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1.0 Introduction and General Description of Station

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1.1 Introduction

This Final Safety Analysis Report is submitted in support of Duke Energy's application for facility operating licenses for the two-unit McGuire Nuclear Station located on the shore of Lake Norman in Mecklenburg County, North Carolina. The station is shown on Duke's Service Area Map, Figure 1-1.

The site, located in a rural area, has no unusual characteristics. Anticipating the development of a nuclear station at this location, Duke submitted a preliminary report on this site to be reviewed informally by the Atomic Energy Commission staff in 1965, and no site features were found to be adverse.

The Nuclear Steam Supply System for each unit is a pressurized water reactor with four coolant loops, similar to other systems that were under construction at the time. The Nuclear Steam Supply System and the design and fabrication of initial cores are supplied by Westinghouse Electric Corporation. Replacement steam generators were provided by Babcock & Wilcox International. The original turbine-generators for each unit were provided by Westinghouse. Turbine-generator upgrades were performed by Siemens Energy, Inc. Each containment consists of a freestanding cylindrical steel structure enclosed by a separate reinforced concrete Reactor Building. The Containment and Reactor Building are designed by Duke.

Each generating unit is authorized to operate at a reactor core full steady state power level of 3469 megawatts thermal (100%) (see Facility Operating Licence conditions 2C(1)). The expected net electrical output is approximately 1185 MWe. All core physics and core thermal-hydraulic information are based on the reference core design of 3469 MWt. The Containment, Engineered Safety Features, and certain postulated accidents were evaluated during original plant licensing for a core rating of 3579 MWt (1239 MWe) which corresponds to the calculated maximum capacity of the Steam and Power Conversion System.

Construction was initiated on the Standby Nuclear Service Water Pond under an exemption granted by the AEC on June 23, 1971. An additional exemption granted on December 22, 1971, allowed construction to proceed on all below-grade structures, and construction permits were granted on February 28, 1973. Construction was completed in time for fuel loading of Unit 1 in January, 1981, and Unit 2 in January, 1982. Unit 1 began commercial operation in December, 1981, with Unit 2 in March, 1984. A License Renewal Application was applied for and accepted by the NRC which allows an extended period of operation for McGuire Unit 1 until midnight, June 12, 2041 and Unit 2 until midnight, March 3, 2043.

Duke is fully responsible for the complete safety and adequacy of the station, and consistent with long-standing practice, Company 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 location of a nuclear generating station as evidenced by a 2500-foot exclusion radius; remoteness from population centers; sound rock foundation for most major structures; freedom from flooding; an abundant supply of cooling water from Lake Norman; and favorable conditions of hydrology, geology, seismology and meteorology. The Standby Nuclear Service Water Pond, designed for the safe shutdown seismic conditions, provides a reliable, ultimate heat sink for rejection of decay heat under accident conditions.

1.2.2 Station Description

1.2.2.1 Principal Design Criteria

The principal design criteria have been developed to comply with the AEC's "General Design Criteria for Nuclear Power Plants," Appendix A to 10CFR 50 (May 21, 1971). 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 1-17.

1.2.2.3 Nuclear Steam Supply System

Each Nuclear Steam Supply System consists of a pressurized water reactor with a four-loop Reactor Coolant System. The mechanical, thermal-hydraulic, and nuclear design of the reactor core is similar to other systems operating or under construction, as indicated in Section 1.3.

The reactor core is composed of uranium dioxide pellets enclosed in Zircaloy tubes with welded end plugs. The tubes are supported in assemblies by spacer grid assemblies and the upper and lower end fitting assemblies. The reactor core is initially loaded in regions of three different enrichments. The rod cluster control assemblies consist of clusters of stainless steel clad neutron absorber rods inserted into guide tubes located within the fuel assemblies.

The four steam generators are vertical shell and U-tube heat exchangers with integral moisture separating equipment.

An electrically heated pressurizer establishes and maintains the reactor coolant pressure. The pressurizer also provides a surge chamber and a water reserve to accommodate reactor coolant volume changes during operation.

The reactor coolant pumps are vertical, single speed, centrifugal units equipped with controlled leakage shaft seals.

A complete description of the Nuclear Steam Supply System is presented in Chapter 4 and Chapter 5.

1.2.2.4 Engineered Safety Features

Engineered Safety Features are provided to prevent accident propagation or to limit the consequences of postulated accidents which might otherwise lead to damage of the system and release of fission products. The principal criterion is to limit the potential off-site radiation dose from a Design Basis Accident 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 forming 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 System 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 during accident conditions and helps relieve post-accident thermal expansion of air in the annulus.

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

The Hydrogen Control Systems protect the Containment from an excess accumulation of combustible gas following a loss-of-coolant accident (refer to Sections 6.1.7, 6.2.5, 6.2.6).

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 Containment 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 Engineered Safety Features are separate and independent for each unit. Sufficient redundancy for each system is provided to assure proper functioning, which satisfies 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 station is electrically separated from the transmission system.

The Reactor Protection and Engineered Safety Features Actuation Systems automatically initiate appropriate action whenever the parameters monitored by these systems reach pre-established setpoints. These systems act to trip the reactor, actuate core cooling, close isolation valves and intiate the operation of standby systems as required.

1.2.2.6 Electrical and Emergency Power Systems

Normal power supply to the station auxiliaries is from the generator bus through an auxiliary transformer. Each unit is provided with two independent off-site power supplies capable of supplying power to Engineered Safety Features, and each of the two nuclear units is provided with two independent on-site emergency power supplies.

Each nuclear unit has two redundant and independent electric power distribution systems to supply electric power to the Engineered Safety Features equipment. Each of these two electric power distribution systems per unit has three power supplies as follows:

1. The 230 kV transmission network through two auxiliary transformers for Unit 1.
2. The 525 kV transmission network through two auxiliary transformers for Unit 2.
3. Two independent diesel-electric generators arranged to supply its distribution system.

In the event one of the circuits described in (1) and (2) above is unavailable, a manual cross tie between Unit 1 and Unit 2 is provided.

1.2.2.7 Instrumentation and Control

Extensive instrumentation and control is incorporated into station design to assure safe, efficient operation. A significant portion of this instrumentation and control is used to assure proper unit control (see Section 1.2.2.5) and proper operation of the electrical and emergency power systems (see Section 1.2.2.6). The balance of the instrumentation and control is interfaced with the station auxiliary and support systems to assure proper control and monitoring. Instrumentation and control is covered in detail in Chapter 7 and in the applicable chapters which describe the mechanical systems.

1.2.2.8 Steam and Power Conversion System

The steam and power conversion systems for each unit are designed to remove heat energy from the reactor coolant, deliver it in the form of steam to the turbine-generator, and convert it to electric 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

Both new and spent fuel are stored in the fuel pool and transferred to and from the Containment via the fuel transfer tube. Spent fuel is handled and stored under water. Separate fuel pools are provided for each of the two units. The system is designed to prevent inadvertent criticality and to minimize the possibility of mishandling or faulty operation that could cause fuel assembly damage and/or fission product release. New fuel may also be stored dry in the new fuel storage facilities.

1.2.2.10 Cooling Water

Condenser cooling water is taken from Lake Norman and discharged back into Lake Norman in an area removed from the intake to allow for sufficient dissipation of the heated water. An ample supply of cooling water is assured by Lake Norman which has a volume of 1,093,000 acre feet and an average flow of 2670 cfs at Cowans Ford Dam.

1.2.2.11 Radioactive Waste Management

The Waste Gas System is designed to collect, filter, monitor, store, and release as necessary, the gaseous effluent from processed reactor coolant. The Liquid Waste Recycle System and the Liquid Waste Monitor and Disposal System include capability for collection, storage, treatment, monitoring, disposal and recording of liquid wastes. Solid radioactive wastes (mechanical filters and spent resins) are packaged and stored on-site. Low-level dry active wastes (DAW) and secondary resins are packaged and shipped off-site for ultimate disposition at a licensed disposal facility.

1.2.2.12 Shared Facilities and Equipment

Separate and similar systems and equipment are provided for each unit of the two unit McGuire Nuclear Station except as noted in Table 1-1. In these instances where some components of a system are shared by both units, only those components which are shared are shown. (Note: This section is not applicable to 10CFR50 Appendix A General Design Criterion 5, Sharing of Structures, Systems, and Components.)

Although systems and equipment for Units 1 and 2 are functionally identical, the tables, figures, descriptions, etc., in this document generally reflect Unit 1 with minor differences existing between the two units. Major differences between the two units are discussed in detail.

1.2.2.13 Standby Shutdown Facility

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 for up to 72 hours 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 McGuire, in the event that normal and emergency systems are rendered inoperable. The facility is credited with the ability to cope with a station blackout (SBO) event of 4 hour duration. The facility is a steel-frame and masonry structure consisting of a diesel generator room, electrical equipment room, battery room, control room, and shared equipment. The facility, which is not safety-related, is designed in accordance with criteria for Category III structures (as defined in Table 3-1) and is not designed to withstand design basis seismic loadings.

1.2.2.14 FLEX Building Facilities

The FLEX Building Facilities are designed to store the portable equipment (generators, pumps, hoses, etc.) necessary to meet the requirements of NEI 12-06, Diverse and Flexible Coping Strategies (FLEX) Implementation Guide.

There are three FLEX buildings that meet the N+1 strategies, which are located outside the Proteted Area. The three FLEX buildings are shown in Figure 1-18 and are located to provide adequate separation between each other.

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1.3 Comparison Tables

1.3.1 Comparisons with Similar Facility Designs

Table 1-2 provides a listing of principal similarities and principal differences from other power reactor facilities with a Westinghouse NSSS. Chapter 4 provides further comparison data for the reactor core.

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 McGuire Nuclear Station. In addition to these changes, numerous items of design development have been incorporated in the design descriptions of the McGuire FSAR, where only criteria or functional requirements were stated in the PSAR.

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1.4 Identification of Agents and Contractors

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

The Engineering Department has the responsibility for specification of materials and equipment, design of the structures and systems and preparation of construction and installation drawings. The Commodities and Facilities Department has the responsibility for all site construction activities. The Nuclear Generation Department has the responsibility for preoperational testing and initial startup as well as operation of the station. 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.

Duke has over 75 years of experience in the design, construction and operation of electric generating stations.

Duke's work in nuclear power began in the early 1950's when selected Company personnel received nuclear training. Since 1955 Duke personnel have been involved full time in nuclear projects. Through Carolinas-Virginia Nuclear Power Associates, Duke participated in design and operation of the Carolinas-Virginia Tube Reactor in South Carolina, which produced electricity from 1963 to 1967, as part of a five-year operating research program. Many engineering personnel in the Duke organization have had prior nuclear experience as well as extensive experience in the power fields. Technical qualifications and nuclear experience of key Duke personnel are presented in Section 13.1.

Duke contracted with Westinghouse to design, manufacture and deliver to the site two complete Nuclear Steam Supply Systems and initial fuel cores. In addition, Westinghouse supplied technical assistance for erection of the Nuclear Steam Supply System 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.

Replacement steam generators were provided by Babcock & Wilcox International in Cambridge, Ontario, Canada.

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 was retained to assist in the seismic design of piping. In addition, they are 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 was retained by Duke to conduct investigations in geology, seismology, sub-surface conditions and foundations and groundwater hydrology. This work was conducted under the direction of

Professor George F. Sowers, Chairman of the Board of Law Engineering, and Regents Professor of Civil Engineering at Georgia Institute of Technology.

For Seismic design evaluation of electrical equipment, Duke retained Wyle Laboratories of Huntsville, Alabama and ITT Research Institute of Chicago, Illinois. Seismic simulation testing and analysis is conducted at Wyle. ITT Research Institute performed the seismic analysis on the McGuire emergency diesel generator.

Duke has also utilized the services of McNeary Insurance Consulting Services, Inc. of Charlotte, N.C., to assist in the area of fire protection at McGuire.

