or safety valves will be documented and reported annually to the NRC.

1.8.1.34.4 Automatic Trip of Reactor Coolant Pumps During Loss-of-Coolant Accident (II.K.3.5)

The Westinghouse Owners Group (WOG) submitted two reports to the NRC in response to the Westinghouse specific Generic Letters 83-10c and d. The first report, WOG Letter OG-110, December 1, 1983, provided an "Evaluation of Alternate RCP Trip Criteria". The second report, WOG Letter OG-117, March 9, 1984, provided the "Justification of Manual RCP Trip for Small Break LOCA Events". The WOG also provided additional information, WOG Letter OG-137, "Response to NRC Questions on RCP Trip", October 25, 1984, in response to an NRC request for this information, based on the review of the first two submittals.

The NRC issued a "Safety Evaluation by the office of Nuclear Reactor Regulation for the Westinghouse Owners' Group Reactor Coolant Pump Trip" as an attachment to Generic Letter 85-12, June 28, 1985. According to this safety evaluation, the information provided by the WOG 1) for the justification of manual RCP trip, and 2) in support of the alternative RCP trip criteria, is acceptable. The NRC also concluded that the WOG has developed acceptable criteria for tripping the RCPs during small-break LOCAs and to minimize RCP trip for SGTR and non-LOCA events.

Generic Letter 85-12 also requested plant specific information regarding implementation of RCP trip criteria including information on instrumentation uncertainties, potential RCP trip problems, and operator training and procedures. Duke Power Co. responded to this request by a letter from H. B. Tucker to H. L. Thompson, Jr., NRC, August 22, 1985.

1.8.1.34.5 Proportional Integral Derivative Controller Modification (II.K.3.9)

Westinghouse has completed its review of the pressure integral derivative (PID) controller installed on the Catawba PORVs and a value of "zero" for the pressurizer PID controller rate time constant was determined. The Catawba time constant has been adjusted accordingly.

1.8.1.34.6 Proposed Anticipatory Trip Modification (II.K.3.10)

By letter dated July 26, 1982, from W. O. Parker, Jr. to H. R. Denton, Duke Power transmitted an analysis which demonstrated the acceptability of bypassing the reactor trip on turbine trip at power levels below 70%.

1.8.1.34.7 Justification for Use of Certain PORV'S (II.K.3.11)

See Item 1.8.1.22.

1.8.1.34.8 Confirm Existence of Anticipatory Reactor Trip Upon Turbine Trip (II.K.3.12)

See Section 7.2.

1.8.1.34.9 Report on Outages of Emergency Core-Cooling Systems Licensee Report and Proposed Technical Specification Changes (II.K.3.17)

As discussed in Generic Letter 83-37, the NRC has completed their review of EECS data provided by licensees and determined that no changes in the Technical Specifications are required. It is therefore Duke Power's conclusion that the surveillance and reporting requirements in the Catawba Technical Specifications adequately address outages of ECCS systems and components, and no further action is required.

1.8.1.34.10 Effects of Loss of Alternation-Current Power on Pump Seals (II.K.3.25)

At the Catawba Nuclear Station the reactor coolant pump seal water is supplied by the charging pumps and cooled by component cooling water. Nuclear service water in turn cools the component cooling water. In the event of a loss of offsite power, the component cooling water pumps, the nuclear service water pumps, and the charging pumps are all supplied with emergency power from the emergency diesel generators.

1.8.1.34.11 Revised Small-Break Loss-of-Coolant-Accident Methods to Show Compliance with 10 CFR PART 50, Appendix K (II.K.3.30)

This item requires that the analysis methods used by NSSS vendors and/or fuel suppliers for small-break LOCA analysis for compliance with Appendix K to 10 CFR Part 50 be revised, documented, and submitted for NRC approval.

Westinghouse feels very strongly and Duke agrees that the small-break LOCA analysis model currently approved by the NRC for use on Catawba is conservative and in conformance with Appendix K to 10 CFR Part 50. However, (as documented in Letter OG-60, dated June 15, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners Group) to P. S. Check (NRC), Westinghouse believes that improvement in the realism of small-break calculations is a worthwhile effort and has committed to revise its small-break LOCA analysis model to address. NRC concerns (e.g., NUREG-0611, NUREG-0623, etc.). This revised Westinghouse model was submitted to the NRC in a letter (NS-EPR-2681) dated November 21, 1982, from E. P. Rahe (W) to C. O. Thomas (NRC). NRC accepted the Westinghouse model for referencing on May 21, 1985. Generic analyses were performed by the Westinghouse Owners group and documented in WCAP-11145 (L.D. Butterfield letter to J. Lyons -0G-190, June 11, 1986). By letter dated July 1, 1986 (H.B. Tucker to H.R. Denton), Duke Power referenced WCAP-11145 for Catawba. By letter dated October 6, 1986 (C. E. Rossi (NRC) to L. D. Butterfield (W OG), the NRC accepted WCAP-11145 for referencing.

