UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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61 FORSYTH STREET SW SUITE 23T85
ATLANTA, GEORGIA 30303-8931

April 20, 2000

EA 00-101

Duke Energy Corporation
ATTN: Mr. H. B. Barron
Vice President
McGuire Nuclear Station
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: NRC INTEGRATED INSPECTION REPORT NOS. 50-369/00-02 AND 50-370/00-02

Dear Mr. Barron:

This refers to the inspection conducted between February 27, and April 1, 2000, at McGuire Nuclear Station. The enclosed report presents the results of that inspection.

During the five-week period covered by this inspection, your conduct of activities at the McGuire facility was generally characterized by safety-conscious operations, sound engineering and maintenance practices, and careful radiological work controls.

Based on the results of this inspection, the Nuclear Regulatory Commission (NRC) has determined that two violations of NRC requirements occurred. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section VII.B.1 of the Enforcement Policy. The NCVs are described in the subject inspection report. If you contest the violation or severity level of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II, the Resident Inspector at the McGuire Nuclear Station, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.790(a) of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and any response you choose to make will be placed in the NRC Public Document Room.

Sincerely,

/RA/

Charles R. Ogle, Chief
Reactor Projects Branch 1
Division of Reactor Projects

Docket Nos. 50-369, 50-370
License Nos. NPF-9, NPF-17

Enclosure: NRC Inspection Report 50-369/00-02
50-370/00-02

cc w/encl: (See page 3)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-369, 50-370
License Nos:	NPF-9, NPF-17
Report No:	50-369/00-02, 50-370/00-02
Licensee:	Duke Energy Corporation
Facility:	McGuire Nuclear Station, Units 1 and 2
Location:	12700 Hagers Ferry Road Huntersville, NC 28078
Dates:	February 27, 2000 - April 1, 2000
Inspector:	S. Shaeffer, Senior Resident Inspector
Approved by:	C. Ogle, Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

EXECUTIVE SUMMARY

McGuire Nuclear Station, Units 1 and 2
NRC Inspection Report 50-369/00-02, 50-370/00-02

This integrated inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covered a five-week period of resident inspections.

Operations

- A non-cited violation was identified for non-compliance with Technical Specifications (TS) 3.6.10 and 3.0.3 between November 30, 1999 and December 10, 1999, with respect to the Unit 1 annulus ventilation (VE) system. Poor communication among operators resulted in the licensee's failure to recognize that a low pressure gauge reading was potentially indicative of a degraded power supply that rendered the associated VE system train inoperable beyond the specified time limits of TS 3.6.10. Consequently, internal plant procedures for assessing and maintaining operability of the opposite VE train components (i.e., availability of the associated emergency diesel generator) were not implemented to maintain compliance with TS 3.0.3. (Section O8.4)

Maintenance

- Seat leakage was identified on two main steam line drain valves located upstream of the C steam generator main steam isolation valve. The leakage was promptly identified and isolated. Use of alternate valves to maintain compliance with applicable Technical Specification requirements was adequate. (Section M2.1)

Engineering

- A non-cited violation of 10 CFR 50, Appendix B, Criterion V, was identified concerning Unit 1 ice condenser basket coupling screws that were found to be missing during the end-of cycle 13 refueling outage (September 17, 1999 - October 31, 1999) and, therefore, not installed in accordance with station drawings. (Section E8.1)

Report Details

Summary of Plant Status

Unit 1

Unit 1 began the inspection period at 100 percent of licensed thermal power. On March 22, 2000, the unit reduced power to approximately 98.5 percent for a short time as a conservative measure to support realignment of the normal air supply to 1CF-20AB (main feedwater regulating valve for steam generator C). The air supply was being changed due to the development of a leak on the normal supply. No feedwater transients occurred during the activities and the unit was returned to 100 percent power, where it operated for the remainder of the inspection period.

Unit 2 operated at approximately 100 percent power for the duration of the inspection period.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

The inspector conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was professional and safety-conscious. Detailed protection and monitoring of safety-related components by the licensee to limit operability challenges when equipment on the opposite train was unavailable was noted by the inspectors on a number of activities. Other specific events and noteworthy observations are detailed in the sections which follow.