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1.5 Former Section 1.5

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1.6 Material Incorporated by Reference

1.6.1 Westinghouse Topical Reports

Topical Report	FSAR Section
Cohen, J., Development and Properties of Silver Base Alloys as Control Rod Materials for Pressurized Water Reactors, *WCAP-214*, December, 1959.	4.2
R. W. Miller, etal., "Relaxation of Constant Axial Offset Control," *WCAP-1026-PA* (Proprietary), June 1983.	4.3
Kramer, F. W., LOCO: A Digital Computer Code for Loss of Coolant Accident Analyses, *WCAP-1678*, January, 1961.	6.2.1
Olhoeft, J. E., The Doppler Effect for a Non-Uniform Temperature Distribution In Reactor Fuel Elements, *WCAP-2048*, July, 1962.	4.3
Novendstern, E. H., and Sandberg, R. O., Single Phase Local Boiling and Bulk Boiling Pressure Drop Correlations, *WCAP-2850*, April, 1966, (Westinghouse Proprietary), and *WCAP-7916*, June, 1972.	4.4
Balfour, M. G., Christensen, J. A., and Ferrari, H. M., In-Pile Measurement of UO_2 Thermal Conductivity, *WCAP-2923*, 1966.	4.4
Melehan, J. B., "Yankee Core Evaluation Program Final Report," *WCAP-3017-6094*, January 1971.	4.3
Hetsroni, G., Hydraulic Tests of the San Onofre Reactor Model, *WCAP-3269-8*, June, 1964.	4.4
Barry, R. F. LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094, *WCAP-3269-26*, Semptember, 1963.	4.3
Nodvik, R. J., Saxton Core II, Fuel Performance Evaluation, Part 1, Saxton Core II Fuel Performance Evaluation, Part II, *WCAP-3385-56*, July, 1970.	4.3, 4.4
Poncelet, C. G., and Christie, A. M., Xenon-Induced Spatial Instabilities in Large PWRs, *WCAP-3680-20*, (EURAEC-1974), March, 1968.	4.3
Skogen, F. B., and McFarlane, A. F., Control Procedures for Xenon Induced X-Y Instabilities in Large PWRs, *WCAP-3680-21*, (EURAEC-2111), February, 1969.	4.3
Skogen, F. B., and McFarlane, A. F., Xenon-Induced Spatial Instabilities in Three-Dimensions, *WCAP-3680-22*, (EURAEC-2116), September, 1969.	4.3
Cermak, J. O., et al, Pressurized Water Reactor pH - Reactivity Effect Final Report, *WCAP-3696-8*, (EURAEC-2704), October, 1968.	4.3
Leamer, R. D., et al, PUO_2 - UO_2 Fueled Critical Experiments, *WCAP-3726-1*, July, 1967.	4.3
Christensen, J. A., Allio, R. J., and Biancheria, A., Melting Point of Irradiated UO_2, *WCAP-6065*, February, 1965.	4.2, 4.4

Topical Report	FSAR Section
Poncelet, C. G., Burnup Physics of Heterogeneous Reactor Lattices, *WCAP-6069*, June, 1965.	4.4
Poncelet, C. G., LASER - A depletion Program for Lattice Calculation Based on MUFT and THERMOS, *WCAP-6073*, April, 1966.	4.3
Nodvik, R. J., et al, Supplementary Report on Evaluation of Mass Spectrometric and Radiochemical Analyses of Yankee Core 1 Spent Fuel, Including Isotopes of Elements Thorium Through Curium, *WCAP-6086*, August, 1969.	4.3
Chelemer, H., Weisman, J., and Tong, L. S., Subchannel Thermal Analysis of Rod Bundle Cores, *WCAP-7015*, Revision 1, January, 1969.	4.4
Moore, J. S., Power Distribution Control of Westinghouse Pressurized Water Reactors, *WCAP-7208*, September, 1968 (Westinghouse Proprietary) and *WCAP-7811*, December, 1971.	4.3
Barry, R. F., and Altomare, S., The TURTLE 24.0 Diffusion Depletion Code, *WCAP-7213-P-A*, February, 1975, *WCAP-7758-A*, February, 1975.	4.3
McFarlane, A. F., Core Power Capability in Westinghouse PWRs, *WACP-7267-L*, October, 1969 (Westinghouse Proprietary), and *WCAP-7809*, December, 1971.	4.3
Burnett, T. W. T., Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors, *WCAP-7306*, April, 1969.	7.2
Langford, F. L., and Nath, R. J., Jr., Evaluation of Nuclear Hot Channel Factor Uncertainties, *WCAP-7308-L*, April, 1969, (Westinghouse Proprietary) and *WCAP-7810*, December, 1971.	4.3
Shefcheck, J., Application of the THINC Program to PWR Design, *WCAP-7359-L*, August, 1969, (Westinghouse Proprietary), and *WCAP-7838*, January, 1972.	4.4
Fabic, S., Loss-of-Coolant Analysis: Comparison Between *BLODWN-2*, Code Results and Test Data, *WCAP-7401*, November, 1969.	3.9
Moore, J. S., Westinghouse PWR Core Behavior Following a Loss-of-Coolant Accident, *WCAP-7422*, September 1971.	3.9
Golik, M. A., Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems, *WCAP-7477-L*, (Proprietary), March, 1970.	5.2
Hazelton, W. S., Addendum 1 to Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems, *WCAP-7477-L* Add. 1, (Proprietary), May, 1971.	5.2
Katz, D. N., Solid-State Logic Protection System Description, *WCAP-7488-L*, March, 1971 (Westinghouse NES Proprietary); and *WCAP-7672*, May, 1971.	7.2, 7.3
Szy Slow Ski, J. J., and Salvatori, R., Determination of Design Pipe Breaks for the Westinghouse Reactor Coolant System, *WCAP-7503*, Rev. 1 February, 1972.	5.2

Topical Report	FSAR Section
Lipchak, J. B., and Bartholomew, R. R., Test Report Nuclear Instrumentation System Isolation Amplifier, *WCAP-7506-L*, October, 1970 (Westinghouse NES Proprietary); and *WCAP-7819*, Rev. 1 January, 1972.	7.2
Loving, J. J., In-Core Instrumentation (Flux-Mapping System and Thermocouples), *WCAP-7607*, July, 1971.	7.7
Shabbits, W. O., Dynamic Fracture Toughness Properties of Heavy Section Z 533 Grade B Class 1 Steel Plate, *WCAP-7623*, December, 1970.	5.2
Anderson, S. L., Clemons, L., Jr., Moser, J. S., Sejvar, J., Radiation Analysis Design Manual - 4 Loop Plant, *WCAP-7664*, Revision 1.	12.1
Cadek, F. F., Interchannel Thermal Mixing with Mixing Vane Grids, *WCAP-7667-P-A*, January, 1975 (Proprietary), and *WCAP-7755-A*, January, 1975.	4.4
Nay, J. A., Process Instrumentation for Westinghouse Nuclear Steam Supply Systems, *WCAP-7671*, April, 1971.	5.2
Motley, F. E., and Cedek, F. F., DNB Results for New Mixing Vane Grid (R), *WCAP-7695-P-A*, January 1975, (Proprietary); and *WCAP-7958-A*, January, 1975.	4.4
Motley, F. E., and Cedek, F. F., DNB Test Results for R Grids with Thimble Cold Wall Cells, *WCAP-7695*, Addendum 1-P-A, January, 1975, (Proprietary); and *WCAP-7958*, Addendum 1-A, January, 1975.	4.4
Haller, J. T., Engineered Safeguards Final Device or Actuator Testing, *WCAP-7705*, March, 1973.	7.3
Gangloff, W. C., and Loftus, W. D., An Evaluation of Solid-State Logic Reactor Protection in Anticipated Transients, *WCAP-7706-L*, (Westinghouse NES Proprietary) and *WCAP-7706*, February, 1973.	7.2, 7.3
Hazelton, W. S., Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems, *WCAP-7735*, August, 1971.	5.2
Igne, E. G., and Locante, J., Environmental Testing of Engineered Safety Features Related Equipment, *WCAP-7744*, Volume 1, August, 1971.	6.3
Barry, R. F., et al, The PANDA Code, *WCAP-7757*, September, 1971.	4.3
Cooper, K., Miselis, V., and Starek, R. M., Overpressure Protection for Westinghouse Pressurized Water Reactors, *WCAP-7769*, Revision 1, June, 1972.	5.2, 15.2
Blanchard, A. E., and Katz, D. N., Solid-State Rod Control System, Full Length, *WCAP-7778*, December, 1971.	7.7
Dollard, W. J., Nuclear Fuel Division Quality Assurance Program Plan, *WCAP-7800*, Revision 3, April, 1974.	4.2
Vogeding, E. D., Seismic Testing of Electrical and Control Equipment, *WCAP-7817*, December, 1971.	3.10, 7.5

Topical Report	FSAR Section
Vogeding, E. D., Seismic Testing of Electrical and Control Equipment, (WCID Process Control Equipment), *WCAP-7817*, Supplement 1, December, 1971.	7.5
Potochnik, L. M., Seismic Testing of Electrical and Control Equipment (Low Seismic Plants), *WCAP-7817*, Supplement 2, December, 1971.	3.10
Vogeding, E. L., Seismic Testing of Electrical and Control Equipment (Westinghouse Solid-State Protection System), (Low Seismic Plants), *WCAP-7817*, Supplement 3, December, 1971.	3.10, 7.5
Reid, J. B., Seismic Testing of Electrical And Control Equipment (NUCANA 7300 Series) (Low Seismic Plants) *WCAP-7817*, Supplement 4, November, 1972.	3.10
Vogeding, E. L., Seismic Testing of Electrical and Control Equipment (Engineered Safeguards Test Cabinet) (Low Seismic Plants), *WCAP-7817*, Supplement 7, September 1976.	3.10
Deleted Per 2008 Update.	
Bohm, G. J., Indian Point Unit No. 2 Internals Mechanical Analysis for Blowdown Excitation, *WCAP-7882*, December, 1971.	3.9
Bordelon, F., and Nahavandi, A., A Space-Dependent Loss of Coolant Accident and Transient Analysis for PWR System (SATAN Digital Computer Code), *WCAP-7845*, January, 1972.	5.2
Garber, I., Isolation Tests Process Instrumentation Isolation Amplifier Westinghouse Computer and Instrumentation Division Nucana 7300 Series, *WCAP-7862*, September, 1972.	7.2
Kraus, S., Neutron Shielding Pads, *WCAP-7870*, Including Appendix B, February, 1972.	3.9
Singleton, N. R., et al., Four Loop PWR Internals Assurance and Test Program, *WCAP-7879*, July, 1972.	3.9
McFarlane, A. F., Power Peaking Factors, *WCAP-7912-L*, March, 1971 (Westinghouse Proprietary) and *WCAP-7912*, March, 1972.	4.3, 4.4
Reid, J. B., Process Instrumentation for Westinghouse Nuclear Steam Supply System (4 Loop Plant Using WCID 7300 Series Process Instrumentation, *WCAP-7913*, January, 1973.	7.2, 7.3
Fabic, S., Description of the BLODWN-2 Computer Code, *WCAP-7918*, Revision 1, October, 1970.	3.9
Morrone, A., Damping Values of Nuclear Power Plant Components, *WCAP-7921-AR*, May, 1974.	3.7
Anderson, S. L., Hazelton, W. S., and Yanichko, E. E., Basis for Heatup and Cooldown Limit Curves, *WCAP-7924*, August, 1972.	5.2

Topical Report	FSAR Section
Cedek, F. F., and Motley, F. E., and Dominicis, D. F., Effect of Axial Spacing on Interchannel Thermal Mixing with R Mixing Vane Grid, *WCAP-7941-P-A*, January, 1975, (Proprietary); and *WCAP-7959-A*, January, 1975.	4.4
Gesinski, T. L., Fuel Assembly Safety Analysis for Combined Seismic and Loss-of-Coolant Accident, *WCAP-7950*, July, 1972.	3.7
Hochreiter, L. E., Chelemer, H., and Chu, P. T., THINC-IV, An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores, *WCAP-7956*, June, 1973.	4.4
Lee, J. C., et al, Axial Xenon Transient Tests at the Rochester Gas and Electric Reactor, *WCAP-7964*, June, 1971.	4.3
Motley, F. E., Cadek, F. F., "Application for Modified Spacer Factor to Grid Typical and Cold Wall Cell DNB," *WCAP-7988* (Westinghouse Proprietary), and *WCAP-8030-A* (Non-Proprietary), October 1972.	4.4
Blanchard, A. E., and Calpin, J. E., Digital Rod Position Indication, *WCAP-8014*, December, 1972.	7.7
Hochreiter, L. E., Application of the THINC-IV Program to PWR Design, *WCAP-8054*, October, 1973, (Westinghouse Proprietary), and *WCAP-8195*, October, 1973.	4.4
Ice Condenser Containment Pressure Transient Analysis Method, *WCAP-8077*, (Proprietary), March, 1973, and *WCAP-8078*, March, 1973.	6.2.1
Cloud, R. L., Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop, *WCAP-8082*, June, 1973 (Westinghouse NES Proprietary), and *WCAP-8173*, July, 1973.	3.6, 3.7, 5.2
Test Plans and Results for the Ice Condenser System *WCAP-8110*, April, 1973.	6.2
Test Plans and Results for the Ice Condenser System *WCAP-8110*, Supplement 1, April, 1973.	6.2
Test Plans and Results for the Ice Condenser System *WCAP-8110*, Supplement 2, June, 1973.	6.2
Test Plans and Results for the Ice Condenser System *WCAP-8110*, Supplement 3, July, 1973.	6.2
Test Plans and Results for the Ice Condenser System *WCAP-8110*, Supplement 4, November, 1973.	6.2
Ice Condenser System Lower Inlet Door Shock Absorber Test Plans and Results, *WCAP-8336*, May, 1974 (Westinghouse NES Proprietary), and *WCAP-8110*, Supplement 5, May, 1974.	6.2
Final Report-Ice Condenser Full-Scale Tests at the Waltz Mill Facility, *WCAP-8282*, February, 1974 (Westinghouse NES Proprietary), and *WCAP-8110*, Supplement 6, May, 1974.	6.1, 6.2