1.8.1.34.12 Plant-Specific Calculations to Show Compliance with 10 CFR 50.46 (II.K.3.31)

See Section 1.8.1.34.11 above.

1.8.1.34.13 Upgrade Emergency Preparedness (III.A.1.1)

See Section 13.3.

1.8.1.34.14 Upgrade Emergency Support Facilities (III.A.1.2)

See Section 13.3.

1.8.1.34.15 Primary Coolant Sources Outside Containment (III.D.1.1)

Periodic leak tests 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 Atmosphere Hydrogen Concentration Level Analyzer, Boron Recycle, Nuclear Sampling System, Chemical Volume and Control, Liquid Waste, and Waste Gas. These tests 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.

Separate periodic test procedures will be written to assure that excessive leakage is detected on a timely basis. These tests will be run at least weekly and will require that systems carrying radioactive fluids outside of containment be visually inspected for excessive leakage. Visual inspections in High Rad Areas (≥ 100 mR/hr) are not required to be performed. In addition to periodic test procedures, tank and sump monitoring, operator rounds, radiation protection surveys and proper system operation can be used to verify that gross leakage is not occurring. Appropriate corrective action will be taken if excessive leakage is detected.

1.8.1.34.16 In-Plant Radiation Monitoring (III.D.3.3)

See Section 12.5.3.

1.8.1.34.17 Control Room Habitability (III.D.3.4)

See Section 6.4.

1.8.2 References

1. Muench, R., "Report on Small Break Accidents for Westinghouse Nuclear Steam Supply System (NSSS)," WCAP-9600 (Proprietary), and WCAP-9601 (Non-Proprietary), June 1979.

2. Docherty, P. J., and Gresham, J., "Report on Small Break Accidents for Westinghouse Nuclear Steam Supply System (NSSS) with Upper Head Injection (UHI)," WCAP-9639 (Non-Proprietary), December 1979.

3. Thompson, C. M., et al, "Inadequate Core Cooling Studies of Scenarios with Feedwater Available Using the NOTRUMP Computer Code," WCAP-9753 (Proprietary), and WCAP-9754 (Non-Proprietary), June 1980.

4. Tauche, W., "Loss of Feedwater Induced Loss of Coolant Accident Analysis Report," WCAP-9744 (Non-Proprietary), May 1980.

5. Hitchler, M. J., et al, "NUREG-0578 2.1.9.C Transient and Accident Analysis," WCAP-9691 (Non-Proprietary), March 1980.

6. Mark, R. H., and Thompson, C. M., "Inadequate Core Cooling Studies of Scenarios with Feedwater Available for UHI Plants, Using the NOTRUMP Computer Code," WCAP-9762 (Proprietary), June 1980.

7. NTD, Nuclear Safety Department, "Analysis of Delayed Reactor Coolant Pump Trip During Small Loss of Coolant Accidents for Westinghouse Nuclear Steam Supply System," WCAP-9584 (Proprietary), WCAP-9585 (Non-Proprietary), August 1979.

8. *Wood, D. C., Gottshall, C. L., "Probalistic Analysis and Operational Data in Response to Item II.K.3.2 for Westinghouse NSSS Plants," WCAP-9804, February 1981.*
9. *Meyer, T. A., "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants," WCAP-10019, December 1981.*

THIS IS THE LAST PAGE OF THE TEXT SECTION 1.8.

1.9 Response to Generic Letter 83-28

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

This section addresses "Required Actions Based on Generic Implications of Salem ATWS Events (Generic Letter 83-28)" and describes actions and programs related to the implementation of the requirements of each item in the Generic Letter. All actions have been implemented and accepted by the NRC (see references 35 and 36), although Status discussions in this section have not all been updated to reflect their closure status.

