U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-289
License No.	DPR-50
Report No.	97-05
Licensee:	GPU Nuclear Corporation
Facility:	Three Mile Island Station, Unit 1
Location:	P.O. Box 480 Middletown, PA 17057
Dates:	April 28, 1997 - June 15, 1997
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EXECUTIVE SUMMARY

Three Mile Island Nuclear Power Station
Report No. 50-289/97-05

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 7 week period of resident inspection and the Regional review of the fuel cask shipment for Unit 1. The results of the engineering core inspection and the motor operated valve program will be documented in separate correspondence.

Plant Operations

- The engineered safeguards actuation system (ESAS) configuration, procedures, and test requirements were consistent between the plant hardware and system documentation to support the design safety function. A positive example of TMI's program to maintain the plant design basis current and up to date, was noted for the ESAS UFSAR enhancements and corrections submitted by the system engineer (Section O2.1).

- The complex ESAS surveillance test procedure verified that the safety system design requirements were satisfied and minimized unnecessary challenges to the emergency equipment. The surveillance tests were coordinated and completed without an impact on plant operation or safety equipment availability (Section O2.1).

Maintenance

- The fuel pin shipping cask activities were very well controlled by maintenance, radiological controls, and engineering personnel. Supervisory oversight was maintained throughout the evolution (Section M1.1).

- The experienced I&C technician applied excellent troubleshooting and calibration techniques throughout the repair and surveillance activities for the RM-G-25 radiation monitor. During the review of the RM-G-25 safety tag application the detector drain valve and inlet isolation valve tags were found to be attached to the wrong valve. The safety impact was minimal because both valves were closed and tagged. The operations personnel immediately corrected the tagged valves and initiated a corrective action process (CAP) form to evaluate the root cause for the self checking error (Section M1.1).

Engineering

- GPU's immediate corrective actions addressed the QCL downgrade deficiencies and were verified by nuclear safety assessment to be effective. QCL activities were stopped until the engineering procedure EP-011, "Methodology for Preparing the Quality Classification List," was revised and training was provided to all personnel involved with the QCL process (Section E3.1).

- EP-011, "Methodology for Preparing the Quality Classification List," was revised to formalize the QCL process and included written detailed standards related to component and program changes. For example, EP-011, section 4.5.1, "Downgrades," was revised to require a written safety determination/safety evaluation when the quality classification of a component is changed from a higher to a lower classification. In addition, the procedure was added to the safety review program described by Technical Specification No. 6.5.1.12 and a safety evaluation was written to document the bases for the revision (Section E3.1).

- GPU Nuclear took action to ensure that QCL deficiencies with a potential to impact plant safety were addressed. For the components with the greatest potential impact on safety (NSR), safety reviews concluded that there was no effect on the operability of those components. Even though component documentation lowered the classification, the component spare parts remained at the original higher classification. This combined with the quality assurance controls employed through maintenance procedures, provided reasonable assurance that repaired components remained operable (Section E3.1).

Plant Support

- In the area of plant support, we found that your preparation, planning, and coordination of a spent fuel shipment was very good. Although appropriate emergency response information was provided in a timely manner in response to a mini-drill scenario involving a postulated accident with a fuel shipment, the shift supervisor did not maintain a constant communication link until all of the needed information was provided to the responder.

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Report Details

Summary of Plant Status

Unit 1 remained at 100% power throughout the inspection period.

I. Operations

O1 Conduct of Operations (71707)[1]

O1.1 General Comments

Using Inspection Procedure 71707, "Plant Operations," the inspectors conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was professional and safety-conscious; specific events and noteworthy observations are detailed in the section below. In particular, the inspectors continue to observe excellent performance and coordination of the quarterly engineered safeguards actuation system (ESAS) surveillance test.