Topical Report	FSAR Section
Final Report-Ice Condenser Full-Scale Tests at the Waltz Mill Facility, *WCAP-8282*, Addendum 1, May, 1974 (Westinghouse NES Proprietary), and *WCAP-8110*, Supplement 7, May, 1974.	6.2
Stress and Structural Analysis and Testing of Ice Baskets, *WCAP-8304*, (Westinghouse NES Proprietary) and *WCAP-8110*, Supplement 8, May, 1974.	6.2
Ice Fallout From Seismic Testing of Fused Ice Basket, WCAP-8110, Supplement 9-A, May, 1974.	6.2
Deleted per 2002 Update.	
Static Testing of Production Ice Baskets, *WCAP-8110*, Supplement 10, September, 1974.	6.2
Hill, K. W., Motley, F. E., and Cadek, F. F., Effects of Local Heat Flux Spikes on DNB in Non-Uniform Heated Rod Bundles, *WCAP-8174*, August, 1973, (Westinghouse Proprietary), and *WCAP-8202*, August, 1973.	4.4
Ploudo, V. J., Schreiber, R. E., Skaritka, J., Operational Experence with Westinghouse Cores, *WCAP-8183*, Latest Revision.	4.2, 11.1
Schreiker, R. E., And Lorii, J. A., Operational Experiences With Westinghouse Cores, *WCAP-8183*, December, 1976.	11.1
Hellman, J. M., (Ed.), Fuel Densification Experimental Results and Model for Reactor Operation, *WCAP-8219*, October, 1973.	4.1, 4.2, 4.4
Lamb, V. F. and Capezzuto, F., Topical Report - Seismic Testing and Functional Verification of By-Pass Loop Reactor Coolant RTD's, *WCAP-8234-A*, June 1974.	3.10
Gesinski, L. T., "Safety Analysis of 17 x 17 Fuel Assembly for Combined Seismic and Loss of Coolant Accident", *WCAP-8236*.	4.2
Documentation of Selected Westinghouse Structural Analysis Computer Codes, *WCAP-8252*, Westinghouse Electric Corporation, March, 1974.	3.9
Source Term Data for Westinghouse Pressurized Water Reactors, *WCAP-8253*, May, 1974.	11.1
Lipchak, J. B., Nuclear Instrumentation System, Westinghouse Electric Corporation, *WCAP-8255*, January, 1974.	7.2, 7.7
Shepard, R. M., Massie, H. W., Mark, R. H. and Docherty, P. J., "Westinghouse Mass and Energy Release Data for Containment Design," *WCAP-8264-P-A* (Proprietary) and *WCAP-8312-A, Revision 1* June, 1975 (Non-Proprietary).	6.2
Nakazato, S., Demario, E. E., Hydraulic Flow Test of the 17 x 17 Fuel Assembly, *WCAP-8278*, February, 1974 (Westinghouse Proprietary) and *WCAP-8279*, February, 1974.	4.2

Topical Report	FSAR Section
Gesinski, L., Chiamg, D. Nakazato, S., Safety Analysis of the 17 x 17 Fuel Assmbly for Combined Seismic and Loss of Coolant Accident, *WCAP-8288*, December, 1973.	4.2
Hill, K. W., Motley, F. E., Cadek, F. F., Wenzel, A. H., Effect of 17 x 17 Fuel Assembly Geometry on DNB, *WCAP-8296-P-A*, February, 1975 (Proprietary) and *WCAP-8297-A*, February, 1975.	4.4
Motley, F. W., Wenzel, A. H., Cadek, F. F., The Effect of 17 x 17 Fuel Assembly Geometry on Interchannel Thermal Mixing, *WCAP-8298-P-A*, (Proprietary) and *WCAP-8299-A*, January, 1975.	4.4
Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests, *WCAP-8317-A*, July, 1975.	3.9
Entrietlo, J. F., Control of Delta Ferrite in Austenitic Stainless Steel Weldments, *WCAP-8324*, May, 1974.	5.2
Westinghouse Anticipated Transients Without Reactor Trip Analysis, *WCAP-8330*, August, 1974.	4.3, 15.8
Bordelon, F. M., and Zordan, T. A., Westinghouse Emergency Core Cooling System Evaluation Model-Summary, *WCAP-8339*, June, 1974.	6.2,
Grim, N. P., and Colenbrander, H. G. C., Long Term Ice Condenser Containment Code - LOTIC Code, *WCAP-8354-P-A* (W NES Proprietary) and *WCAP-8355-A*, April, 1976.	6.2
Hsieh, T. and Raymond, M., Long Term Ice Condenser Containment Code - LOTIC Code, *WCAP-8354-P*, Supplement 1 (Westinghouse Proprietary) and *WCAP-8355-A*, Supplement 1, April, 1976.	6.2,
Hellman, J. M., and Yang, J. W., Effects of Fuel Densification Power Spikes on Clad Thermal Transients, *WCAP-8359*, July, 1974.	4.3
Fischer, D. G., Qualification of Westinghouse Seismic Testing Procedure for Electrical Equipment Tested Prior to May, 1974, Westinghouse Electric Corporation, *WCAP-8373*, August, 1974.	3.10, 7.5
George, R. A., Lee, Y. C., Eng. G. H., Revised Clad Flattening Model, *WCAP-8377*, (Westinghouse Proprietary), and *WCAP-8381*, July, 1974.	4.2
Morita, T., et al., Topical Report, Power Distribution Control and Load Following Procedures, *WCAP-8385*, (Westinghouse Proprietary) *WCAP-8403* (Non-Proprietary), September, 1974.	4.4
[HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED] *Baldwin, M. S., Merrian, M. M., Schenkel, H. S. and Van De Walle, D. J., "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," WCAP-8424, Revision 1, May 1975.*	*15.3*
Deleted Per 2011 Update	

Topical Report	FSAR Section
Burke, T. M., Mey, C. E., and Shefcheck, J., Analysis of Data from the Zion (Unit 1) THINC Verification Test *WCAP-8453* (Westinghouse Proprietary) and *WCAP-8454* (Non-Proprietary), December, 1974.	4.4
Safety Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries, Fall, 1974, *WCAP-8485*, March, 1975.	4.2
UHI Plant Internals Vibration Measurement Program and Pre and Post Hot Functional Examinations, *WCAP-8517*, April, 1975.	3.9
Motley, F. E., Wenzel, A. H. and Cadek, F. F., "Critical Heat Flux Testing of 17 x 17 Fuel Assembly Geometry with 22 Inch Grid Spacing," *WCAP-8536* (Proprietary), May, 1975 and *WCAP-8537*, May, 1975.	4.4
Seismic Qualification of Reactor Trip Switchgear Assembly with Type DS-Model 416 Breakers, *WCAP-8587*, Supplement 2, ESE-20A.	3.10
"Augmented Startup and Cycle 1 Physics Program Supplement 1," *WCAP-8575* Supplement 1, June, 1976 (Westinghouse Proprietary) and *WCAP-8576*, June, 1976 (Non-Proprietary).	4.3
Jarecki, S. J. and Vogeding, E. L. Multifrequency and Direction Seismic Testing of Relays, *WCAP-8674*, December, 1975.	3.10
Dominick, I. E. and Orr, W. L., "Experimental Verification of Wet Fuel Storage Criticality Analyses," *WCAP-8682* (Proprietary) and *WCAP-8683*, December, 1975.	4.3
Equipment Qualification Test Report Barton Pressure Transmitters Group A lot 4 and 5 *WCAP-8687*, Supplement 2, E01A.	3.10
Equipment Qualification Test Report Barton Differential Pressure Transmitters Group A lot 4 and 5 *WCAP-8687*, Supplement 2, E03A.	3.10
Skaritka, J., Editor "Fuel Rod Bow Evaluation," *WCAP-8691*, Revision 1, July, 1979.	4.2, 4.4
Motley, F. E., Hill, K. W., Cadek, F. F. and Shefcheck J., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," *WCAP-8762*, July, 1976 (Proprietary) and *WCAP-8763*, July, 1976 (Non-Proprietary).	4.4
Eggleston, F. T., "Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries, Spring 1976," *WCAP-8768*, June, 1976.	4.3
Bloyd, C. H., Singleton, N. R., and Ciaramitaro, W., *Verification of Neutron Pad and 17 x 17 Guide Tube Design by Preoperational Tests on the Trojan 1 Power plant*, *WCAP-8780*, May, 1976.	3.9
WCAP-15063-P-A, Rev. 1, “Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)”, SER dated April 24, 2000.	4.2

Topical Report	FSAR Section
Jarecki, S. J. et. al., Seismic Operability Demonstration Testing of the Westinghouse 7300 Series Process Instrumentation System Bistables, *WCAP-8829*, November, 1976.	3.10
Jarecki, S. J. et al, Seismic Operability Demonstration Testing of the Nuclear Instrumentation Systems Bistable Amplifier, *WCAP-8831*, October, 1976.	3.10
Skaritka, J. (Ed.), Hybrid B_4C Absorber Control Rod Evaluation Report, *WCAP-8846-A*, October, 1977.	4.2
Siroky, R. M., and Marasco, F. W., “7300 Series Process Control System Noise Tests,” *WCAP-8892-A*, June, 1977 (Non-Proprietary).	7.2
Risker, D. et. al., “Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis,” *WCAP-8963* (Proprietary), November, 1976, and *WCAP-8964*, August, 1977.	4.2
Bitting, R. A., Seismic Testing of Veritrak Model 59 Transmitters, *WCAP-8965*, April, 1977.	3.10
Moore, J. S., "Nuclear Design of Westinghouse Pressurized Water Reactors with Burnable Poison Rods," *WCAP-9000-L*, Revision 1 (Proprietary), July, 1969 and *WCAP-7806*, December, 1971.	4.3
Carter, F. D., Inlet Orificing of Open PWR Cores, *WCAP-9004*, January, 1969, (Westinghouse Proprietary), and *WCAP-7836*, January, 1972.	4.4
Ellis, A. E. and Miller, R. B.; "Environmental Qualification of Safety Related Class 1E Process Instrumentation," *WCAP-9157* August, 1977.	3.11
Graham, K. F., and Forker, H. M., "A N-16 Transit Time Flow Measurement System (TTFM) Description and Performance," *WCAP-9172* (Proprietary), October 1977.	4.4
Beaumont, M. D., et. al., (Ed.), Properties of Fuel and Core Component Materials, *WCAP-9179*, Revision 1 (Proprietary) and *WCAP-9224*, July, 1978.	4.2
Davidson, S. L., Motley, F. E., Lee, Y. C., Bogard, T., Bryan, W. J., "Verification Testing and Analyses of the 17 x 17 Optimized Fuel Assembly," *WCAP-9401*, March, 1979 (Proprietary) and *WCAP-9402*, March, 1979.	4.2, 4.4
Camden, T. M. et. al., "PALADON - Westinghouse Nodal Computer Code," *WCAP-9485* (Proprietary) and *WCAP-9486* (December, 1978).	4.3
Davidson, S. L., Iorii, J. A., "Reference Core Report, 17 x 17 Optimized Fuel Assembly", *WCAP-9500-A*, May, 1982.	4.1, 4.2, 4.3
Miller, R. B., Qualification Testing of ITT/Barton Transmitters Production Lot No. 2, *WCAP-9885*, April, 1981.	3.10, 3.11
(Sequoyah Preliminary Results Letter), *Verification of Upper Head Injection Reactor Vessel Internals by Preoperational Test on the Sequoyah 1 Power Plant*, *WCAP-9945*, March, 1981.	3.9

Topical Report	FSAR Section
Skaritka, J., "Westinghouse Wet Annular Burnable Absorber Evaluation Report," *WCAP-10021*, Revision 1, October, 1982.	4.3
Harris, A. J., et. al., "A Description of the Nuclear Design and Analysis Programs for Boiling Water Reactors," *WCAP-10106*, June, 1982.	4.3
"Westinghouse LOCA Mass and Energy Release Model for Containment Design" - March, 1979 Version, *WCAP-10325*, April, 1979.	6.2
WCAP-10444-P-A, Westinghouse Reference Core Report Vantage 5 Fuel Assembly, September, 1985.	4.2
S. E. Yanichko, T. V. Congedo, W. T. Kaiser, Analysis of Capsule U from the Duke Power Company McGuire Unit 1 Reactor Vessel Radiation Surveillance Program, *WCAP-10786*, February, 1985.	5.2
Liu, V. S., et. al., "ANC: A Westinghouse Advanced Nodal Computer Code," *WCAP-10965-P-A*, December, 1985.	4.3
S. E. Yanichko, T. V. Congedo, W. T. Kaiser, Analysis of Capsule V from Duke Power Company McGuire Unit 2 Reactor Vessel Radiation Surveillance Program, *WCAP-11029*, January, 1986.	5.2
Skaritka, J. (Ed.), "Fuel Rod Bow Evaluation", WCAP-8691, Rev. 1, July 1979.	4.2
Skaritka, J.,"Wet Annular Burnable Absorber Evaluation Report", WCAP-10021-P-A, Rev. 1, October 1983.	4.2
S.L. Davidson, "VANTAGE 5 Fuel Assembly Reference Core Report", WCAP-10444-P-A, September 1985.	4.2, 4.4
S.L. Davidson and T.L. Ryan, "VANTAGE + Fuel Assembly Reference Core Report", WCAP-12610-P-A, April 1995.	4.2, 4.4
L.D. Smith, et. al.,"Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids, WCAP-15025P-A, April 1999.	4.4
"Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P-A (Proprietary), Volume I, Revision 2, and Volumes II-V, Revision 1, and WCAP-14747 (Non-Proprietary), Westinghouse Electric Company, March 1998	15.6
"Best Estimate Analysis of the Large Break Loss of Coolant Accident for the McGuire and Catawba Nuclear Stations," WCAP-15440, Rev. 0, Westinghouse Electric Company LLC, July, 2000.	15.6
Meyer, P.E., "NOTRUMP, "A Nodal Transient Small Break and General Network Code", WCAP-10079-P-A (Proprietary), August 1985.	15.6
Rupprecht, S. D., et al., "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," WCAP-11145-P-A (Proprietary) and WCAP-11372-A (Non-Proprietary), October 1986.	15.6