1.9.1 Post Trip Review—Program Description and Procedure (Item 1.1)

Requirement

The requirement as specified in the Generic Letter was to describe the program for ensuring that unscheduled reactor shutdowns are analyzed and that a determination is made that the plant can be restarted safely (Reference 1).

Response

The reactor trip investigation program was implemented at Catawba in order to determine causes of the trip, identify and assess any abnormal responses to the trip, ready the unit for restart, develop corrective actions to prevent recurrence, address any abnormal plant responses, and to satisfy reporting requirements.

The investigation program is initiated immediately following the trip (or other unscheduled shutdown) in order to ensure all available information is gathered and that any abnormal equipment performance will not impact continued safe operation of the plant. The program is divided into four distinct phases: post trip review, restart decision, independent review, and subsequent review. Each phase is discussed in detail in Reference 2.

The program is governed by a site directive with supporting guidance. The review is conducted by knowledgeable individuals familiar with plant design, operating characteristics, safety requirements, and plant specific transient behavior. These individuals are trained and experienced as discussed in Reference 2.

Status

The NRC has reviewed the post trip review program and procedures for Catawba and has found them to be acceptable. The NRC issued an SER (Reference 3) and the item is closed.

1.9.2 Post Trip Review—Data and Information Capability (Item 1.2)

Requirement

The Generic Letter (Reference 1) required licensees and applicants to have and to describe equipment to monitor and record plant parameters sufficient to correctly diagnose the cause of unscheduled reactor shutdowns and to verify proper functioning of safety-related equipment during an unscheduled shutdown using systematic safety assessment procedures (1.9.1).

Response

Each unit has three primary sources for collecting data for analyzing unscheduled reactor shutdowns. There are:

1. *Plant Computer Sequence of Events (SOE) Inputs,*
2. *Plant Computer Alarm Log, and*
3. *Plant Computer Transient Monitoring application.*

Items 1 and 2 are used to determine the sequence of events while Item 3 is used for analog data trending.

The (SOE) Inputs on the plant computer records various plant parameters including those pertinent to reactor trip investigations. When an input parameter alarms, the fact is stored to a plant computer log file with the precise time of the event. A representative list of parameters monitored is listed in Reference 2.

The alarm log is associated with the plant computer. The plant computer monitors analog and digital inputs and records to the alarm log all analog alarms and digital change of state alarms-for-safety-related pumps, valves, and motors.

The transient monitoring application is a program on the plant computer that records data, both before and after an event. The transient monitoring application is "tripped" or activated automatically when the reactor trip breaker opens. The data may be retrieved in graphic or tabular form for analysis. Parameters available on the transient monitor are listed in Reference 2.

In addition to the above equipment, information is available from control from strip charts, operator interviews, and control room logbooks. Post trip review data is maintained for the lifetime of the plant (Reference 4).

Status

The NRC has reviewed Duke's submittals (References 2, 4, 6), and issued a safety evaluation report (Reference 7). The SER closed Item 1.2 with the exception of the record retention issues. Subsequently, Duke established a category of "Lifetime" for post-trip review data (Reference 4), thus accepting the NRC's position. The issue remains open pending formal closure by the NRC.

1.9.3 Equipment Classification and Vendor Interface (Reactor Trip System Components) (Item 2.1)

Requirement

This item required the identification of components required to trip the reactor as safety-related on documentation controlling these components including maintenance procedures. Also required was the establishment and maintenance of a vendor interface program to ensure reliability of reactor trip system components by maintaining awareness of vendor information and recommendations and that these are incorporated into station procedures and manuals as appropriate (Reference 1).

Response

All components of the reactor trip system whose functioning is required to trip the reactor are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities.

Most of the reactor trip system components, including the reactor trip breakers, are supplied by Westinghouse. Duke has a continuing program in place for interface with Westinghouse which ensures that information concerning safety-related equipment is complete, current and controlled for the life of the plant.

Technical bulletins are received at Duke by a single coordinator who confirms receipt using a receipt acknowledgement form. At least annually, Westinghouse transmits a list of Technical Bulletins such that Duke may verify the receipt of all applicable information.

Upon receipt of a Technical Bulletin, it is transmitted to the appropriate Duke Power group to review for applicability and safety significance. If applicable, it is then transmitted to the station staff for implementation.

Status

The NRC has evaluated Duke's response to this item, and found that Duke's program is acceptable. (References 8, 9). The item is closed.