O1.2 10 CFR 50.72 Notification

a. Inspection Scope (71707)

During the inspection period, the licensee made the following notification to the NRC as required by 10 CFR 50.72. The inspector reviewed the event for impact on the operational status of the facility and equipment.

b. Observations and Findings

On March 2, 2000, the licensee reported a condition outside the design basis of the plant in accordance with 10 CFR 50.72(b)(ii)(B). During a review of criticality calculations for the McGuire Unit 1 and 2 spent fuel pool (SFP) storage requirements, potential non-conservative assumptions were identified that could have resulted in a Keff value in excess of 0.95 at zero parts per million (ppm) boron concentration in the spent fuel pool. The design basis of the SFPs is that the stored fuel will remain less than or equal to 0.95 Keff when fully flooded with unborated water. During the inspection period, the inspector verified that each SFP had boron concentrations in excess of Core Operating Limits Report (COLR) limit of 2675 ppm, as required by Technical Specification (TS). This TS parameter was being verified by the licensee on a weekly basis. Upon further evaluation, the licensee determined that no additional compensatory measures were required to assure current operability. Engineering reviews confirmed that the SFPs would remain operable with boron concentrations in excess of 500 ppm. The licensee submitted Licensee Event Report (LER) 50-369/00-03-00, Non-Conservatism in Spent Fuel Pool Criticality Calculation, on March 30, 2000, which further addressed this issue (see Section O8.3).

c. Conclusions

The inspector concluded that the licensee reported the described event in accordance with regulatory requirements. Immediate corrective actions were appropriate.

O3 Operations Procedures and Documentation

O3.1 Review of Operator Aid Computer (OAC) Points for Parameters Deleted from Alarms (71707)

During the inspection period, the inspector reviewed the licensee’s method for periodically reviewing OAC points, alarms, or those parameters which have defaulted values and questioned whether an adequate review was being performed. A lack of review for these types of OAC points resulted in operation outside of the design basis for a similar facility. Although the inspector determined these types of parameters had been reviewed in the past at a varying periodicity, the licensee confirmed that there was no current expectation or requirement to review these OAC points on a periodic basis. The licensee initiated Problem Investigation Process report (PIP) M00-0780 to document the lack of periodic reviews and oversight of the process by which OAC parameters could be deleted from service (i.e., broad authority existed allowing operations personnel to take points out of service). Planned corrective actions included adding the list of OAC affected points to the reactor operator turnover checklist, performing more frequent reviews on each shift, and developing an approval process system to maintain controls over the deletion of OAC points. Based on a review of the current OAC points deleted from service, no current operability or other problems were identified by the inspector. The licensee’s corrective actions were considered appropriate.

O8 Miscellaneous Operations Issues (92901, 90712)

O8.1 (Closed) LER 50-369/00-001-00: Failure to Meet Temperature Requirements of TS Action Statement When More Than One Centrifugal Charging Pump or Safety Injection Pump Was Capable of Injecting Water into the Reactor Coolant System

The subject of this LER was previously reviewed in detail in Inspection Report (IR) 50-369,370/99-08. Non-Cited Violation (NCV) 50-369/99-08-01, Non-Compliance with TS 3.4.12, Low Temperature Over-Pressure Protection during Unit 1 Restart, was identified concerning the subject non-compliance. Identified licensee corrective actions were appropriate for the issue. This LER is closed.

O8.2 (Closed) LER 50-369/00-002-00: McGuire Units 1 and 2 in a Condition Outside Design Basis of the Plant due to Refueling Water Storage Tank (RWST) Level Channels in a Trip Condition for an Indefinite Period of Time

This issue involved a licensee identified condition that resulted in operation outside the design basis of the plant. Specifically, a deficiency was identified in the current TS action statement concerning one inoperable level channel of the RWST. Currently, the TS allows for the option of placing the affected channel in the trip condition for an indefinite period of time. This provision was applicable to both McGuire units. The current TS was implemented with the Improved Technical Specifications (ITS) in November 1998. Prior to implementation of ITS, the McGuire TS addressed the time an inoperable RWST level channel could be placed in the trip condition.