Topical Report	FSAR Section
Lee, H., Tauche, W. D., Schwarz, W. R., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", WCAP-10054-P-A (Proprietary), August 1985.	15.6
Bordelon, F.M. et.al., "LOCTA-IV Program: Loss of Coolant Transient Analysis," WCAP-8301 (Proprietary) and WCAP-8305 (Non-Proprietary) June 1974.	15.6
Thompson, C. M., et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," WCAP-10054-P-A (Proprietary), Addendum 2, Revision 1, July 1997	15.6
Shimeck, D. J., "1-D Heat Conduction Model for Annular Fuel Pellets," WCAP-14710-P-A (Proprietary), WCAP-14711-NP-A (Non-Proprietary), May 1998.	15.6
Akers, J. J. ed., "Model Changes to the Westinghouse Appendix K Small Break LOCA NOTRUMP Evaluation Model: 1988 – 1997," WCAP-15085, July 1998.	15.6

1.6.2 Babcock & Wilcox Reports

Report	FSAR Section
Deleted Per 2006 Update	
Deleted Per 2006 Update	
Deleted Per 2006 Update	
Deleted Per 2011 Update	
Deleted Per 2006 Update	
Deleted Per 2006 Update	
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Deleted Per 2006 Update	
Deleted Per 2006 Update	
Deleted Per 2006 Update	
Deleted Per 2006 Update	
Deleted Per 2006 Update	
Deleted Per 2006 Update	
Deleted Per 2006 Update	

1.6.3 Duke Reports

Report	FSAR Section
1. McGuire Electrical Schematics, Revision 3, January, 1975.	7.0
2. Quality Assurance Program *DUKE-1*, April, 1974.	17.0
3. McGuire Nuclear Station Emergency Plan, Rev. 8, November, November, 1982.	13.3
4. “An Analysis of Hydrogen Control Measures at McGuire Nuclear Station,” Revision 9, October 20, 1983.	6.2
5. “Response To NUREG-0588,” Environmental Qualification of Class 1E Equipment Through Rev. 2.	3.11
6. “McGuire Nuclear Station Inservice Inspection Plan,” Revision 6, March 6, 1984.	5.2
7. McGuire Nuclear Station Security Plan	13.7
8. McGuire Nuclear Station Environmental Impact Report	2.0
9. Deleted Per 2006 Update	
10. Duke Power Company, "McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01," *DPC-NE-2004P-A*, December, 1991.	4.3, 4.4
11. "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology," *DPC-NE-2005-PA*, December, 2008.	15.0, 15.1, 15.2, 15.3, 15.4, 15.6
12. Duke Power Company, "McGuire Nuclear Station, Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," *DPC-NF-2010PA*, June, 1985.	4.3
13. Duke Power Company, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," *DPC-NE-2011-PA*, June, 2009.	4.3, 15.4
14. Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology," *DPC-NE-3001-PA*, May, 2009.	4.3, 15.1, 15.4
15. "FSAR Chapter 15, System Transient Analysis Methodology," *DPC-NE-3002-A*, Duke Power Company, Revision 4b, SER dated April 6, 2001.	4.3
16. "McGuire Nuclear Station, Catawba Nuclear Station, Startup Physics Test Program," Duke Power Company, April, 1988.	4.3
17. "Catawba/McGuire Nuclear Design Guidelines," December 1, 1987.	4.3
18. Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, "Mass and Energy Release and Containment Response Methodology." DPC-NE-3004-PA, Rev. 1, Duke Power Co., SER dated February 29. 2000	6.2

Report	FSAR Section
19. 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, April 26, 1996.	4.3
20. Deleted Per 2006 Update	
21. DPC-NE-2004P-A, Revision 1, McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, Duke Power Company, February 1997.	4.4
22. DPC-NE-2005P-A, Rev. 1, Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, November 1996.	4.4
23. DPC-NE-2007P-A, Duke Power Company Fuel Reconstitution Analysis Methodology, Duke Power Company, October 1995.	4.2
24. DPC-NE-2009-P-A, Rev. 2"Duke Power Company Westinghouse Fuel Transition Report," Duke Power Company, SER dated Dec. 18, 2002.	4.2, 4.3, 4.4, 15.1
25. DPC-NE-1005-P-A, Duke Power Company, "Nuclear Design Methodology using CASMO-4/SIMULATE-3 MOX", SER dated August 20, 2004.	4.3
26. Deleted Per 2018 Update	
27. McGuire Nuclear Station Cyber Security Plan	13.7.2

1.6.4 Other Reports

Report	FSAR Section
1. Application of the Active Ice Management Concept to the Ice Condenser Ice Mass Technical Specification, *ICUG-001*. Ice Condenser Utility Group, Revision 2, June 2003.	6.2

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1.7 Division 1 Regulatory Guides

The purpose of NRC Regulatory Guides are (1) to describe methods acceptable to the NRC Regulatory Staff of implementing parts of the Commission's Regulations and (2) to provide guidance to applicants concerning information needed by the Regulatory Staff in its review of applications for permits and licenses. Because these guides are not intended as substitutes for regulations, compliance with these guides is not required.

Table 1-4 summarizes Duke Energy's position on each UFSAR applicable Division 1 Regulatory Guide and lists the sections which discuss the subject matter of each guide.

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1.8 Response to TMI Concerns

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

1.8.2 Shift Technical Advisor (I.A.1.1)

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

1.8.3 Operations Shift Manager Administrative Duties (I.A.1.2)

See Section 1.8.11.

1.8.4 Shift Manning (I.A.1.3)

The shift crew composition for operation of McGuire Units 1 and 2 will be in accordance with Section 5.0 of the Technical Specifications.

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

See Section 13.2.

The 3/28/1980 NRC letter regarding qualifications of reactor operators, as referenced in Technical Specification 5.3 (Unit Staff Qualifications), is clarified by NUREG-0737 (Clarification for TMI Actions), Regulatory Guide 1.8 Revision 2 (Qualification and Training of Personnel for Nuclear Power Plants), NUREG-1021 (Operator Licensing Examination Standards for Power Reactors), and Information Notice 98-37 (Eligibility of Operator License Applicants).

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

See Section 13.2.

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

See Section 13.2.

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

See Sections 13.1 and 13.4.

1.8.9 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 reviewed by the NRC. Duke has reviewed these reports and made the necessary modifications to the McGuire emergency procedures and training program. Revisions have been made, as necessary, due to the removal of UHI.

Duke Energy has provided a response to supplement 1 to NUREG-0737, which included a description of the Emergency Procedure Upgrading Program for McGuire. Section 6, “Emergency Procedure Upgrade Program” of that response described Duke's plans for incorporating the Westinghouse Emergency Response Guidelines (ERG'S) into the McGuire emergency procedures.

On June 1, 1983 the NRC Staff's Safety Evaluation of Emergency Response Guidelines was transmitted to Mr. J. J. Sheppard, Chairman, Westinghouse Owners Group by Darrell G. Eisenhut, NRC/ONRR. The guidelines set listed in Table 2 of the SER served as the starting point for the development of the Catawba and McGuire Nuclear Stations' plant specific technical guidelines. Duke's review of further modifications to the guidelines, made by Westinghouse as they developed Revision 1 to the ERG set, resulted in changes to this guideline set. By letters dated June 18, 1984, July 25, 1984, August 29, 1984, October 17, 1984, April 30, 1985, and October 29, 1985, from H. B. Tucker to H. R. Denton, Duke Energy provided descriptions of deviations from the Westinghouse Owners Group generic guidelines, including deviations from the basic version and Revision 1 of the guidelines. These deviations included, but were not limited to, the safety-significant deviations between the generic guidelines and the Catawba-Specific Technical Guidelines. In Catawba SER Supplements 4 and 6 the NRC approved these deviations. As these approvals were received, applicable changes were made to McGuire procedures.

By letters dated September 28, 1992 and November 18, 1992, Duke advised the NRC that the McGuire Emergency Operating Procedures (EOP's) were being upgraded to revision 1B of the Westinghouse Owners Group Emergency Response Guidelines (ERG's). The November 18, 1992 letter further detailed changes to the EOP development program in that the Duke Energy specific Emergency Procedure Guidelines (EPG's) would be eliminated when the new set of EOP's were issued.

By letter dated May 26, 1994, Duke notifited the NRC that deviation documents were being developed to address differences between the generic guidelines and the plant specific emergency procedures.

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

An administrative procedure has been revised to incorporate a detailed checklist of applicable items for shift turnover. In addition, periodic test procedures, Daily and Monthly Surveillance Items, have been written to ensure an adequate evaluation of shift turnovers.

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

Nuclear Generation issues a corporate management directive that clearly establishes the command duties of the Control Room Senior Reactor Operator and emphasizes the Control Room Senior Reactor Operator primary management responsibility for safe operation of the plant. This directive is reissued annually.

The Control Room Senior Reactor Operator 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

Control Room Senior Reactor Operator and control room operators are defined in an administrative procedure.

1.8.12 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 McGuire Security Plan controls access to all vital areas of the plant including the control area.

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

The operating experience program is described in AD-PI-ALL-0400, AD-PI-ALL-0401 and AD-PI-ALL-0402.

1.8.14 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 Control Room Senior Reactor Operator 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 Control Room Senior Reactor Operator 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 plant computer 1.47 bypass application. 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.

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

See Sections 13.5 and 14.1.

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

Station emergency procedures have been made available for review by the NRC as described in Section 13.5.

1.8.17 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.

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

Training performed during low power testing was described in a letter from W. O. Parker to H.R. Denton dated 9/8/80.

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

Duke has installed a reactor vessel head high point vent that is remotely operable from the McGuire control room. A one-inch line has been added to the existing reactor vessel manual vent line with the connection located before the first isolation valve. The new vent line contains two parallel flow paths with redundant fail closed solenoid valves in each flow path. The valves have been designed to pass non-condensible gases, water, steam, and mixtures thereof. Under normal operation these valves are deenergized. Valve position is indicated in the control room. Train A emergency power serves both isolation valves in one flow path, and Train B emergency power serves both isolation valves in the parallel flow path. A flow limiting orifice has been installed in the common line downstream of the isolation valves.

The McGuire Reactor Coolant System (RCS) vent system is safety grade, seismically qualified, meets the requirements of IEEE 279-1971, and satisfies the single failure criteria. This vent system is operable.

As mentioned above, this head vent system has been designed to single failure criteria. If any single failure prevents a venting operation through one flow path, the second flow path is available for venting the RCS. The two isolation valves in each flow path provide a single failure method of isolating RCS venting.

Inadvertent actuation of RCS venting is limited by the use of fail closed solenoid isolation valves. In addition, the use of an orifice in the common line downstream of the valves limits the flow to less than the make up capacity of the RCS charging pumps.

Exhaust from the vent system is directed to the Pressurizer Relief Tank (PRT) and therefore will not impinge upon vital equipment. A path is available to allow venting hydrogen from the reactor vessel head to the Waste Gas System storage tanks, via the pressurizer relief tank, should such an option be judged desirable. The PRT is located in the lower containment which is ventilated and cooled by four air handling units. In addition, the hydrogen skimmer system has ducts in the lower containment high points to disperse any accumulated hydrogen.

Assuming that 100% hydrogen is being vented from the reactor vessel head, the vent system flow rate is 14 cfm. This allows the venting of the gas volume in the reactor head in approximately 1 hour.

The power-operated relief valves (PORV) are used to vent the RCS pressurizer. The PORV's are discussed in the McGuire FSAR Section 5.2.2. The RCS vent is located at the top of the reactor vessel head which is the high point of the reactor vessel and coolant loops. This system in conjunction with the PORV's provides a venting capability for the entire RCS with the exception of the U-tube steam generators.

A postulated break of the reactor vessel head vent line upstream of the flow limiting orifice would result in a small LOCA of not greater than one inch diameter. Such a break is similar to those analyzed in WCAP-9600 for hot leg breaks or pressurizer vapor space breaks. Extensive discussions were provided in WCAP-9600 regarding the applicability of the break flow models employed as well as other specific modeling features employed for small LOCA analysis. System response for this postulated break location would closely parallel that described in Section 3.2 of WCAP-9600. Since the break location in the head vent line would behave similarly to the hot leg break case presented as Case F of that section, the discussion presented in WCAP-9600 for that study applies to this postulated break. As such, this postulated break would result in no calculated core uncovery.

A flow diagram of the McGuire RCS including the RCS vent system is provided in the McGuire FSAR Figure 5-1.

1.8.20 Plant Shielding (II.B.2)

1. See Section 12.1.6.3.

1.8.21 Post-accident Sampling (II.B.3)

Deleted paragraphs per 2002 revision.

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. Post-accident dose assessment for sampling indicates that within appropriate time constraints, personnel exposures would be well below GDC-19 criteria.

1.8.22 Training for Mitigating Core Damage (II.B.4)

Duke has modified the training program in order to place increased emphasis on the operation and significance of any 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.

1.8.23 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 valves covered by the EPRI program are pressurizer safety valves, pressurizer PORV, and PORV block valves.

Plant specific information on the McGuire pressurizer safety valves, PORV's and PORV block valves was transmitted by H.B. Tucker's letter of June 30, 1982 to H. R. Denton.

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

PORV

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.

Safety Valve

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.

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

The Auxiliary Feedwater System is described in Section 10.4.10. A reliability analysis was submitted on August 13, 1980.

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-9751) based on the method described in Enclosure 1 of the March 10, 1980 letter has been performed and was submitted by an August 13, 1980 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 McGuire Auxiliary Feedwater System, compared to the reliabilities as defined and reported for Westinghouse plants in NUREG-0611, is medium for a loss of main feedwater transient, low for a loss of main feedwater coupled with a loss of offsite power transient, and medium 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 testing, unscheduled maintenance of components and component hardware failures.