O2 Operational Status of Facilities and Equipment

O2.1 Engineered Safety Feature System Walkdown (71707)

a. Inspection Scope

The inspectors used Inspection Procedure 71707 to walkdown the accessible portions of the engineered safeguards actuation system (ESAS) and associated safety system interconnections. The inspection included a review of the ESAS procedures, Technical Specifications (TSs), updated final safety analysis report (UFSAR) and engineering documentation associated with system changes.

b. Observations and Findings

The inspectors reviewed the UFSAR section 7.1.3, "Engineered Safeguards Actuation System," Technical Specification sections 3.3, "Emergency Core Cooling, Reactor Building Emergency Cooling and Reactor Building Spray Systems, " and 3.4, "Decay Heat Removal," and the associated emergency core cooling system (ECCS) operating procedures. The system configuration, procedures, and test requirements were consistent between the plant hardware and system documentation.

The ESAS monitors plant parameters to detect a reactor coolant system (RCS) leak and initiates the high pressure injection system, low pressure injection system, BS system, Reactor Building (RB) cooling system, and a RB isolation. In addition, the ESAS signal is used to start the emergency diesel generators and control diesel load sequencing.

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[1]Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

The ESAS parameters, procedures and design bases information contained in the UFSAR were up to date. The current UFSAR contained one minor error in Table 7.1-2, "Engineered Safeguards Actuated Devices." The table referenced two building spray (BS) valves BS-V4A/B, sodium thiosulfate suction isolations, that were permanently locked closed. The system engineer had submitted an UFSAR change request on February 14, 1997, to correct the error. Additional ESAS UFSAR enhancements were included with the same change request.

The ESAS surveillance test procedures were written to perform all of the TS test requirements over a three to four day period. The complex alignment, execution, and restoration of multiple safety related components was coordinated between operations, maintenance, engineering, chemistry, and radiation protection departments. As written, the procedures verified the safety system design requirements and also minimized unnecessary challenges to the emergency equipment. The procedure contained detailed cautions and notes at key locations to ensure the performance of critical steps were highlighted for the plant operators. The personnel performance for the ESAS test continues to be excellent. The tests were coordinated and completed without an impact on plant operation or safety equipment availability.

ESAS hardware was installed and maintained as designed. Additional seismic supports were added to enhance the ability of the system to function during a postulated seismic event. Housekeeping was excellent in the ESAS equipment locations throughout the plant.

c. <u>Conclusions</u>

The ESAS configuration, procedures, and test requirements were consistent between the plant hardware and system documentation to support the design safety function. A positive example of TMI's program to maintain the plant design basis current and up to date, was noted for the ESAS UFSAR enhancements and corrections submitted by the system engineer.

The complex ESAS surveillance test procedure verified that the safety system design requirements were satisfied and minimized unnecessary challenges to the emergency equipment. The surveillance tests were coordinated and completed without an impact on plant operation or safety equipment availability.

II. Maintenance

M1 Conduct of Maintenance (62707, 61726)

M1.1 General Comments

a. Inspection Scope

The inspectors observed all or portions of the following maintenance and surveillance work activities:

- Job Order No. 132500, "EG-Y-1A Air Start Compressor Pressure Switch Replacement."
- Job Order No. 134314, "Inspect and Align Solid State Controls for Inverter 1C."
- Special Test Procedure (STP) 1-97-0030, "Fuel Cask Operations for Hot Cell Project."
- Surveillance Procedure 1303-5.1, "RB Emergency Cooling Isolation System Logic Channel/Component Test."
- Surveillance Procedure 1303-5.2, "Emergency Loading Sequence and HPI Logic Channel/Component Test."
- Surveillance Procedure 1303-4.21, "Post Accident Monitoring Channel Test."
- Surveillance Procedure 1302-17.2, "RM-G-24 and 25 Calibration."
- Surveillance Procedure 1303-3.1, "Control Rod Movement."

b. Observations and Findings

On June 4, 1997, the inspector observed activities associated with the shipment of a fuel cask containing six fuel pins. The fuel pins were being shipped to GE Nuclear to perform diagnostic testing to investigate crud buildup and local fuel rod cladding damage. The inspector observed the movement of the shipping cask from the Unit 1 fuel handling building truck bay to the Unit 1 spent fuel pool cask pit. The inspector noted that the shipping cask movement was very well controlled by maintenance, radiological controls, and engineering personnel and supervision in accordance with STP 1-97-0030. Additional details related to the fuel shipment are documented in section R8.1 of this report.