With no time restrictions incorporated into the ITS action statement, a channel placed in the trip condition cannot be considered the design basis single failure assumed during a design basis accident. A single failure of an additional RWST level channel coincident with a design basis loss-of-coolant accident (LOCA) with another channel in trip as allowed by ITS would result in premature swapover of the low head safety injection pumps to the containment sump. As a result of this premature swapover, both trains of low head safety injection pumps could fail, which could result in a complete loss in recirculation phase post-accident cooling.

The inspector verified that all three channels of RWST level instrumentation of both McGuire units were currently operable. The licensee established administrative controls to limit the time (48 hours based on other similar units' TS) the plant could operate in this condition until the TS could be revised to include additional conservatism to account for uncertainties associated with the modeling assumptions. The licensee's LER review identified past instances since the implementation of ITS where an RWST level channel was placed in the trip condition during channel operational tests and calibrations. However, given that these instances were for pre-planned evolutions, the potential for the affected channels to have been left in the trip condition for a substantial period of time was low. Typical durations for these types of activities was less than 24 hours. The licensee estimated that for an inoperable channel in the trip condition for a year, the increase in core damage frequency (CDF) would be approximately 7.94E-09/reactor year. Based on the above, the inspector did not identify any regulatory issues and considered the risk significance of the subject LER low. Corrective actions for this issue included a planned TS revision to provide a limit on the time an RWST level channel could remain in the trip condition. This LER is closed.

O8.3 (Closed) LER 50-369/00-003-00: Non-Conservatism in Spent Fuel Pool Criticality Calculation

The subject LER was previously described in Section O1.2. Immediate corrective actions included verification of the current and previous spent fuel pool boron concentrations as compared to the TS value. Design reviews were also conducted which verified that the minimum soluble boron concentration for any credible SFP dilution event was greater than 937 ppm. This is greater than the calculated acceptable value of 500 ppm to maintain the SFPs operable. Based on the above, the inspector concluded that the risk significance for an inadvertent criticality was low. The licensee's planned corrective actions included a TS revision to incorporate additional conservatism to account for uncertainties in the licensee's revised SFP modeling calculations and revision to the Updated Final Safety Analysis Report (UFSAR) to specify storage requirements using boron credit methodology. There were no regulatory issues identified during the review of this LER; therefore, it is closed.

O8.4 (Closed) LER 50-369/99-002-(00, 01): Annulus Ventilation (VE) Inoperable In Excess of TS Allowed Time

The subject LER described circumstances where an identified degraded power supply for a VE system pressure switch rendered the Unit 1 train A of VE inoperable between November 30, 1999, and December 10, 1999. The degraded power supply was identified on December 9, 1999, by maintenance technicians performing troubleshooting (per a work request written on November 30, 1999) of a low VE pressure gauge reading. The affected power supply also controlled VE system heating, ventilation, and air conditioning (HVAC) dampers via pressure switches in the 1A VE train. The inability of the pressure switches to control and position the VE system dampers effectively rendered the train inoperable.

Subsequent operability reviews identified that on December 7, 1999, the 1B train of VE had been inoperable for approximately 15 hours when the 1B emergency diesel generator (EDG) was removed from service for scheduled maintenance. This rendered both trains of the VE system technically inoperable. The licensee reviewed the subject 15-hour period and concluded that the 1B VE train had been capable of maintaining post-LOCA Unit 1 annulus pressure within the required range for the entire time the 1A train of VE was inoperable, with the exception of having the 1B EDG available as a power supply. The inspector reviewed scheduled work for December 7, 1999, and did not identify 1B VE train work that would render the train non-functional. The inspector also independently developed a time line of events and verified the circumstances described in the LER.

The licensee determined that the root cause of the extended train A inoperability was a failure of the control room senior reactor operator (CRSRO) to properly communicate his concern with TS VE operability to appropriate personnel. This inadequate communication resulted in the Shift Work Manager (SWM) incorrectly prioritizing the work request, which ultimately prevented the suspect VE gauge indication problem from receiving the proper level of attention. Specifically, the initially identified problem concerning an unexpected gauge reading was due to a failed power supply affecting the gauge and TS related equipment. Consequently, the inoperability of the train A VE system was not determined until December 9, 1999.