1.8.25 Short-Term Recommendations

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

Automatic Initiation

Safety-grade automatic initiation and safety-grade emergency power for the Auxiliary Feedwater System (AFS) are features of the McGuire Nuclear Station design (reference FSAR Chapter 7 and Chapter 10).

The automatic initiation circuitry for the AFS meets the single failure criteria. Additionally, for most failures which could prevent the automatic start of an individual auxiliary feedwater pump, manual initiation of the affected pump is available from the control room. However, should the auxiliary feedwater pump in one safety train not be available due to any single failure, the redundant safety train is available with no loss of system function.

The actual ESF automatic initiating signal for the autostart of the CA motor driven pumps on the trip of both main feedwater pumps is NOT designed as safety-related at MNS and is not single failure proof. Additionally, the AMSAC initiating circuitry, which also provides an automatic start signal to the CAMD Pump, is NOT designed as safety-related. Both the ESF initiating signal on Loss of Both Main Feedwater Pumps and the AMSAC initiating signal, which are inputs to the CAMD PUMP auto-start circuitry, are designed to be anticipatory of a loss of feedwater condition. Their associated process sensors and relays are non-safety, and their functions are not relied upon in the UFSAR Chapter 15 analysis. The AMSAC initiating signal is credited in the ATWS analysis. Since these electrical inputs are non-safety, their wiring is separated from the safety related autostart control circuit via relay contacts in the safety-related auxiliary feedwater auto-start control circuit. This ensures that a single failure of the non-safety circuits will not disable the safety-related automatic start circuitry of the CAMD Pumps.

In the final stages of plant shutdown, the main feedwater pumps must be tripped. Therefore, the automatic start of the motor-driven auxiliary feedwater pumps upon trip of both main feedwater pumps or steam generator low-low level must be bypassed. This bypass is accomplished manually by means of a bypass switch located in the control room. When the bypass is instated a light is energized on the bypass control switch. Additionally, status light indication of the bypass is provided on the associated status light panel. This bypass is administratively controlled by use of operating procedures.

When the plant is in the startup mode, station procedures require that the bypass of the above motor-driven auxiliary feedwater pump start signals be removed. In addition to the station procedures, an automatic means to remove the bypass has been provided. This automatic bypass removal will be generated when the P-11 set point is reached. The P-11 set point is derived from Reactor Coolant System (RCS) pressure (~ 1955 psig) and is the same signal used to unblock safety injection actuation. The P-11 set point is considered the appropriate signal to automatically remove the bypass of the above motor-driven auxiliary feedwater pump start signals because the reactor is not brought critical until RCS operating temperature and pressure conditions have been reached.

In the unlikely event that the following plant conditions exist: (1) the plant is in the startup mode and below ~ 1955 psig (P-11 set point), (2) the turbine-driven auxiliary feedwater pump train is not available, (3) the bypass of the motor-driven auxiliary feedwater pump start signals has not been removed, and (4) either both main feedwater pumps trip on a low-low steam generator signal is generated due to a steam line rupture or feedwater line break; adequate time is available for the reactor operator to manually initiate the auxiliary feedwater system. The low-low steam generator level alarm would alert the operator to the need to check the auxiliary feedwater pumps and assure they are running. The operator would have at least ten minutes upon receipt of this alarm to take this action. This is a conservative estimate of the time available for operator action based upon a comparison of the steam/feed line break at full power with a similar accident under startup conditions; i.e., subcritical pressure less than 1955 psig, decay heat at much lower levels. The turbine-driven auxiliary feedwater pump does not have a bypass feature.

<u>Indication</u>

Safety grade indication of auxiliary feedwater flow to each steam generator has been provided in the McGuire control room. Provisions for calibration and testing were incorporated into the design of this instrumentation.

Control grade flow instrumentation in the lines of each steam generator and in the suction piping to each auxiliary feedwater pump is also provided. This control grade flow instrumentation is provided from the highly reliable battery-backed 120 VAC Auxiliary Control Power System (reference FSAR Chapter 8). Provisions for calibration and testing are included in the design of this control grade flow instrumentation.

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

For each McGuire unit, two groups of pressurizer heaters with a nominal, initial capacity of 416 Kw each are supplied from the redundant 600 VAC Essential 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 automatically shed from the emergency power sources via the diesel generator load sequencer upon the occurrence of a safety injection actuation signal (SIAS). 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 for manually loading the pressurizer heaters onto the emergency power sources following an SIAS are available to the operator.

The Class 1E interfaces for main power and control power are protected by safety-grade circuit breakers.

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, connection, 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 McGuire, 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.

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

See Section 6.2.5.

1.8.25.4 Containment Isolation Dependability (II.E.4.2)

Containment isolation at McGuire is derived from any of the following diverse parameters: (Figure 7-1)

1. low steamline pressure
2. low-low pressurizer pressure
3. high containment pressure
4. manual

Containment isolation valve control systems are designed such that resetting of the containment isolation signal will not cause any containment isolation valve to automatically reposition. Deliberate operator action is required to reposition a containment isolation valve following reset of a containment isolation signal.

Tables 6-111 through 6-114 list the containment piping penetrations, their functions, and other pertinent information. The McGuire Containment Isolation System design has been reviewed with careful consideration given to the definition of essential and non-essential systems. This review resulted in no change in the list of systems considered essential. However, the following modifications to the Containment Isolation System have been made.

1. Previously the Containment Ventilation Unit Drain Header containment isolation valves (see Table 6-113) closed upon receipt of the high-high containment pressure signal (Phase B Isolation Signal). This has been changed to signal the valves to close on a Phase A Isolation Signal. The purpose of this line is only to prevent draining ventilation unit condensate to the containment floor sumps during normal operation and then having to process it through the waste system. It does not serve to protect the ventilation units. Therefore, since containment isolation takes priority over waste processing concerns, it is necessary to isolate this line as early as possible.
2. Two 50 gpm sump pumps are provided in each of two Containment Floor and Equipment Sumps and one 50 gpm pump is provided in the Incore Instrumentation Sump. These pumps started automatically on high sump level signals. They discharge to the 10,000 gallon Floor Drain Tank in the Auxiliary Building. The automatic start feature for these pumps has been replaced with manual start capability only. The automatic stop on low sump level has been retained. Sump level indication and high level alarm has been provided to the operator in the control room for the Containment floor and equipment sumps. A high level annunciator alarm and a hi-hi level computer alarm are provided in the control room for the incore instrument room sump. For pump protection, an interlock has been provided to trip the pumps or prevent them from starting if one of the containment isolation valves in the sump discharge header closes. The pumps also trip on a high containment radioactivity signal.

The majority of containment fluid line penetrations are closed upon receipt of the Phase A containment isolation signal which is initiated by one of the following parameters:

1. low steamline pressure (2/3)
2. low-low pressurizer pressure (2/4)
3. high containment pressure (1 psig) (2/3)
4. manual

The remaining penetrations are isolated upon receipt of the Phase B containment isolation signal which is initiated by the following parameters:

1. Containment high-high pressure signal (3 psig) (2/4)
2. Manual Phase B actuation signal.

1. Reactor Coolant Pump Motor Cooling Water Supplies

   This service is maintained until the Phase B isolation signal is received to prevent possible motor damage following events such as main steam break outside containment, or small isolable LOCA's, which may not result in containment pressure reaching the high-high (3 psig) setpoint, and for which pump operation may become desirable or even necessary.

2. Reactor Coolant Pump Thermal Barrier Cooling Water Supply

   The thermal barriers act as a backup to the seal water injection flow which cools the main seals of the reactor coolant pumps. If seal flow is interrupted for any reason, the thermal barrier, which is located further down on the pump shaft (between the seals and the impeller), cools the reactor coolant which will flow up through the seals, thus preventing possible seal damage. Because of the sensitive nature of the seals it is desirable to maintain thermal barrier cooling flow until the Phase B isolation is received in order to guarantee pump availability for events such as those cited in 1 above.

3. Containment Ventilation Unit Cooling Water Supply

   The service is maintained until the Phase B signal is received to minimize containment pressure rise for events, such as small isolable LOCA's, main steam breaks outside containment, and blackout. Reference Section 6.2.1.1.3.1 for a more detailed description.

4. Main Steam Isolation Valve Closure

   Closure of the main steam isolation valves results in loss of normal heat sink, loss of normal pressure control, and actuation of main steam safety valves, with associated system transients. In order to minimize unnecessary closures, these valves are closed upon receipt of the Phase B isolation signal.

5. Instrument Air Supply to Containment

   Isolation of instrument air to Containment causes loss of control air to the pressurizer power operated relief valves, spray valves, containment ventilation unit cooling water control valves, and various other valve controllers. It is desirable to maintain operability of these valves following many postulated events, such as those cited in 1. above which result in a Phase A containment isolation, but which may not result in containment pressure increasing to the high-high (3 psig) Phase B isolation set point.

The McGuire containment purge and vent isolation valves are closed automatically upon receipt of a high radiation signal as indicated in Section 6.2.4.1. However, these valves are sealed closed during Modes 1-4.

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

Noble Gas Monitors

Vent monitors for noble gases are provided with a range adequate to cover both normal and postulated accident conditions. The previously installed noble gas monitors at McGuire covered the range of 10^{-7} µCi/cc to 10^{+3} µCi/cc. A gross gamma detector was added to these monitors to extend the range up to 10^{+5} µCi/cc. This detector is inserted through the wall of the unit vent. The detector is sensitive to the 80 Kev energy range noble gases and has a minimum of one decade overlap with the existing noble gas monitor. If an event were to occur to cause the activity being released to be in the range of this additional detector, the noble gas monitor sample will be isolated. This action will prevent the noble gas monitor from becoming contaminated and rendering erroneous indications when activity released starts decreasing.

The present radiation monitoring system provides detection of volatile and non-volatile radioactive contamination of secondary systems. A condenser air ejector monitor continuously monitors gaseous activity released to the unit vent by the condenser air ejector exhaust. A steam generator sample monitor continuously monitors non-volatile activity in all steam generators. An alarm on either of these monitors provides control room operators with an indication of steam generator tube failure. By valving steam generator samples individually through the steam generator sample monitor, chemistry personnel can identify and if desired isolate the affected steam generator. The condenser air ejector monitor would quantify the level of radioactivity released to the environment prior to isolation of the affected steam generator.

To quantify the level of radioactivity released in the event the affected steam generator is not isolated, Duke uses a steam radiation monitoring system. This system uses four area radiation monitors (GA Model RD-1A) mounted in the doghouse, one near each of the four main steam lines. Continuous display of the monitor readout is provided in the control room. Indication is also provided on the plant computer (OAC). The radioactivity range covered is 10^{-1} to 10^{4} mR/hr and correlated with design-basis reactor coolant inventory spectra to isotopic concentrations.

Deleted paragraph(s) per 2003 update.

The containment hydrogen purge exhaust discharge through the unit vent and is monitored by the unit vent radiation monitors.

Containment Pressure

Wide range and narrow range continuous indication of containment pressure has been provided in the control room. The wide range measurement and indication extends from -5 to 60 psig, the narrow range instrumentation extends from -5 to 20 psig. Each of the redundant differential pressure transmitters is located in an electrical penetation room is equipped with one-half inch tubing impulse lines. Each impulse line has a fail-closed 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 is provided in the control room. In addition, one channel of both wide and narrow range containment pressure is recorded.

Containment Water Level

Two containment floor and equipment sumps are provided on the floor of the lower containment (El 725') 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 McGuire encompasses the entire floor of the lower containment. The two ECCS recirculation lines take suction just inside the Containment wall at elevation 725' 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 0-20 feet (El 725' to 745') which is equivalent to a lower containment volume of approximately 1,000,000 gallons.

The redundant float level transmitters utilized in this instrumentation are located inside containment. Continuous indication from each transmitter is provided in the control room. In addition, one channel of containment water level is recorded.

Containment Hydrogen Monitoring

Continuous indication of hydrogen concentration in the containment atmosphere is provided in the control room after appropriate solenoid valves are energized as directed by appropriate procedures. The containment hydrogen monitoring system will be available within 12 hours of a LOCA inside containment. This hydrogen monitoring system consists of two redundant analyzer systems with a dual range of 0 to 10%/0 to 30% hydrogen by volume. These analyzers 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.

The hydrogen monitors are no longer required to support mitigation of design-basis accidents as the revised 10 CFR 50.44 no longer defines a design-basis LOCA hydrogen release. The hydrogen monitors are therefore no longer required to be safety-related as they no longer meet the definition of a safety-related component as defined in 10 CFR 50.2. The hydrogen monitors are used to diagnose the course of beyond design-basis accidents.

Each containment sample header has three inlet samples available for monitoring.

1. Top of containment
2. Operating level
3. Steam Generator 1B Cavity

All sample selection and switching is accomplished manually by the operator from the local analyzer control panel.

Containment High Range Radiation Monitoring

Two physically and electrically separated radiation monitors have been installed inside the McGuire containment. 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 covers the range from 10^0 to 10^8 R/hr. Additional correlation curves are provided to estimate post-accident isotopic inventories. No overlapping of ranges is required. Monitor sensitivity to 62 Kev is $9.8X10^{-12}$ Amps/Rad/hr and the sensitivity to 52 Kev is $9.0X10^{-12}$ Amps/Rad/hr. Seismic qualification of the monitor is in accordance with IEEE 344-1975 and environmental qualification meets IEEE 323-1971.