On June 10, 1997, the inspectors reviewed the RM-G-25, "Main Condenser Off-gas Radiation Monitor," troubleshooting and surveillance activities. The surveillance procedure 1303-4.21, "Post Accident Monitoring Channel Test," was performed on the new to ensure the detector responded properly to a high radiation source. The instrumentation and controls (I&C) technician's radiation monitor/detector troubleshooting and calibration techniques were very proficient and experienced. During the review of the RM-G-25 safety tag application, No. 97-0538, an error was noted by the inspector. The tags for the detector drain valve VA-V-33 and the detector inlet isolation valve VA-V-32 were placed on the wrong component. The safety impact was minimal because both valves were closed and tagged. The operations personnel immediately corrected the valve tags and initiated a corrective action process (CAP) form to evaluate the root cause and associated corrective actions related to the error. This violation of the station tagging procedure is of minor significance and is being treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy. **(NCV 50-289/97005-02)**

c. Conclusions

The fuel pin shipping cask activities were very well controlled by maintenance, radiological controls, and engineering personnel. Supervisory oversight was maintained throughout the evolution.

The experienced I&C technician applied excellent troubleshooting and calibration techniques throughout the repair and surveillance activities for the RM-G-25 radiation monitor. During the review of the RM-G-25 safety tag application the detector drain valve and inlet isolation valve tags were found to be attached to the wrong valve. The safety impact was minimal because both valves were closed and tagged. The operations personnel immediately corrected the tagged valves and initiated a corrective action process (CAP) form to evaluate the root cause for the self checking error.

III. Engineering

E3 Engineering Procedures and Documentation (37551, 92903)

E3.1 Equipment Quality Classification Documentation related to CAL No. 97-008 Items 1&2

a. Inspection Scope

The inspectors reviewed the equipment quality classification list (QCL) documentation related to confirmatory action letter (CAL) No. 97-008, Items No. 1 & 2. The CAL commitments included taking immediate actions to preclude additional inappropriate equipment downgrades and to determine the impact of the equipment downgrade program at TMI. The QCL immediate and short term corrective actions were completed by GPU and documented in a letter to the NRC on April 30, 1997.

b. Observations and Findings

The inspectors reviewed the documentation related to the QCL corrective actions. The nuclear safety assessment (NSA) department issued a stop work notice, No. NSA-TMI-97-001, Rev.1, on March 1, 1997, to prevent any additional QCL program downgrades until proper corrective actions were completed. Also, the Maintenance Director issued a memorandum and a list of all down graded components that were in the generation maintenance system (GMS-2) computer system prior to the stop work order. The components in question were temporarily upgraded to nuclear safety related (NSR) until material non-conformance reports (MNCRs) or engineering evaluations were completed to approve the proper classification. NSA also verified that the immediate corrective actions were incorporated and effective. Because of the potential impact on safe plant operation.

GPU engineering evaluated and corrected the QCL deficiencies related to the downgrade of NSR components. A data base review revealed that 1129 components were downgraded to regulatory required (RR) or "Other" (non-safety related). GPU determined that 71 components, of the 1129 total, should have remained at the NSR classification. The 71 components were returned to the NSR status and a complete review of parts used in the maintenance and modification of the components was performed for the downgrade time frame. Of the 71 components, 11 had non-safety grade parts installed.

The QV department initiated MNCRs to evaluate and document the acceptability and operability of the 11 components. The inspectors reviewed the MNCR documentation associated with all 11 components. Each component was entered into the corrective action process (CAP) data base to ensure that the corrective actions were tracked to completion. Three of the components installed in the safety related river water systems were replaced with NSR parts. An emergency feedwater steam control valve, MS-V-13B, was analyzed and found exceptable and will be replaced during a planned outage with and NSR component. Six components related to the main feedwater and diesel generators were scheduled for a formal commercial grade dedication to document acceptability. One component, a diesel building temperature controller, was installed after performance of a commercial grade dedication in 1994. A written engineering evaluation and disposition was included in each MNCR package to provide a bases for each components NSR application. The documentation was detailed, complete and provided a sound bases for equipment operability.