Based on design information, system drawings, emergency procedures, and the UFSAR, the inspector assessed the potential safety significance of the event. For the 15-hour period when the 1B train was inoperable due to the lack of an EDG power supply, the licensee calculated that sufficient voltage existed in the degraded power supply to allow the 1A train of VE dampers to realign to a recirculation mode following initial vacuum drawdown of the annulus. However, insufficient voltage was present to allow the 1A train to open the exhaust damper above setpoint when annulus pressure increases from containment leakage and heat up of the annulus air. Based on a qualitative assessment of the conservative assumptions in the LOCA dose model, the short duration of the event (15 hours, which is less than the 24 hour TS action for an inoperable Reactor Building), and low likelihood of a large break LOCA coincident with a loss of offsite power (LOOP), the inspector concluded that there was no appreciable increased risk. A review of the probabilistic risk assessment (PRA) confirmed that the large early release frequency (LERF) would not have been significantly affected by a degraded VE system and there is no effect on CDF. In addition, the major contributions to LERF were identified as containment by-pass sequences due to interfacing system LOCAs. The VE system does not mitigate these types of accident sequences.

The inspector determined that the corrective actions noted in PIP M99-5583 and the associated LER appropriately addressed the root cause of not identifying and resolving the VE train A inoperability. However, this resulted in a violation of TS requirements. Specifically, TS 3.6.10 (annulus ventilation system) requires, in part, that with one VE train inoperable greater than 7 days, the unit must be in Mode 3 within 6 hours and Mode 5 within 36 hours. Contrary to the above, between November 30, 1999, and December 10, 1999, the 1A train of VE was inoperable and Unit 1 remained in Mode 1 operation. In addition, TS 3.6.10 does not address two trains of the same unit's VE system being inoperable simultaneously. As such, the provisions of TS 3.0.3 applied, requiring that the unit be placed in Mode 3 within 7 hours and Mode 4 within 13 hours. Contrary to the above, on December 7, 1999, both trains of the Unit 1 VE system were inoperable for a 15-hour period and Unit 1 remained in Mode 1 operation. This Severity Level IV violation is being treated as an NCV, consistent with Section VII.B.1 of the NRC Enforcement Policy. It is identified as the second example of NCV 50-369/00-02-01: TS Non-Compliance With Respect to the Unit 1 Annulus Ventilation System. This violation is in the licensee's corrective action program as PIP M99-5583. This LER is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (61726, 62707)

The inspector reviewed a variety of maintenance and/or surveillance activities during the inspection period, focusing on testing and maintenance activities that included the following specific items:

- PT/2/A/4350/002B, Revision 039, Diesel Generator 2B Operability Test
- PT/0/A/4150/009, Revision 008, Reactor Coolant System Dilution
- OP/0/A/6100/06, Revision 053, Reactivity Balance Calculation
- TT/0/A/2493/005, Revision 000, B YC Condenser Head Pressure Control Valve Stability Testing
- OP/0/A/6450/011, Revision 051, Control Area Ventilation/Chilled Water System
- OP/1/A/6400/006, Revision 009, Nuclear Service Water System

b. Observations and Findings

The inspector witnessed selected surveillance tests to verify that approved procedures were available and in use, test equipment was calibrated, test prerequisites were met, system restoration was completed, and acceptance criteria were met. In addition, the inspector reviewed or witnessed routine maintenance activities to verify, where applicable, that approved procedures were available and in use, prerequisites were met, equipment restoration was completed, and maintenance results were adequate. The maintenance and surveillance activities were properly approved by operations personnel and were included on the plan of the day. Work associated with risk significant structure, systems, or components was properly evaluated to determine its impact on the plant's risk profile. Appropriate TS action statements and selected licensee commitments were implemented. Applicable Technical Specification surveillance requirements (TSSR) and/or the COLR limits were also satisfied.

c. Conclusions

The inspector concluded that the reviewed routine maintenance and surveillance activities were adequately completed.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Identified Main Steam Leakage

a. Inspection Scope (61726, 62707)

The inspector reviewed the resolution of a degraded material condition which resulted in main steam leakage through main steam line drains located upstream of the main steam isolation valves (MSIVs). PIP M00-0708, related plant drawings, and the UFSAR were reviewed.