One monitor is powered from the Train A vital instrument bus, and the other monitor is powered from the Train B vital instrument bus. Analog meters (one per train) continuously indicate monitor output in the control room. A continuous strip chart recorder (one train) 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.

The monitors are mounted on the primary shield wall at an elevation of at least 750+2 (10 feet or more above the maximum post-LOCA water level at 200° and 335° in the lower containment).

1.8.27 Inadequate Core Cooling Instruments (II.F.2)

Subcooling Monitor

The margin to saturation is calculated from Reactor Coolant System (RCS) pressure and temperature measurements. The information is calculated and displayed by the Plant Computer and the Subcooling Margin Monitor (SMM) which is part of the Inadequate Core Cooling

Monitoring (ICCM) System. For the Plant Computer 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 plant computer output consists of a 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. Program logic variables such as source for pressure value (wide or low range sensor) and containment conditions (normal (<3 psig) or degraded (>3 psig)) are also provided. Alarm status is indicated by messages on the graphic display and the Alarm work station. 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.

There are two Class 1E channels of SMM which are provided by the ICCM (one SMM channel per train of ICCM). For each channel of the SMM, the average of the five highest value incore thermocouples for that channel (20 Class 1E T/C's are installed per channel) are used to represent core exit conditions. Each channel also uses wide range hot leg RTD's to measure the hot leg temperatures for two of the RCS loops.

The SMM performs calculations and a comparison to adjusted saturation curves (adjusted for possible measurement uncertainties) to compute margins and initiate alarms if appropriate.

The SMM output consists of alphanumeric/graphic plasma flat panel displays (one per channel) located on the main control boards in the Control Room.

The Class 1E instrumentation associated with the SMM channels is environmentally qualified in accordance with UFSAR Section 3.11 and the Duke Energy NUREG-0588 submittal. The qualification applies from the sensor (qualification assumed per II.F.2 guidance) to the final display device. For the plasma displays, qualification applies from the sensor to the display. ICCM inputs to the plant computer display are through qualified Class 1E isolation amplifiers.

The Class 1E instrumentation associated with the SMM is seismically qualified in accordance with UFSAR Section 3.10. This instrumentation would operate with the required accuracy after, but not necessarily during, a safe shutdown earthquake.

The availability of the Class 1E SMM primary displays is addressed in the Technical Specifications.

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 and SMM calculations.

Reactor Vessel Level Measurement

The reactor vessel level instrumentation system (RVLIS) is of Westinghouse design. The RVLIS is of a Westinghouse UHI design 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 and from the hot legs to the top of the reactor vessel. Under naturalcirculation 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 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 impulse 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, impulse-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 Manual, October 1983”.

Core Exit Thermocouple System

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-1E 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 1E T/C's are cabled directly from the T/C penetrations to the alphanumeric/graphic plasma flat panel primary displays located on the main control boards in the Control Room. Inputs to the backup plant computer displays is through qualified Class 1E isolation amplifiers.

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 enthalpy rise across the core. This is available to the operator via the plasma displays in the control room.

    b. The plasma displays are the operator's primary display having the following capabilities:

        1) A spatially oriented core map is available on demand indicating temperature and enthalpy rise at each core exit thermocouple location.

        2) The core exit thermocouples are an input into the saturation monitor program to assist operator actions for inadequate core cooling procedures.

        3) Direct readout capability is provided for all safety related thermocouple temperatures. This readout range extends from 32°F to 2300°F.

        4) Trending of selected thermocouples to show temperature - time history is available on demand.

        5) Alarm capabilities are provided through the saturation monitor program.

        6) Addressed in the Control Room Design Review.

c. Backup display is provided in the control room via the plant computer for any of the thermocouples. Using the plant computer, readings can be taken well within the six minute time guidance. The range of this backup display extends from 32°F to 2300°F.

d. See Item 1b6 above.

e. The following consists of an evaluation of the McGuire Nuclear Station Core Exit Thermocouple System compliance with 10CFR50, Appendix B to Item II.F.2. The paragraph numbers relate directly to Appendix B paragraph numbering.

1) The Class 1E instrumentation is environmentally qualified in accordance with UFSAR Section 3.11 and the Duke Energy NUREG-0588 submittal. The qualification applies from the sensor (qualification assumed per II.F.2 guidance) to the final display device. For the primary plasma displays, qualification applies from the sensor to the display.

The Class 1E instrumentation is seismically qualified in accordance with UFSAR 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 backup display and associated hardware beyond the isolator. The isolation device would be accessible for maintenance during accident conditions.

2) No single failure within the core exit 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 core exit thermocouple system has two reliable portions: (a) the primary Class 1E portion which has battery-backed power sources, (b) the nonsafety backup display portion which has a separate battery-backed power source.

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 8.3.2.1.4 up to and including the isolation device. Direct recording and trending capabilities are provided for any of the 40 safety related core exit thermocouple channels.

3) The Class 1E Core Exit Thermocouple Instrumentation and primary displays are powered by Class 1E power sources.

The backup core exit displays are powered by battery-backed power sources.

4) Incore instrumentation channel availability is addressed in the Technical Specifications.

5) The provisions of Duke Energy's Quality Assurance Program as described in UFSAR Chapter 17 and Topical Report Duke 1A have been applied to the Class 1E portion of the Core Exit Thermocouple System. The backup 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 6.

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 McGuire incore design facilitates the administrative control of the access to setpoint adjustments, calibration adjustments, and test points.

14) The McGuire design minimizes the existence of conditions which could lead to anomalous indications and confusion of the operator.

15) The McGuire design facilitates 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 Regulatory Guide 1.118.

f. The primary and backup displays are energized from independent battery-backed power sources. The primary displays 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 Section 8.3.2.1.4 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 Section 7.1.2.2 and Section 8.3.2.1.4 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 primary displays will be addressed in the Technical Specifications.

i. The provisions of Duke Power Company's Quality Assurance Program as described in FSAR Chapter 17 and Topical Report Duke 1A was applied to the Class 1E portion of the Core Exit Thermocouple System. The backup 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.

1.8.28 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 Chapter 8). 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 Chapter 8). 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 Chapter 8).

1.8.29 IE Bulletins on Measures to Mitigate Small-Break LOCAs and Loss of Feedwater Accidents (II.K.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.

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.

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 plant computer 1.47 bypass application. 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 McGuire Work Request Program governs all maintenance activities performed at McGuire. 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.

C.1.17

The design of McGuire 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.

1.8.30 Commission Orders on B&W Plants (II.K.2)

1.8.30.1 Thermal Mechanical Report - Effect of High-Pressure Injection on Vessel Integrity for Small-Break Loss-of-Coolant Accident with No Auxiliary Feedwater

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.

1.8.31 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. 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 McGuire Nuclear Station. The McGuire specific response was provided on January 20, 1982 in response to Generic Letter 81-21.

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) were utilized in the implementation of McGuire plant specific operating procedures.

1.8.32 Sequential Auxiliary Feedwater Flow Analysis (II.K.2.19)

Subsequent to the issuance of NUREG-0737, the NRC has completed a generic review on this subject and concluded that the concerns expressed in Section 1.8.32 are not applicable to NSSSs with inverted U-tube steam generators such as those designed by Westinghouse. Therefore, this item is not applicable and no further action is necessary.

1.8.33 Final Recommendations of B&O Task Force (II.K.3)

1.8.33.1 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 WCAP-9804 for further discussion.

1.8.33.2 Reporting Safety Valve and Relief Valve Failures and Challenges (II.K.3.3)

Duke Energy will promptly report to the NRC any failure of a McGuire 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.

1.8.33.3 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.

1.8.33.4 Proportional Integral Derivative Controller Modification (II.K.3.9)

Westinghouse has completed its review of the proportional integral derivative (PID) controller installed on the McGuire PORVs and a value of “zero” for the pressurizer PID controller rate time constant was determined. The McGuire time constant has been adjusted accordingly.

1.8.33.5 Proposed Anticipatory Trip Modification (II.K.3.10)

The design of McGuire Nuclear Station did not feature a reactor trip on turbine trip. This trip was removed from the McGuire design to prevent unnecessary reactor trips, particularly during initial startup. Unnecessary reactor trips should be avoided to minimize reactor coolant system thermal cycles and challenges to the reactor coolant system protective devices. The removal of this anticipatory trip was possible due to the full load rejection capability of McGuire.

The McGuire trip system keeps surveillance on process variables which are directly related to equipment mechanical limitations, such as pressure, pressurizer water level (to prevent water discharge through safety valves) and also on variables which directly affect the heat transfer capability of the reactor (e.g., flow, reactor coolant temperatures). Still other parameters utilized in the reactor trip system are calculated from various process variables. In any event, whenever a direct process or calculated variable exceeds a setpoint, the reactor will be shut down in order to protect against either gross damage to fuel cladding or loss of system integrity which could lead to release of radioactive fission products into the containment.

An analysis was conducted to determine the potential for pressurizer PORV challenges following a turbine trip from full power both with and without an immediate reactor trip on turbine trip. This analysis considered both normal plant response and cases assuming the failure of certain central systems that can influence challenges to the pressurizer PORVs. Two types of control system failures were considered: failure of all steam dump valves to open on demand (not including the steam generator PORVs); and complete failure of pressurizer spray to function on demand. Partial failures (for example, failure of half of the steam dump valves) were not considered.

The analysis demonstrated that if all of the steam dump valves failed to open the pressurizer PORVs would be challenged regardless of the presence or absence of an immediate reactor trip on turbine trip at full power. If there was no failure of the steam dump valves the absence of the subject trip would result in challenges to the pressurizer PORVs whereas the presence of such a trip would not challenge the PORVs.

Installation of a direct reactor trip on turbine trip would only protect against PORV challenges initiated by a narrow range of events, that is turbine trips not initiated by a reactor trip or a safety injection and occurring at or near full power. Furthermore, valves identical to the McGuire PORVs and PORV block valves have been subjected to extensive steam flow testing. This testing was conducted at Duke's Marshall Steam Station in conjunction with the EPRI valve testing program. The testing demonstrates that the McGuire PORVs and PORV block valves meet all functional and design requirements and provides added assurance of proper PORV and PORV block valve operation.

Duke has installed a direct reactor trip on turbine trip to provide this additional protection against PORV challenges. The reactor trip on turbine trip will be generated by either of the following signals:

1. Four-out-of-four turbine stop valves closed
2. Two-out-of-three turbine auto-stop oil pressure low

The four turbine stop valve signals will be developed through the actuation of independent limit switches mounted on the stop valve assemblies. Each of the four turbine stop valve signals can be tested individually from the control room through the digital electrohydraulic (DEH) control panel. The turbine auto-stop oil system is the medium through which a turbine trip is initiated. Turbine auto-stop oil pressure is measured by three independent pressure switches which are mounted located adjacent to the turbine.

The limit switches and pressure switches used in this application are similar to those used in other Class 1E applications in the plant. Although the main turbine-generator is not seismic Category I, these limit switches and pressure switches are seismically qualified and the associated cables will be installed in accordance with the McGuire separation criteria.

Each turbine stop valve limit switch and each turbine auto-stop oil pressure switch provide an input to both trains of the solid state protection system (SSPS). If either logic function as described above is satisfied, a reactor trip signal will be generated provided reactor power is greater than approximately 48% (P8).

It should be noted that the reactor trip on turbine trip was originally part of the McGuire SSPS design with the exception that the P8 interlock will be substituted for the P7 interlock. It is therefore concluded that reinstituting this trip will not degrade the existing protection system since all separation, testing, and reliability considerations are in accordance with the original SSPS design.

1.8.33.6 Justification for Use of Certain PORV'S (II.K.3.11)

See Section 1.8.23.

1.8.33.7 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 completed their review of ECCS data provided by licensees and determined that no changes in the Technical Specifications were required. It was therefore Duke Energy's conclusion that the surveillance and reporting requirements in the McGuire Technical Specifications adequately address outages of ECCS systems and components, and no further action was required.

1.8.33.8 Effects of Loss of Alternating-Current Power on Pump Seals (II.K.3.25)

At the McGuire 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.

1.8.33.9 Revised Small-Break Loss-of-Coolant-Accident Methods to Show Compliance with 10CFR 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 10CFR 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 McGuire is conservative and in conformance with Appendix K to 10CFR 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. McGuire provided a spectrum (i.e., pipe equivalent diameters of 2, 3, 4 and 5 inches) of plant-specific small-break analyses using the NOTRUMP code described by WCAP-10079 and WCAP-10054 (non-proprietary versions are WCAP-10080 and WCAP-10081). The NRC approved McGuire's plant-specific analyses in B.J. Youngblood (NRC) letter to H.B. Tucker (Duke), February 24, 1987.

1.8.33.10 Plant-Specific Calculations to Show Compliance with 10CFR 50.46 (II.K.3.31)

See Section 1.8.33.9 above.

1.8.33.11 Upgrade Emergency Preparedness (III.A.1.1)

See Section 13.3.

1.8.33.12 Upgrade Emergency Support Facilities (III.A.1.2)

See Section 13.3.

1.8.34 Primary Coolant Sources Outside Containment (III.D.1.1)

Periodic leak rate test procedures have been written for systems carrying radioactive fluids outside of containment. The following systems are included: Safety Injection, Residual Heat Removal, Containment Spray, Nuclear Sampling, Boron Recycle, Chemical Volume and Control, Refueling Water, Liquid Waste, and Waste Gas. These tests are performed at refueling cycle intervals or less. The tests are conducted 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.