In addition to the NSR downgrades, a number of RR components were downgraded to "Other with quality assurance (QA) processes applied (3834 items)" or "Other without QA applied (1978 items)." All 3834 items that were downgraded from RR to "Other with QA" have had their classification returned to RR. The inspector reviewed a sample of the RR downgraded items in the GMS-2 computer data base. All component records reviewed determined that the components were returned to the RR classification as stated in the April 30th letter. The GPU evaluation of the downgraded components determined that there was no negative impact on past or future operation of plant systems for the following reasons. Although the 3834 items were downgraded, the materials and parts for the downgraded items were not programmatically downgraded and many of the parts which were available for use remained at the RR classification. During the classification downgrading process activities associated with the equipment were maintained within QA scope, and required corrective and preventative maintenance activities were performed using QA program procedures. The procedures which were used required a comparison of replacement parts with existing parts and required post maintenance testing to verify that the equipment would perform its intended function.

GPU evaluated the 1978 items that were downgraded from RR to "Other without QA" to determine and document the proper component classification. The evaluation resulted in 649 items being returned to RR and the rest reclassified as Other. The impact of the downgrade on the 649 items was determined to have no negative impact on past or future operation of plant systems for the following reasons. The materials and parts for the downgraded components were not programmatically downgraded and many remained at the RR classification. The approved maintenance procedures required a comparison of replacement parts with existing parts and post maintenance testing to verify that the equipment would perform its intended function.

Based on these facts, GPU concluded that the operability of the affected components was not impacted during the period of time when they were at the lower classification. The inspector's review of the documentation associated with the RR downgrades did not reveal any operability concerns.

The inspector reviewed the content and implementation of engineering procedure EP-011, "Methodology for Preparing the Quality Classification List." The corporate Engineering Division procedure was revised on April 29, 1997, and training was provided to all personnel involved with the QCL process. The procedure was added to the safety review program described by TS 6.5.1.12 and safety evaluation No. SE-945100-099 documented the bases for the revision. The new procedure has formalized the QCL process and included written detailed standards related to component and program changes. For example, EP-011, section 4.5.1, "Downgrades," was revised to require a written safety determination/safety evaluation when the quality classification of a component is changed from a higher to a lower classification. The definition of regulatory required (RR) components was clarified in the procedure and "Exhibit 9A" provided a list of regulatory references that would result in an RR classification level. The applicability of the procedure was changed to reflect the intent of the program quality controls.

c. <u>Conclusions</u>

GPU's immediate corrective actions addressed the QCL downgrade deficiencies and were verified by NSA to be effective. QCL activities were stopped until the engineering procedure EP-011, "Methodology for Preparing the Quality Classification List," was revised and training was provided to all personnel involved with the QCL process.

EP-011, "Methodology for Preparing the Quality Classification List," was revised to formalize the QCL process and included written detailed standards related to component and program changes. For example, EP-011, section 4.5.1, "Downgrades," was revised to require a written safety determination/safety evaluation when the quality classification of a component is changed from a higher to a lower classification. In addition, the procedure was added to the safety review program described by TS 6.5.1.12 and a safety evaluation was written to document the bases for the revision.

GPU Nuclear took action to ensure that QCL deficiencies with a potential to impact plant safety were addressed. For the NSR components with the greatest potential impact on safety, safety reviews concluded that there was no effect on the operability of those components. Most materials and parts for downgraded components remained at the previous classification. This combined with the quality assurance controls employed through maintenance procedures, provided reasonable assurance that repaired components remained operable. The TMI corrective actions were sufficient to conclude that confirmatory action letter No. 97-008, Items No. 1 & 2, were addressed satisfactorily.

IV. Plant Support

R8 Miscellaneous RP&C Issues (86759, 92904)

R8.1 Fuel Shipment

a. Background and Inspection Scope (86750)

GPU made an arrangement with GE Nuclear-Vallecitos Nuclear Center, in Pleasanton, California, to perform diagnostic testing on specific fuel pins to investigate crud buildup and local fuel rod cladding damage. On June 6, 1997, six fuel pins were shipped from Three Mile Island Nuclear Plant to GE Nuclear - Vallecitos Nuclear Center, in Pleasanton, California. An inspection was performed to determine if adequate precautions and preparations had been taken to safely ship the six spent fuel pins. Information was gathered through direct observations of vehicle loading and shipment preparations; performance of independent radiological surveys; reviews of shipping paperwork; discussions with cognizant personnel; and the conduct of a mini-emergency response information drill.