b. Observations and Findings

On February 23, 2000, seat leakage was identified on 2SM-18 and 2SM-134 (C MSIV inlet drains) located just upstream of the C MSIV. These normally closed valves, located in series within class B piping, were exhibiting seat leakage into the Unit 2 auxiliary feedwater pump room to the B groundwater sump. This leakage was detected by the actuation of a fire detection system alarm in the auxiliary feedwater (AFW) pump room. Valve 2SM-18 is considered a containment isolation valve; however, as it is installed in a closed system from penetration 2M393, it is not required to be leak tested or leak tight for containment integrity. Corrective actions were to isolate the common header for the drain valves by closing 2SM-124 and 2SM-125, which stopped the leak into the pump room. The inspector reviewed the use of the these alternate, class G, non-seismic qualified valves to maintain compliance with containment penetration TS 3.6.3.C.1. This TS requires that for closed systems with one containment isolation valve inoperable, within 72 hours, the affected penetration flow path be isolated by at least one closed manual valve. For seismic-related failures involving the loss of the class G piping valves 2SM-124 and 2SM-125, the resulting leakage would be less than the worst case dose from a steam generator tube rupture concurrent with a stuck open power operated relief valve, which is the applicable UFSAR Chapter 15 accident. Based on the inspector's review, no problems were identified with the use of the valves to meet the subject TS.

c. Conclusions

Seat leakage was identified on two main steam line drain valves located upstream of the C steam generator main steam isolation valve. The leakage was promptly identified and isolated. Use of alternate valves to maintain compliance with applicable Technical Specification requirements was adequate.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Review of Non-Safety Inputs to Solid State Protection System (SSPS) (37551)

During the inspection period, the inspector reviewed the routing of non-safety inputs to the SSPS within the Unit 1 and 2 turbine buildings, with applicable engineering personnel. The inspector referenced industry events identified in NRC Information Notice (IN) 95-10, Potential For Loss of Automatic Engineered Safety Features Actuation, PIP M95-0265, and the design assumptions for high energy line breaks within the McGuire turbine buildings. The walkdowns of the subject SSPS inputs identified that the routing of the SSPS input cables were generally well protected from one high energy line break affecting more than one train of SSPS inputs. For sampled areas where both trains could be potentially impacted, the licensee provided evidence of appropriate monitoring for erosion of the subject piping. Based on a review of PIP M95-0265 and walkdown of the SSPS inputs, the inspector concluded that the licensee had adequately reviewed the applicable problems detailed in IN 95-10. No findings were identified.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Unresolved Item (URI) 50-369/99-08-02: Review of Past Operability Evaluations for Missing Ice Condenser Coupling Screws

This URI was identified to review the licensee's final past operability determination concerning three Unit 1 ice condenser baskets where the number of missing coupling ring screws exceeded the specified acceptance criteria as identified in Westinghouse Nuclear Safety Advisory Letter 98-012. The NRC staff reviewed the licensee's operability determinations for conditions identified in the subject URI and independently evaluated the conclusions and assumptions identified in PIP M99-04374. This PIP detailed the locations and number of ice condenser basket coupling screws which were identified during the end-of cycle 13 refueling outage (September 17, 1999 - October 31, 1999) as missing or never installed and exceeded the acceptance criteria for missing coupling screws based on vendor information. Based on the reviews, the staff concluded that the identified missing screw locations did not adversely impact the past operability of the McGuire Unit 1 ice condenser. However, based on the identified material condition problem, the inspector concluded that a violation of 10 CFR Part 50, Appendix B occurred. Specifically, Criterion V, requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, and drawings, and shall be accomplished in accordance with the established instructions, procedures, and drawings. Contrary to this, the ice condenser coupling screws on station drawing MC1201.17, sheets 984, 985, or 986, were not installed for ice condenser baskets identified in PIP M99-4374. This Severity Level IV Violation is being treated as a NCV consistent with Section VII.B.1 of the NRC Enforcement Policy. It is identified as NCV 50-369/00-02-02: Ice Condenser Coupling Screws Not Installed in Accordance with Station Drawings. This violation is in the licensee's correction action program as PIP M99-04374. This URI is closed.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 General Comments (71750)

The inspector made frequent tours of the controlled access area and reviewed radiological postings. The inspector observed that workers were adhering to the requirements of wearing protective clothing. The inspector also determined that locked high radiation doors were properly controlled, high radiation and contamination areas were properly posted, and radiological survey maps were updated to accurately reflect radiological conditions in the respective areas.