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

After the occurrence of an uncontrolled gaseous release at North Anna Unit 1 in September 1979, and the NRC issuance of IE Circular 79-21, an evaluation of the potential release pathways for liquid and gaseous material was performed for McGuire Nuclear Station. This evaluation resulted in several minor station modifications to further reduce the possibility of an uncontrolled release at McGuire. These modifications included the installation of catch basins, curbs, shelters, and covers in various areas of the station.

1.8.34.1 In-Plant Radiation Monitoring (III.D.3.3)

See Section 12.1.4.

1.8.34.2 Control Room Habitability (III.D.3.4)

See Section 6.4.

1.8.35 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 NUREG-0737, 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.

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1.9 Response to Generic Letter 83-28

This section addresses the history of Duke's response to "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.

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).

Summary of Duke's Response

The reactor trip investigation program was implemented at McGuire 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 station 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 reviewed the post trip review program and procedures for McGuire and 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 (Item 1.1).

Summary of Duke's Response

Each unit has three primary sources available on the plant computer (OAC) for collecting data for analyzing unscheduled reactor shutdowns. These are:

1. Sequence of Events (SOE) Inputs
2. Alarm Log, and
3. Transient Monitor, Application.

Items 1 and 2 are used to determine the sequence of events while the transient monitor application is used for analog data trending.

Sequence of Events (SOE) inputs to the plant computer record various plant parameters including those pertinent to reactor trip investigations. When an input parameter alarms, the fact is stored to a log file with the precise time of the event. A representative list of parameters monitored is listed in Reference 2.

The plant computer monitors and records all digital alarms and safety-related pumps, valves, and motors change of status indications that are received in the control room. This information is stored in an alarm log file, and is available for review or printing.

The transient monitor application records data both before and after an event. The monitor is "tripped" automatically on any one of a number of events (e.g. reactor trip breaker opening.) 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 reviewed Duke's submittals (References 2, 4, 5, 6) and issued a safety evaluation report (Reference 7). The SER closed Item 1.2 with the exception of the record retention issue. Subsequently, Duke established a category of "Lifetime" for post-trip review data (Reference 4), thus accepting the NRC's position. The NRC confirmed compliance with GL83-28 for this issue in an inspection report 87-11 (Reference 29).

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).

Summary of Duke's 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 Energy group to review for applicability and safety significance. If applicable, it is then transmitted to the station staff for implementation.

Any vendor service on reactor trip system components is either controlled by Dukes's Quality Assurance Program, or by the vendor's QA program. If the service is vendor QA controlled, Duke is responsible for identifying to the vendor that QA is required. (Reference 2).

Status

The NRC evaluated Duke's response to this item, and found that Duke's program was 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

Part One 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.

Summary of Duke’s Response

The *McGuire 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 Energy 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 Administrative Policy Manual for Nuclear Stations (APM) established procedures to ensure that safety-related technical information is reviewed, evaluated, and resolved. The APM also requires the evaluation of all safety-related equipment failures. These requirements are further discussed in Reference 11 and the APM.

The NRC requested additional information on the NUTAC and VETIP programs, for which Duke responded (Reference 5 and Reference 11). The NRC did not issue an SER.

Status

The NRC 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.)

For Part Two the NRC issued Generic Letter 90-63 which relaxed the staff position on this item and replaced the requirements with a new set of requirements. Duke supported NUTAC and participates in VETIP (Reference 11). The NRC requested additional information and evaluation of the NUTAC program (Reference 15). Duke responded that the VETIP is valid and sufficient, and the NRC should reconsider its evaluation (Reference 5). The item remains open.

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).

Summary of Duke's 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 (Reference 16).

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.

Summary of Duke's 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 (Reference 16).

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.

Summary of Duke's Response

Two modifications that apply to the McGuire Reactor Trip Breakers have been recommended by Westinghouse. Both modifications have been implemented on the McGuire Reactor Trip Breakers. (Reference 2.)

Status

This item was reviewed and found to be acceptable. This item is closed. (Reference 17.)

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: (1) periodic maintenance including lubrication, housekeeping and other vendor recommendations, (2) trending of parameters, (3) life testing of the breakers, (4) 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 Duke will trend the breaker performance data to ensure reliable operation of the RTB.

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, 5, and 17 through 23) and are closed. The requirements of Parts Three and Four were removed by a supplement to Generic Letter 83-28 (Reference 31).

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.

Summary of Duke's Response

The modification was implemented at McGuire prior to the issuance of Generic Letter 83-28. A further modification was implemented to provide for independent fusing of safety-related and non-safety-related circuits.

Status

The modifications were reviewed by the NRC and found acceptable as documented in Reference 24. The issue has not been formally closed as an item in the Generic Letter, but no further action is expected.

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 McGuire, 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.

Summary of Duke's Response

The reactor trip system at McGuire is designed to allow on-line functional testing. Functional testing is performed for both McGuire Units as described in the McGuire Unit 2 operating license.

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 54 (Unit 1) and 35 (Unit 2).

Status

The NRC reviewed Duke's submittals and found them to be adequate. Items 4.5.1 and 4.5.2 are closed (References 16 and 27, respectively). Changes to the Technical Specifications were submitted (References 25, 26) and approved (Reference 28). Item 4.5.3 on on-line functional testing was closed by SER per Reference 32.

1.9.12 Implementation Inspection

A special inspection was conducted March 16-20, 1987 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 50-369/87-11 and 50-370/87-11 (Reference 29).

The inspection team examined Duke's implementation of the post-trip review program at McGuire 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 McGuire; this included a review of the "McGuire Nuclear Station Quality Standards Manual for Structures, Systems, and Components" (generally referred to as the QSM) and its implementation. The team also reviewed qualified reviewer training. The inspectors identified two items during this review. First, the QSM had not been updated since issue approximately two years earlier. The QSM has since been updated to Nuclear Fleet Procedure AD-EG-ALL-1107, "Quality Classifications" and a program has been instituted for continued updating, as required.

The second item involved the certification exams for qualified reviewers, in that the same certification exam was being used for all training classes including retests. In response to the item, Duke is developing a pool of questions to safeguard future exams.

The vendor interface program includes the control of vendor technical information. The inspectors verified the development and implementation of a detailed program to control and distribute information. Specific technical bulletins were reviewed and followed through the process. Quality Assurance audits were reviewed. No discrepancies were noted in this area.

Post maintenance testing of the reactor trip circuit breakers was reviewed including observation of maintenance and testing by the inspector. No deficiencies were noted in maintenance or testing of the breakers, though the inspector offered editorial changes to the procedure for enhancement of the procedure.

The inspector also verified vendor recommended modifications to the reactor trip breakers and associated documentation with no discrepancies. The inspector noted the cleanliness of the reactor trip breaker environment.

The inspection team also reviewed the preventative maintenance and surveillance program for reactor trip breakers. The inspectors were satisfied with the program and its implementation with the exception of trending of reactor trip breaker parameters (Item 4.2.2). As this was the subject of several communications between NRC and Duke (References 1, 2, 5, 17, 19, 20), the inspection team did not cite this item as a violation of commitment, but rather as a violation of license amendment two (unit two only) which required trending of data. Since data was being recorded just not formally trended, the violation was cited as Level five, the least severe of violation levels. Duke has established a formal trending program.

In sum, the inspection verified the implementation of an effective program pursuant to Generic Letter 83-28 with only one administrative violation.

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 May 30, 1985.

4. Letter from Mr. H. B. Tucker (DPC) to Mr. H. R. Denton (NRC), dated October 1, 1986.

5. Letter from Mr. H. B. Tucker (DPC) to Mr. H. R. Denton (NRC), supplemental response to items 1.2, 2.2, 3.1, 4.2, and 4.5 of Generic Letter 83-28, dated May 24, 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. D. S. Hood (NRC) to Mr. H. B. Tucker (DPC), transmitting SER for Generic Letter 83-28, Item 2.1 (Part 2) for McGuire Nuclear Station, dated Dec. 29, 1986.
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 1, 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. D. S. Hood (NRC) to Mr. H. B. Tucker (DPC), transmitting SER for Generic Letter 83-28, Items 2.2.1, dated October 14, 1987.
15. Letter from Ms. E. G. Adensam (NRC) to Mr. H. B. Tucker (DPC), transmitting Requests for Additional Information for Generic Letter 83-28 Items 1.2, 2.2, and 4.5, dated May 1, 1985.
16. Letter from Mr. T. M. Novak (NRC) to Mr. H. B. Tucker (DPC), transmitting SER for Items 3.1.1, 3.1.2, 3.1.3, 3.2.1, 3.2.2, 3.2.3, 4.1, and 1.5.1 of Generic Letter 83-28, dated October 31, 1985.
17. Letter from Mr. T. M. Novak (NRC) to Mr. H. B. Tucker (DPC), transmitting Request for changes to the Preventative Maintenance Program for Reactor Trip Breakers - McGuire Units 1 and 2, dated February 22, 1985.
18. Letter from Mr. H. B. Tucker (DPC) to Mr. H. R. Denton, Attention: Ms. E. G. Adensam (NRC), delaying response to Reference 17 dated April 25, 1985.
19. Letter from Mr. B. J. Youngblood (NRC) to Mr. H. B. Tucker (DPC), transmitting Interim Report Regarding Maintenance and Trending for Reactor Trip Breakers - McGuire Nuclear Station, Units 1 and 2, dated February 17, 1986.
20. Letter from Mr. H. B. Tucker (DPC) to Mr. H. R. Denton, Attention: Mr. B. J. Youngblood (NRC) responding to Reference 19, dated April 14, 1986.
21. Letter from Mr. D. S. Hood (NRC) to Mr. H. B. Tucker (DPC), transmitting RAI regarding Item 4.2.2 of Generic Letter 83-28 for Catawba and McGuire, dated July 23, 1987.
22. Letter from Mr. H. B. Tucker (DPC) to the NRC, supplemental response to Item 4.2.2, dated August 21, 1987.
23. Letter from Mr. D. S. Hood (NRC) to Mr. H. B. Tucker (DPC) transmitting SER for Items 4.2.1 and 4.2.2 for McGuire, dated October 20, 1987.
24. NUREG-0422 (SER for McGuire Nuclear Station), Supplement 7, May 1983.

25. 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.

26. Letter from Mr. H. B. Tucker (DPC) to Mr. H. R. Denton, Attention: Ms. E. G. Adensam (NRC), providing supplemental information for Reference 25, dated September 11, 1985.

27. Letter from Mr. D. S. Hood (NRC) to Mr. H. B. Tucker (DPC), transmitting SER for Item 4.5.2, dated January 28, 1987.

28. Letter from Mr. B. J. Youngblood (NRC) to Mr. H. B. Tucker (DPC), transmitting Facility Operating License Amendments 54 (Unit 1) and 35 (Unit 2), date April 7, 1986.

29. IE Inspection Report 50-369/87-11 and 50-370/87-11, transmitted by Mr. A.R. Herdt's (NRC) letter to Mr. H.B. Tucker (DPC), dated May 1, 1987.

30. Letter from Mr. H. B. Tucker (DPC) to the NRC, transmitting response to Violation 50-370/87-11-01, dated May 29, 1987

31. Letter transmitting GL 83-28 Supplement 1, dated October 7, 1992.

32. Letter from NRC, dated June 29, 1989 closing out GL 83-28 item 4.5.3.

33. Letter from T.M. Novak (NRC) to H.B. Tucker (Duke), dated May 17, 1983, Safety Evaluation Related to McGuire Unit 1 Reactor Trip Breakers.

34. Letter from E.G. Adensam (NRC) to H.B. Tucker (Duke), dated July 21, 1983, Evaluation of Undervoltage Trip Attachment Failures.

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1.10 Drawings and Other Detailed Information

1.10.1 Piping and Instrumentation Diagrams

Table 1-5 lists each diagram and provides a cross-reference between the drawing number and the FSAR figure number.

Figure 6-1 provides the symbols and abbreviations used in the flow diagrams.

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1.11 Regulatory Guide 1.97, Revision 2 - Review for McGuire Nuclear Station

1.11.1 Introduction

The Duke Energy response addressing Regulatory Guide 1.97, Revision 2 for McGuire Nuclear Station has been formulated as an integral part of the overall company plan to respond to Nuclear Regulatory Commission Generic Letter 82-33, which transmitted Supplement 1 to NUREG-0737 "Requirements for Emergency Response Capability." The overall company plan consisted of the following task areas in addition to Regulatory Guide 1.97:

1. Control Room Review (CRR)
2. Safety Parameter Display System (SPDS)
3. Emergency Procedure (EP) upgrade
4. Emergency Response Facilities

These other task areas and their relationships with the Regulatory Guide 1.97 Revision 2 review effort are described in detail in the appropriate section of Response to Supplement 1 NUREG-0737.

1.11.2 Background

Duke Energy began developing a formal review plan to address Revision 2 of Regulatory Guide 1.97 in January, 1981. This plan is part of the larger integrated plan originated to respond to Nuclear Regulatory Commission Generic Letter 82-33. The formal Regulatory Guide 1.97 Revision 2 review plan was established in June, 1982.

1.11.3 Scope

The document contains Duke Energy's report on the compliance of McGuire Nuclear Station with NUREG 0737 Supplement 1 which references U. S. Nuclear Regulatory Commission Regulatory Guide 1.97, Revision 2 issued in December 1980. Included in the report are descriptions of the Duke Energy accident monitoring instrumentation position and a detailed comparison table as requested in Supplement 1 to NUREG-0737.