1.9.4 Equipment Classification and Vendor Interface (Programs for all Safety-Related Components) (Item 2.2)

Requirement

This item required the description of Duke's program for ensuring that all safety-related components are identified as such on all documentation used in the plant. The description should include the process to control the program, the utilization process, and the design verification and qualification programs.

Part two required the establishment and maintenance of a vendor interface program to ensure vendor information for safety-related components is complete, current, and incorporated into or referenced by appropriate station documents, and maintained for the life of the plant.

Response

The Catawba Nuclear Station Quality Standards Manual for Structures, Systems and Components provides guidance for the determination if a system or component is nuclear safety-related. The manual also contains tabular listings of systems, subsystems, components, and structures for which the determination has been done previously. The manual and its revisions are approved and issued by the Vice President, Nuclear Production Department, with appropriate administrative procedures implemented to control distribution. More detailed discussion may be found in Reference 2.

For part two, vendor interface, Duke is a participant in the Nuclear Utility Task Action Committee (NUTAC) established by the Institute for Nuclear Power Operations (INPO) which developed the Vendor Equipment Technical Information Program (VETIP). The VETIP is fully described in Reference 10. Further Duke Power implementation of the VETIP includes the Nuclear Plant Reliability Data System (NPRDS) and the Significant Event Evaluation and Information Network (SEE-IN) as described in Reference 11.

Duke's Nuclear System Directives (NSD) established procedures to ensure that safety-related technical information is reviewed, evaluated, and resolved. The NSD also requires the evaluation of all safety-related equipment failures. These requirements are further discussed in Reference 11 and the NSD.

Status

The NRC has evaluated Duke's responses to part one (References 2, 12, 13) and has found Duke's response to be acceptable and part one is closed. (Reference 14).

The NRC has also evaluated Duke's responses to part two (References 2, 11) and has found Duke's response to be acceptable and closed the item (Reference 14).

1.9.5 Post-Maintenance Testing (Reactor Trip System Components) (Item 3.1)

Requirement

The requirement specified that licensees (and applicants) review test and maintenance procedures and the Technical Specifications to assure post maintenance operability testing of safety-related components in the reactor trip system is required and that the testing demonstrates component operability. Vendor and engineering recommendations should also be reviewed and included as appropriate. The Technical Specifications should also be reviewed to identify any post maintenance test requirements that degrade rather than enhance safety, and propose any changes that may be necessary (Reference 1).

Response

Existing procedures and programs require that all safety-related and technical specification-related components be tested after maintenance before being returned to service. This testing demonstrates the equipment is capable of performing its intended safety functions.

Duke has reviewed the vendor technical information for the Reactor Trip Breakers and has verified that the information is incorporated into plant procedures and Technical Specifications as appropriate.

Duke has reviewed Westinghouse recommendations for other Reactor Trip System Components and has ensured that they are incorporated into plant procedures as appropriate.

The Technical Specifications have been reviewed and no post maintenance testing requirements were found which may degrade safety (Reference 2).

Status

This item was reviewed and found to be acceptable. This item is closed (References 16, 17).

1.9.6 Post-Maintenance Testing (All Other Safety-Related Components) (Item 3.2)

Requirement

The requirements in this item are the same as Section 1.9.5 except this item covers those safety-related components that are not a part of the Reactor Trip System.

Response

Existing procedures and programs require that all safety-related and Technical Specification-related components be tested after maintenance before being returned to service. This testing demonstrates that the equipment is capable of performing its intended safety function.

Administrative procedures are established to control the distribution of vendor manuals and to ensure incorporation into plant procedures as appropriate.

The Technical Specifications have been reviewed, and no requirements were found that degrade safety.

Status
This item was reviewed and found to be acceptable. This item is closed (References 16, 17).

1.9.7 Reactor Trip System Reliability (Vendor-Related Modifications) (Item 4.1)

Requirement
All vendor recommended reactor trip breaker modifications shall be reviewed to verify that either the modification has been implemented, or a written evaluation of the technical reasons for not implementing a modification exists.

Response
A Westinghouse letter dated March 31, 1983 recommended that several critical dimensions on the undervoltage trip attachment (UVTA) be verified. On April 21, 1983, a Westinghouse letter was issued requiring UVTA replacement with devices having modified shaft grooves for the retaining ring. This replacement was completed at Catawba before fuel loading and all UVTA's were verified to comply with the dimensional tolerances identified by Westinghouse.

No other vendor-related modifications have been identified for the Reactor Trip System components supplied by Westinghouse.