V. Management Meetings

X1 Exit Meeting Summary

The senior resident inspector presented the inspection results to members of licensee management at the conclusion of the inspection on April 5, 2000. The licensee acknowledged the findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

Barron, B., Vice President, McGuire Nuclear Station
Bradshaw, S., Superintendent, Plant Operations
Byrum, W., Manager, Radiation Protection
Cash, M., Manager, Regulatory Compliance
Dolan, B., Manager, Safety Assurance
Evans W., Security Manager
Geer, T., Manager, Civil/Electrical/Nuclear Systems Engineering
Jamil, D., Station Manager, McGuire Nuclear Station
Patrick, M., Superintendent, Maintenance
Peele, J., Manager, Engineering
Loucks, L., Chemistry Manager
Thomas, K., Superintendent, Work Control
Travis, B., Manager, Mechanical Systems Engineering

INSPECTION PROCEDURES USED

IP 37551: Onsite Engineering
IP 62707: Maintenance Observations
IP 61726: Surveillance Observations
IP 71707: Conduct of Operations
IP 71750: Plant Support
IP 90712: In-Office Review of Written Reports of Non-routine Events
IP 92901: Followup - Operations
IP 92902: Followup - Maintenance
IP 92903: Followup - Engineering

ITEMS OPENED AND CLOSED

Opened

50-369/00-02-01	NCV	TS Non-Compliance With Respect to the Unit 1 Annulus Ventilation System (Section O8.4)
50-369,370/00-02-02	NCV	Ice Condenser Coupling Screws Not Installed in Accordance with Station Drawings (Section E8.1)

Closed

50-369/00-001-00	LER	Failure to Meet Temperature Requirements of TS Action Statement When More Than One Centrifugal Charging Pump or Safety Injection Pump Was Capable of Injecting Water into the Reactor Coolant System (Section O8.1)
50-369/00-002-00	LER	Units 1 and 2 Outside Design Basis of Plant Due to Refueling Water Storage Tank Level Channels in a Trip Condition for an Indefinite Period of Time (Section O8.2)
50-369/00-003-00	LER	Non-Conservatism in Spent Fuel Pool Criticality Calculation (Section O8.3)
50-369/99-002-(00, 01)	LER	Annulus Ventilation Inoperable in Excess of TS Allowed Time (Section O8.4)
50-369,370/99-08-02	URI	Review of Past Operability Evaluations for Missing Ice Condenser Coupling Screws (Section E8.1)

LIST OF ACRONYMS USED

AFW	-	Auxiliary Feedwater
CDF	-	Core Damage Frequency
CFR	-	Code of Federal Regulations
COLR	-	Core Operating Limits Report
CRSRO	-	Control Room Senior Reactor Operator
EDG	-	Emergency Diesel Generator
HVAC	-	Heating, Ventilation, and Air Conditioning
IN	-	Information Notice
IR	-	Inspection Report
ITS	-	Improved Technical Specification
LER	-	Licensee Event Report
LERF	-	Large Early Release Frequency
LOCA	-	Loss-of-Coolant Accident
LOOP	-	Loss of Offsite Power
MSIV	-	Main Steam Isolation Valve
NCV	-	Non-Cited Violation
NRC	-	Nuclear Regulatory Commission

NRR	-	NRC Office of Nuclear Reactor Regulation
OAC	-	Operator Aid Computer
PDR	-	Public Document Room
PIP	-	Problem Investigation Process
ppm	-	Parts per million
PRA	-	Probabilistic Risk Assessment
RWST	-	Refueling Water Storage Tank
SFP	-	Spent Fuel Pool
SPSS	-	Solid State Protection System
SWM	-	Shift Work Manager
TS	-	Technical Specifications
TSSR	-	Technical Specification Surveillance Requirements
UFSAR	-	Updated Final Safety Analysis Report
URI	-	Unresolved Item
VE	-	Annulus Ventilation