b. Observations and Findings

The six fuel pins had previously been placed inside a canister that was contained inside a type B shipping cask for transportation. The inspector directly observed the loading of the shipping cask onto the transport trailer; installation of impact limiters; installation of a personnel barrier; and final vehicle transport inspections. These evolutions were well coordinated and showed evidence of effective planning. Only minor problems were encountered during shipment preparations, and licensee staff responded immediately to correct the problems. For example, a stripped bolt hole thread on the transport trailer was repaired to allow the personnel barrier to be installed, and a GPU mechanic reconnected a broken electrical wire to repair the transport trailer running lights. The inspector observed the performance of radiological surveys on the cask and transport vehicle, and performed independent radiation surveys. Radiation and contamination levels were well below the requirements listed in 49 CFR Part 173.441 and Part 173.443, and the inspector's survey results closely matched radiation survey data obtained by the radiological controls staff. Department of Transportation (DOT) Yellow III labels were placed on two sides of the shipping cask, and "radioactive" placards were used on the transport vehicle as required by DOT for highway route controlled quantity (HRCQ) shipments.

The inspector also reviewed the preparation of shipping papers and noted that appropriate information was included in the shipping paperwork package. Examples included a bill of lading; basic description of shipment; radionuclide data; shipper's certification; certificate of compliance for the shipping cask; and a 24-hour emergency response telephone number.

The inspector observed a briefing provided to the truck drivers from the radwaste group supervisor and a representative from GE Nuclear. The shipping coordinator discussed the information contained in the bill of lading, emergency response information, and shipping paperwork. The representative from GE Nuclear discussed trip coordination, planning, communications, and logistics. Briefings were thorough, clear, and conducted at a comfortable pace. Particular emphasis was placed on providing answers to questions, and ensuring that the drivers knew what phone numbers to call in the event of an emergency.

The inspector also conducted a mini-drill while the shipment was in progress to determine if appropriate emergency response information could be obtained in a timely manner by contacting the emergency response telephone number listed on the shipping paperwork. On June 9, 1997, at 8:35 a.m., the inspector called the emergency response number listed on the shipping paperwork and the phone was answered by a shift supervisor in the Unit 1 control room at TMI. The inspector identified himself as an NRC inspector and announced that a drill was being conducted. The inspector stated that the following scenario was "drill-related." A radioactive fuel shipment number RS-97-040-I, that originated at Three Mile Island on June 6, 1997, had been involved in a motor vehicle accident, and the personnel barrier had been knocked off, and the tractor had been damaged. The shift supervisor was asked if it was okay to approach the vehicle and change out the tractor. The shift supervisor stated that he would have to check, located the shipping paperwork, and was not able to provide an immediate answer relative to emergency response information. The shift supervisor confirmed the details of the accident, and stated that he would call back to provide this information. Approximately 12 minutes after the initial call, the shift supervisor called the inspector back and was able to provide information relative to potential hazards and emergency response information. At this time, the inspector asked additional questions such as "If an injured person was in the tractor, would it be OK to approach the vehicle to provide the individual first aid?" and "Would it be necessary to wear dosimetry devices?". The shift supervisor stated that for the scenario described, medical problems took priority over radiological concerns, and that priorities for first aid were higher than priorities for measuring radiation levels.

The inspector concluded that appropriate emergency response information was provided within a timely manner (i.e., within 15 minutes). Information Notice No. 92-62, "Emergency Response Information Requirements for Radioactive Material Shipments," contains a recommendation for the site contact to remain on the telephone line with the emergency responder until all questions are answered relative to the postulated shipment accident. The emergency response information available to the control room staff met the NRC requirements contained in the transportation regulation 49 CFR Part 172.602 and Part 172.604. The station administrative procedure also contained the regulatory required written guidance and was consistent with the transportation regulations.