Status
This item was reviewed and found to be acceptable. This item is closed. (Reference 18).

1.9.8 Reactor Trip System Reliability (Preventative Maintenance and Surveillance Program for Reactor Trip Breakers) (Item 4.2)

Requirement
The Reactor Trip Breaker Preventative Maintenance and Surveillance programs were to be described. The programs are to include periodic maintenance including lubrication, housekeeping and other vendor recommendations, trending of parameters, life testing of the breakers, and periodic replacement of breakers or components consistent with demonstrated life cycles.

Response
Duke is committed to maintain the Reactor Trip Breakers in accordance with Westinghouse (vendor) recommendations. The manufacturer recommendations are detailed in “Maintenance Program Manual MPM-WOGRTSDS 416-01 for Westinghouse Type DS-416 Reactor Trip Circuit Breakers and Associated Switchgear.” Duke has taken exception to several of the activities in the manual but these activities were not related to the safety function of the breaker. Duke has received concurrence from Westinghouse for these items.

Duke considers that trending of data is not useful in predicting RTB component failure. The periodic maintenance procedures, performance tests and checks, as well as performance tolerance measurements are adequate to detect a degraded condition. However, it was *concluded that it would be less trouble to trend the requested data rather than continuing to argue the point. Duke will therefore be trending data for Catawba.*

Life cycle testing of the breakers and components has been performed by Westinghouse for the Westinghouse Owner's Group. Duke is committed to replacing breakers or components consistent with demonstrated service lives.

Status

Parts one and two of this item have been the subject of a great deal of correspondence (References 2, and 19 through 28) and have been closed. Parts three and four have not had any correspondence since the original response (Reference 2) and remain open.

1.9.9 Reactor Trip System Reliability (Automatic Actuation of Shunt Trip Attachment for Westinghouse and B&W Plants) (Item 4.3)

Requirement

The plant shall be modified by providing a safety-related automatic reactor trip system actuation of the breaker shunt trip attachments.

Response

A modification was implemented before fuel loading to provide automatic actuation of the reactor trip breaker shunt trip attachments. The modification was described in a letter from H. B. Tucker to Harold R. Denton dated April 18, 1983 and was discussed in a Duke Power-NRC staff meeting on April 19, 1983.

McGuire Safety Evaluation Report, Supplement 7, included a discussion of the modification. The modification will include an independent fusing design for safety-related and non-safety-related circuits similar to the design planned for McGuire.

The shunt trip attachments will be considered safety-related (Class 1E) for all future replacements, modifications, maintenance, and testing.

Status

The modifications were reviewed by the NRC and found acceptable as documented in Reference 29. The issue is considered closed.

1.9.10 Reactor Trip System Reliability (Improvements in Maintenance and Test Procedures for B&W Plants) (Item 4.4)

This item did not apply to Catawba, there is no response or other correspondence.

1.9.11 Reactor Trip System Reliability (System Functional Testing) (Item 4.5)

Requirement

All plants shall conduct on-line functional testing of the reactor trip system, including independent testing of the diverse trip features. For Westinghouse plants, the diverse trip features include the breaker undervoltage and shunt trip features. Plants should be able to perform on-line testing or provide justification or alternatives for assuring high reliability. Technical Specifications should be reviewed to determine that test intervals are consistent with *achieving high reactor trip system availability. For operating plants, changes to the Technical Specifications will be required.*

Response

The reactor trip system at Catawba is designed to allow on-line functional testing. Functional testing is performed for Catawba as described in the Technical Specifications.

Duke developed proposed changes to the Technical Specifications consistent with the conclusions of WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System". These changes were submitted to the NRC by letter dated July 22, 1985 and supplemented September 11, 1985. The changes were incorporated into the Technical Specifications by License Amendments 9 (Unit 1) and 2 (Unit 2).

Status

The NRC has reviewed Duke's submittals and found them to be adequate. Items 4.5.1 and 4.5.2 are closed (References 17 and 33, respectively). Changes to the Technical Specifications have been submitted (References 30, 31) and approved (Reference 32); however, Item 4.5.3 has not been formally closed.

1.9.12 Implementation Inspection

A special inspection was conducted July 7-11, 1986 by the NRC to assess Duke's compliance with Duke's response to Generic Letter 83-28. The inspection is documented in Inspection Report numbers 50-413/86-26 and 50-414/86-29 (Reference 34).