1.11.4 Plan Description

The plan for addressing Regulatory Guide 1.97, Revision 2 for the McGuire Nuclear Station design consisted of the following three phases:

1. Review
2. Assessment
3. Implementation

The Review phase consisted of the establishment of plant specific accident monitoring instrumentation requirements and a detailed review of existing instrumentation versus these requirements. The plant specific accident monitoring instrumentation criteria and clarifications are established in a Duke Energy position statement contained in McGuire Final Safety Analysis Report (FSAR) Section 1.11.5 and Section 7.5.

The Station Emergency Procedures and Safety Parameter Display System (SPDS) Critical Safety Functions served as inputs for variable selection. Both of these documents were derived from the McGuire plant specific version of the Westinghouse Owners Group Emergency Response Guidelines and associated Critical Safety Function Status Trees. Modifications in either the Emergency Response Guidelines or the associated Critical Safety Function Status Trees will be incorporated as appropriate into the Regulatory Guide 1.97 accident monitoring instrumentation in accordance with the integrated plan for addressing Supplement 1 to NUREG-0737.

Emergency Procedures provide the lead guidance for selection of Type A variables. The SPDS Critical Safety Functions as derived from the Critical Safety Function Status Trees form the basis for selection of the Type B and C variables. The Type D and E variables are selected on the basis of individual plant specific system design requirements.

The Review phase provided the basis for the Duke position statement on accident monitoring and a listing of variances of installed accident monitoring instrumentation versus the plant specific needs. These variances were reviewed by the appropriate parties as provided for in the interfaces set up by the integrated plan described in Section 2.0, “Integrated Plan and Schedule” of the Response to Supplement 1 NUREG-0737.

The Assessment phase utilized the input generated during the Review phase and consisted of an evaluation of all identified variances in accident monitoring instruments to determine what further action was merited. Guidance from the NUTAC committee's report on Regulatory Guide 1.97 (Accident Monitoring Instrumentation) Implementation Guideline was considered in development of the comparison sheets. Control Room and operator interfaces were coordinated with the Control Room Review. The Assessment phase produced two primary end products, specifically, the results of the evaluation described above and a RG 1.97 response report for submittal to the NRC.

The Implementation phase of the plan consisted of designing and installing those modifications to accident monitoring instrumentation as identified in the Assessment phase. This phase was an integrated effort to insure effective orderly implementation of all Control Room Review, SPDS and RG 1.97 modifications or additions.

Duke Energy is an active participant in the Nuclear Utility Task Action Committee on Emergency Response Capability, and resultant NUTAC guidance has been considered and utilized in the review, assessment, and implementation phases of our review plan.

1.11.5 Duke Energy Position on Accident Monitoring Instrumentation

1.11.5.1 Accident-Monitoring Instrumentation

The criteria and requirements contained in ANSI/ANS-4.5-1980, “Criteria for Accident Monitoring Functions in Light-Water Cooled Reactors,” are considered by Duke Energy to be generally acceptable for providing instrumentation to monitor variables for accident conditions subject to the clarifications defined below.

1.11.5.1.1 TYPE A Variables

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 McGuire 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

Modifications to the plant design since the initial review phase of RG 1.97 have created additional specific manually controlled actions for which no automatic control is provided. The following additional variables are determined to be type A for McGuire Nuclear Station:

10. Containment Spray Heat Exchanger RN flow
11. Diesel Generator Cooling Water Heat Exchanger RN flow

These additional variables were not part of or addressed during the initial review, assessment, and implementation phase of RG 1.97, Rev. 2. They were not required for post accident monitoring. However, as a result of the plant modifications, these variables were determined to be needed for post accident monitoring and are documented in RG 1.97, Rev. 2 Review Table 1-6.

1.11.5.1.2 TYPE B and C Variables

Type B and C variable selection is based on the SPDS Critical Safety Functions. 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 McGuire SPDS and the emergency operating procedures, it is Duke's position that they should also be identified as the plant safety functions for accident monitoring (i.e., the basis for Type B & 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.

1.11.5.1.3 Design and Qualification Criteria

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

1.11.5.1.3.1 Design and Qualification Criteria - Category 1

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

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

   Several 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.

   For the present configuration, qualification applies from the sensor through the channel isolation device up to and including the control board display device. The control board display devices were installed as non-safety, but do not share any output isolation devices with non-qualified equipment such as the computer. Although the indication outputs were installed as non-safety and cable separation was not guaranteed for the cable runs between the PCS cabinets and the control boards, this is considered acceptable since the isolator is qualified for safety related use and the indicators, cabling, and cable trays have the appropriate seismic withstand capability to be considered safety related. Power for the indicator is provided from Class 1E sources either directly or via the PCS cabinets. The existing cabling from the PCS cabinets to the control boards is utilized. This is considered acceptable since these cables are armored, contain low energy circuits, and are located in a protected area of the plant. The location of the isolation devices provides for accessibility for maintenance during accident conditions. For variables RCS Pressure, RCS Hot Leg Water Temperature, Pressurizer Level, and Steam Generator Narrow Range Level, the control board recorded channels have been enhanced through the addition of isolators such that the control board recorders will not share isolators with the non-safety computer.

2. No single failure within either the accident monitoring instrumentation, its auxiliary supporting features, or its power sources, concurrent 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 per the McGuire Final Safety Analysis Report Section 8.3.1.2.7 (except as noted in 1 above). 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 UFSAR) 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 Energy Topical Report, "Quality Assurance Program"
McGuire Nuclear Station Final Safety Analysis Report	Chapter 17

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. Recorders which are utilized as the primary display device will be seismically qualified. 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 displayed on non-seismically qualified recorders or 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.

1.11.5.1.3.2 Design and Qualification Criteria - Category 2

1.11.5.1.3.2.1 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 McGuire NUREG 0588 submittal and the UFSAR Section 3.11. Seismic qualification is in accordance with IEEE 344-1971 as described in Section 3.10 of the McGuire UFSAR. Quality Assurance in the design, procurement, and installation of these QA Condition 1 Instrumentation Systems is provided as described in the Duke Energy Topical Report “Duke 1” and UFSAR Chapter 17. These instruments are powered from Class 1E Power sources (as described in Chapter 8 of the UFSAR) and are backed by batteries where a momentary power interruption is not tolerable.

1.11.5.1.3.2.2 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 Energy standard practices. Duke considers that this is adequate to assure the quality of the subject instrumentation.

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

1.11.5.1.3.2.3 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 a monitor or by other appropriate means.

The method of display may be by dial, digital, 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.

1.11.5.1.3.3 Design and Qualification Criteria - Category 3

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, 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.

1.11.5.1.4 Additional Criteria for Categories 1 and 2

In addition to the criteria of Duke Position 1.11.5.1.3, 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.

1.11.5.1.5 Additional Criteria for All Categories

In addition to the above criteria, the following criteria apply to all instruments identified in this document:

1. 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.
2. Whenever means for removing channels from service are included in the design, the design facilitates administrative control of the access to such removal means.
3. The design facilitates administrative control of the access to all setpoint adjustments, module calibration adjustments, and test points.
4. 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 provided recommendations as to the type and location of displays for added instrumentation.

5. To the extent practicable, the instrumentation is designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.
6. To the extent practicable, monitoring instrumentation inputs are from sensors that directly measure the desired variables.
7. 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.
8. Periodic checking, testing, calibration, and calibration verification are in accordance with the applicable portions of the McGuire FSAR Chapter 7.

1.11.5.2 Operation Monitoring (Type D) and Effluent Release Monitoring (Type E) Instrumentation

1.11.5.2.1 Definitions

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.

1.11.5.2.2 Operator Usage

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.

1.11.5.2.3 Design and Qualification Criteria - Types D and E

The design and qualification criteria for safety system operation monitoring (Type D) and effluent release monitoring (Type E) instrumentation are provided in Sections 1.11.5.1.3, 1.11.5.1.4, and 1.11.5.1.5.

1.11.6 Regulatory Guide 1.97 Comparison

Table 1-6 is the result of the Assessment phase of the Regulatory Guide 1.97 Revision 2 review plan. Each page contains information regarding the comparison of the particular variable with the recommendations of the Regulatory Guide. Instrument ranges, design, environmental qualification, type of display, and position statements are provided for each variable named in Table 2, “PWR Variables,” of Regulatory Guide 1.97 Revision 2.

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1.12 McGuire Response to Beyond-Design-Basis External Event Fukushima Related Required Actions

1.12.1 Introduction

On March 11, 2011, an earthquake-induced tsunami caused Beyond-Design-Basis (BDB) flooding at the Fukushima Dai-ichi Nuclear Power Station in Japan. The flooding caused by the tsunami rendered the emergency power supplies and electrical distribution systems inoperable resulting in an extended loss of alternating current (AC) power (ELAP) in five of the six units on the site. The ELAP led to the loss of core cooling as well as spent fuel pool cooling capabilities and a significant challenge to containment. All direct current (DC) power was lost early in the event on Units 1 & 2 and after some period of time at the other units. Units 1, 2, and 3 were affected to such an extent that core damage occurred and radioactive material was released to the surrounding environment.

The US Nuclear Regulatory Commission (NRC) assembled a special task force, the Near-Term Task Force (NTTF) in order to advise the Commission on actions the US Nuclear Industry should undertake in order to preclude a release of radioactive material in response to a natural disaster such as that seen at Fukushima Dai-ichi. NTTF members created NRC Report “Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” referred to as the “90-day Report,” which contained a large number of recommendations for improving safety at US nuclear power sites.

Subsequently, the NRC issued Order EA-12-049, "Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" (Agencywide Documents Access Management System (ADAMS) Package Accession No. ML12054A736) (Reference 1), Order EA-12-051, “Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation” (ADAMS Package Accession No. ML12056A044) (Reference 2) to implement strategies for Beyond-Design-Basis External Events (BDBEE), and reliable spent fuel pool instrumentation, respectively.

1.12.2 Order EA-12-049

NRC Order EA-12-049 was effective immediately and directed Duke Energy to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, maintain containment, and maintain spent fuel pool cooling in the event of a beyond-design-basis external event.

The Nuclear Energy Institute (NEI), working with the nuclear industry, developed guidelines for nuclear stations to implement the strategies specified in NRC Order EA-12-049. These guidelines were published in the NEI 12-06 document entitled “Diverse and Flexible Coping Strategies (FLEX) Implementation Guide” (Reference 3). This guideline was endorsed by the NRC in final interim staff guidance (ISG) document JLD-ISG-2012-01, Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design- Basis External Events, Revision 0, dated August 29, 2012 (ML12229A174) (Reference 4.)

The NEI 12-06 FLEX implementation guide adopts a three-phase approach for coping with a BDB event.

- Phase 1 – the initial phase requires the use of installed equipment and resources to maintain or restore core cooling, containment, and SFP cooling capabilities.

- Phase 2 – The transition phase requires providing sufficient portable onsite equipment to maintain or restore these functions until resources can be brought from off site.
- Phase 3 – The final phase requires obtaining sufficient offsite resources to sustain these functions indefinitely.

This three-phase approach was utilized to develop the FLEX strategies and modifications for McGuire. The NEI 12-06 guidance also establishes programs for the long-term management of FLEX equipment and strategies (See Section 1.12.4)

1.12.3 Order EA-12-051

NRC Order EQ-12-051 directed Duke Energy to implement the following changes to provide additional Spent Fuel Pool Level Instrumentation.

- provide a primary and back-up level instrument that will monitor water level from the normal level to the top of the used fuel rack in the pool;
- provide a display in an area accessible following a severe event; and
- provide independent electrica power to each instrument channel and provide an alternate remote power connection capability.

The Nuclear Energy Institute (NEI), working with the nuclear industry, developed guidelines for nuclear stations to implement this order and published this guidance as NEI 12-02, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation". (Reference 5) This guideline was endorsed by the NRC in final interim staff guidance (ISG) document JLD-ISG-2012- 03, Compliance with Order EA-12-051,

Reliable Spent Fuel Pool Instrumentation. (Reference 7). This guidance was utilized to develop the plant modifications for McGuire.

The NEI 12-02 guidance also establishes programs for the long-term management of FLEX equipment and strategies (See Section 1.12.4)

1.12.4 BDB Program

McGuire has committed to a formal program for mitigating beyond-design-basis external events. Program changes are controlled in accordance with NEI 12-06, Section 11.8, as endorsed by the NRC.

The program describes items such as a list of FLEX equipment, the BDB Storage Buildings, initial and periodic testing, FLEX equipment maintenance, and actions to be taken in the event of equipment unavailability.

The program also describes the Spent Fuel Pool Instrumentation program requirements including procedures, testing and calibration, and quality assurance.

1.12.5 References

1. Order EA-12-049, "Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" dated March 12, 2012 (ML12054A736).
2. Order EA-12-051, "Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" dated March 12, 2012 (ML12056A044).

3. NEI 12-06, Diverse and Flexible Coping Strategies (FLEX) Implementation Guide, Revision 0, dated August 2012 (ML12242A378).
4. NRC Interim Staff Guidance JLD-ISG-2012-01, Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, Revision 0, dated August 29, 2012 (ML12229A174).
5. NEI 12-02, Rev. 1, Industry Guidance for Compliance with NRC Order EA-12-051, “To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation” August 2012.
6. NEI 12-06, Rev. 0 , Diverse and Flexible Coping Strategies (FLEX) implementation Guide. August 2012.
7. JLD-ISG-2012- 03, Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation, August 29, 2012, (ML12221A339)

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