The inspector interviewed the radwaste shipping coordinator and was informed that procedural guidance allows the individual manning the emergency telephone line to obtain assistance with the phone call by contacting other knowledgeable personnel. However, the inspector noted that since emergency response personnel could have a variety of backgrounds (fire, rescue, police, or bystander), and due to uncertainties in establishing communications, it is important for personnel manning emergency telephone lines to maintain constant communications with emergency response personnel until communications are complete. TMI staff's ability to provide timely emergency response information regarding radioactive material shipments in transit will be reviewed in a future inspection (IFI 50-289/97-05-01).

c. Conclusions:

The preparation, planning, and coordination of a spent fuel shipment was very good. Emergency response information was provided in a timely manner in response to a mini-drill scenario involving an accident with a fuel shipment. An area for improvement was noted related to the constant communication of the control room personnel with emergency responders. This item will be reviewed for a future shipment.

V. Management Meetings

X1 Exit Meeting Summary

At the conclusion of the reporting period, the resident inspector staff conducted an exit meeting with TMI management on June 25, 1997, summarizing Unit 1 inspection activities and findings for this report period. TMI staff comments concerning the issues in this report were documented in the applicable report section. No proprietary information was identified as being included in the report.

X2 Engineering Pre-Decisional Enforcement Conference Summary

On May 22, 1997, a predecisional enforcement conference was held to discuss the events and issues involving multiple apparent violations related to GPUN's response to problems identified in the Engineering area related to the plant design, quality classification list, and environmental qualification programs. The details of the apparent violations are described in Inspection Report Nos. 50-289/96-201, dated March 1, 1997; 50-289/97-01, dated March 20, 1997; and 50-289/97-03, dated May 29, 1997. The open meeting was held between the NRC and GPUN at the NRC Region I Office in King of Prussia, Pennsylvania. The purpose of the meeting was to obtain information to enable the NRC to make an enforcement decision, such as understanding of the facts, root cause(s), missed opportunities to identify the apparent violation sooner, corrective actions, significance of the issues and the need for lasting and effective corrective actions.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Etheridge, Acting Radiological Controls/Occupational Safety Director
J. Grisewood, Emergency Preparedness Manager
D. Hosking, NSA Manager
*J. Langenbach, Vice President and Director
R. Maag, Plant Maintenance Director
L. Noll, Plant Operations Director
M. Ross, Director, Operations and Maintenance
J. Schork, Regulatory Affairs
G. Skillman, Technical Functions Site Director
P. Walsh, Engineering Director
J. Wetmore, Manager, Regulatory Affairs

* senior licensee manager present at exit meeting on June 20, 1997.

NRC

B. Buckley, TMI Project Manager, NRR
J. Nick, Reactor Engineer (Acting), DRP

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92903:	Followup - Engineering
IP 92904:	Followup - Plant Support

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-289/97-05-01 (IFI); emergency response information regarding radioactive material shipments.

Closed

NONE

Updated

CAL No. 97-008, Items No. 1 & 2; equipment quality classification list (QCL)

LIST OF ACRONYMS USED

AB	Auxiliary Building
ALARA	As low As Reasonably Achievable
ASME	American Society of Mechanical Engineers
CDF	Core Damage Frequency
CR	Control Room
CFR	Code of Federal Regulations
DBD	Design Basis Documents
ECCS	Emergency Core Cooling System
EPIP	Emergency Plan and Implementing Procedure
ESF	Engineered Safety Feature
ENMCF	Event or Near Miss Capture Form
HRA	High Radiation Area
IFI	Inspection Followup Item
IPE	Individual Plant Evaluation
IR	Inspection Report
ISI	Inservice Inspection
IST	Inservice Testing Program
JO	Job Order
LCO	Limiting Condition of Operation
LER	Licensee Event Report
MNCR	Material Nonconformance Report
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NSA	Nuclear Safety Assessment
PCR	Procedure Change Request
PPB	Part per Billion
PPM	Part per Million
PRA	Probabilistic Risk Assessment
PRG	Plant Review Group
QV	Quality Verification
RCA	Radiological Control Area
RCS	Reactor Coolant System
RP	Radiation Protection
RWP	Radiation Work Permits
SALP	Systematic Assessment of Licensee Performance
SF	Shift Foreman
SRO	Senior Reactor Operator
SS	Shift Supervisor
TI	Temporary Instruction
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VIO	Violation