The inspection team examined Duke's implementation of the post-trip review program at Catawba including procedures and personnel interviews. The team verified the implementation of the program and the qualifications of the personnel involved. No discrepancies were identified.

The inspectors reviewed equipment classification at Catawba; this included a review of the "Catawba Nuclear Station Quality Standards Manual for Structures, Systems, and Component" (generally referred to as the QSM) and its implementation. The team also reviewed qualified reviewer training. One item was noted (an Inspector Follow-up Item) concerning the QSM. Completed QA checklists were not being forwarded to manual holders and thus a single component may be reviewed several times, increasing the possibility of error. This is considered a weakness and will be reviewed again at a later date.

Post maintenance testing of the reactor trip circuit breakers was reviewed including observation of maintenance and testing by the inspector. The inspector discussed the use of handwritten procedures for testing of the reactor trip breakers, but did not question the actual tests and measurements being performed. The inspector did note the procedures were deficient and lacked QC Inspection. This is being tracked as an Inspector Follow-up Item.

The vendor interface program including the control of vendor technical information was reviewed. At the time of the inspection, the station was involved in an upgrade of the vendor manual control system, so the inspector could not review specifics of the system. As Duke was meeting stated commitments, no deficiencies were noted.

The inspector also verified vendor recommended modifications to the reactor trip breakers and associated documentation. The recommendations had been followed with no discrepancies noted.

In summary, the inspection verified the implementation of an effective program pursuant to Generic Letter 83-28 with no violations or deviations--only two inspector follow-up items were noted.

1.9.13 References

1. *Letter from Mr. D. G. Eisenhut (NRC) to all Licensees of Operating Reactors, Applicants for Operating License, and Holders of Construction Permits, (Generic Letter 83-28) dated July 8, 1983).*
2. *Letter from Mr. H. B. Tucker (DPC) to Mr. D. G. Eisenhut (NRC), Response to Generic Letter 83-28, dated November 4, 1983.*
3. *Letter from Mr. T. M. Novak (NRC) to Mr. H. B. Tucker (DPC), transmitting SER for Generic Letter 83-28, Item 1.1, dated June 21, 1985.*
4. *Letter from Mr. H. B. Tucker (DPC) to Mr. H. R. Denton (NRC), dated October 1, 1986.*
5. *Letter from Ms. E. G. Adensam (NRC) to Mr. H. B. Tucker (DPC), transmitting Draft TER for Catawba, dated May 16, 1985.*
6. *Letter from Mr. H. B. Tucker (DPC) to Mr. H. R. Denton (NRC), dated August 23, 1985.*
7. *Letter from Mr. T. M. Novak (NRC) to Mr. H. B. Tucker (DPC), transmitting SER for Generic Letter 83-28 Item 1.2 for McGuire and Catawba Nuclear Stations, dated June 21, 1985.*
8. *Letter from Mr. K. N. Jabbour (NRC) to Mr. H. B. Tucker (DPC), transmitting SER for Generic Letter 83-28, Item 2.1 (Part 1) for McGuire and Catawba Nuclear Stations, dated July 16, 1986.*
9. *Letter from Mr. K. N. Jabbour (NRC) to Mr. H. B. Tucker (DPC), transmitting SER for Generic Letter 83-28, Item 2.1 (Part 2) for Catawba Nuclear Station, dated June 10, 1987.*
10. *Letter from Mr. E. P. Griffing (NUTAC Chairman) to Mr. T. Alexion (NRC), transmitting NUTAC Report, dated March 23, 1984.*
11. *Letter from Mr. H. B. Tucker (DPC) to Mr. D. G. Eisenhut (NRC), dated May 7, 1984.*
12. *Letter from Mr. H. B. Tucker (DPC) to Mr. D. G. Eisenhut (NRC), dated February 2, 1984.*
13. *Letter from Mr. H. B. Tucker (DPC) to the NRC, Attention: Document Control desk, transmitting Duke response to NRC Request for Additional Information (NRC RAI dated February 12, 1987) dated March 30, 1987.*
14. *Letter from Mr. K. N. Jabbour (NRC) to Mr. H. B. Tucker (DPC), transmitting SER for Generic Letter 83-28, Items 2.2.1 and 2.2.2, dated October 26, 1987.*
15. *Letter from Mr. H. B. Tucker (DPC) to Mr. H. R. Denton, Attention: Mr. B. J. Youngblood (NRC) regarding outstanding Generic Letter 83-28 Commitments, dated December 3, 1985.*
16. *NUREG-0954 Supplement 5 ("SER Related to the Operation of Catawba Nuclear Station, Units 1 and 2") February 1986, Appendix K.*
17. *Letter from Mr. K. N. Jabbour (NRC) to Mr. H. B. Tucker (DPC), transmitting SER for Generic Letter 83-28 Items 3.1.1, 3.1.2, 3.2.1, 3.2.2, and 4.5.1, dated July 29, 1987.*
18. *Letter from Mr. K. N. Jabbour (NRC) to Mr. H. B. Tucker (DPC), transmitting SER for Generic Letter 83-28 Items 4.1 and 4.2 (Parts 1 and 2), dated October 26, 1987.*

19. *Letter from Mr. H. B. Tucker (DPC) to Mr. D. G. Eisenhut (NRC), transmitting additional information for Generic Letter 83-28 Items 3.1.2 and 4.2, dated December 31, 1984.*
20. *Letter from Mr. E. G. Adensam (NRC) to Mr. H. B. Tucker (DPC), transmitting RAI for Items 4.1, 4.2.1, and 4.2.2 of Generic Letter 83-28, dated June 4, 1985.*
21. *Letter from Mr. H. B. Tucker (DPC) to Mr. H. R. Denton, Attention: Mr. E. G. Adensam, transmitting response to Reference 20, dated June 24, 1985.*
22. *Letter from Mr. H. B. Tucker (DPC) to Mr. H. R. Denton, Attention: Mr. B. J. Youngblood (NRC) providing status of and schedule for Generic Letter 83-28 Items, dated December 3, 1985.*
23. *Letter from Mr. B. J. Youngblood (NRC) to Mr. H. B. Tucker (DPC) transmitting draft SER for Items 4.1, 4.2.1, and 4.2.2 of Generic Letter 83-28, dated June 10, 1986.*
24. *Letter from Mr. H. B. Tucker (DPC) to Mr. H. R. Denton, Attention: Mr. B. J. Youngblood, providing clarifications and corrections to Reference 21, dated July 2, 1986.*
25. *Letter from Mr. H. B. Tucker (DPC) to Mr. H. R. Denton, Attention: Mr. B. J. Youngblood (NRC), transmitting response to Reference 23, dated August 11, 1986.*
26. *Letter from Mr. D. S. Hood (NRC) to Mr. H. B. Tucker (DPC), transmitting RAI for Item 4.2 (Part 2) of Generic Letter 83-28, dated July 23, 1987.*
27. *Letter from Mr. H. B. Tucker (DPC) to NRC Document Control Desk, responding to Reference 26, dated August 21, 1987.*
28. *Letter from Mr. K. N. Jabbour (NRC) to Mr. H. B. Tucker (DPC), transmitting SER for 4.1 and 4.2 (Parts 1 and 2) of Generic Letter 83-28, dated October 26, 1987.*
29. *NUREG-0954, supplement 4 ("SER Related to the Operation of Catawba Nuclear Station, Units 1 and 2"), December 1984.*
30. *Letter from Mr. H. B. Tucker (DPC) to Mr. H. R. Denton, Attention: Ms. E. G. Adensam (NRC), submitting proposed changes to the McGuire and Catawba Nuclear Station Technical Specifications, dated July 22, 1985.*
31. *Letter from Mr. H. B. Tucker (DPC) to Mr. H. R. Denton, Attention: Ms. E. G. Adensam (NRC), providing supplemental information for Reference 25.*
32. *Letter from Mr. K. N. Jabbour (NRC) to Mr. H. B. Tucker (DPC), transmitting Facility Operating License Amendments 9 (Unit 1) and 1 (Unit 2), dated September 8, 1986.*
33. *Letter from Mr. D. S. Hood (NRC) to Mr. H. B. Tucker (DPC), transmitting SER for Item 4.5.2 of Generic Letter 83-28, dated January 27, 1987.*
34. *IE Inspection Report 50-413/86-26 and 50-414/86-29, transmitted by Mr. V. L. Brownlee's (NRC) letter to Mr. H. B. Tucker (DPC), dated August 26, 1986.*
35. *Duke letter to the NRC dated December 18, 1997, License Amendment Request.*
36. *NRC letter from Peter S. Tam (NRC) to Gary R. Peterson (Duke) dated April 23, 1998, Issuance of Amendments to Facility Operating Licenses for Catawba Units 1 and 2.*

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