U. S. NUCLEAR REGULATORY COMMISSION

REGION I

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Report No.	96-06
Licensee:	GPU Nuclear Corporation
Facility:	Three Mile Island Station, Unit 1
Location:	P.O. Box 480 Middletown, PA 17057
Dates:	August 4, 1996 - September 28, 1996
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EXECUTIVE SUMMARY

Three Mile Island Nuclear Power Station
Report No. 50-289/96-06

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 8 week period of resident inspection for unit 1.

Plant Operations

An improvement was noted in the area of operability determinations. Senior reactor operators (SROs) identified and properly documented a degradation of Auxiliary Building and Fuel Handling Building Ventilation (ABFHV) system flow. The ABFHV flow dropped below the Technical Specification minimum value during the routine purge of the Reactor Building. The shift SRO entered the applicable TS Limiting Condition for Operation (LCO) and initiated efforts to troubleshoot the problem (Section O1.3).

Plant operations performed and implemented multiple detailed on-line safety risk assessments for planned safety related equipment outages. The applicable system Technical Specification limiting conditions for operation were entered and exited correctly for the equipment outage times (Section O1.3).

An area for improvement was noted for the senior reactor operator's (SROs) review and understanding of a maintenance work activity and the associated impact on plant operation. A Shift SRO authorized an emergency feedwater (EFW) controller pushbutton module removal without a complete understanding of the module's impact on the controller operation. The diligence of the system engineer ultimately resulted in the correct understanding of the EFW module operation (Section O1.2).

Licensed operators were very knowledgeable about the decay heat pump generic minimum flow concerns. The operators' had an excellent working knowledge of the small break loss of coolant accident (SBLOCA) emergency operating procedure (EOP). The SBLOCA EOP provided clear and concise written directions for the operators (Section E1.1).

Maintenance

The maintenance and surveillance test activities observed during this inspection were performed satisfactorily and demonstrated that the associated systems could perform their design safety functions (Section M1.1).

The weekly component vibration monitoring program, not required by the IST program, was an example of an excellent maintenance initiative to detect and correct safety related equipment problems before the component fails or becomes inoperable. The mechanical maintenance workers' excellent attention to detail resulted in the replacement of the pump without impacting the other safety related equipment in the river water screenhouse (Section M1.1).

Engineering

The TMI engineering response to the Crystal River decay heat pump minimum flow safety issue was comprehensive, thorough, and demonstrated management's commitment and perseverance to resolve the generic safety issue (Section E1.1).

Engineering management promptly addressed a potential design concern related to the safety related decay river cooling water system. The immediate and long term system operability concerns were addressed by a detailed plant review group evaluation and subsequent engineering reviews (Section E1.2).

Plant Support

TMI identified a repeat problem related to the failure to control a posted high radiation barrier. This issue is considered a violation of the Unit 1 Technical Specifications. The immediate corrective actions were comprehensive (Section R2.1).

The site investigation team performed a detailed, thorough, and timely review of the high radiation barrier incident. Plant Management support was meaningful and focused significant resources on the incident to resolve the recent repeat work problems (Section R2.1).

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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. Improved Technical Specification (TS) operability determinations and documentation by the shift senior reactor operators (SROs), were noted for the emergency feedwater (EFW) and Auxiliary Building ventilation work activities.

The operations department performed and implemented multiple detailed on-line safety risk assessments for planned safety related equipment outages. The applicable system Technical Specification limiting conditions for operation were entered and exited correctly for the equipment outage times.

O1.2 Operability of Emergency Feedwater Equipment/Systems

a. Inspection Scope

An area for improvement was noted during the SRO review of the EFW control module, EF-V-30A, work package. The flow control module was removed from the main control panel to allow the instrumentation and control (I&C) technicians to perform a routine cleaning and inspection of the controller pushbutton and spring assembly. In the past two years, the controllers had experienced intermittent problems when the operators swapped the controllers from the automatic to manual mode of operation.

b. Observations and Findings

The shift SRO authorized the work activity without a complete understanding of the operational impact of the I&C work in relation to the EFW module. It was not clear to the SRO if the controller would swap from the normal automatic mode to a manual mode of operation when the I&C technician removed the pushbutton assembly. The SRO contacted the system engineer to determine the correct controller operation with the pushbutton assembly removed from the controller.

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[1] Topical headings such as 01, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

After a review of the EFW logic prints and associated technical documents, the system engineer informed the shift SRO that the EFW controller would remain in the automatic mode if the pushbutton assembly was removed. However, to verify the correct EFW response the system engineer contacted Foxboro, the flow controller vendor. The vendor informed the engineer that the EFW flow controller could switch from automatic to the manual control mode when removing and installing the pushbutton assembly. The updated information was provided to the control room personnel. The diligence of the system engineer ultimately resulted in the correct understanding of the EFW flow control module operation.

c. Conclusions

An area for improvement was noted for the senior reactor operator's review and understanding of a maintenance work activity and the associated impact on plant operation. One example was noted when a shift SRO authorized an emergency feedwater controller pushbutton module removal without a complete understanding of the module's impact on the controller operation. The diligence of the system engineer ultimately resulted in the correct understanding of the EFW module operation

O1.3 Operability of Auxiliary Building Ventilation Equipment/Systems

a. Inspection Scope

An improvement was noted in the area of the SROs' Technical Specification equipment operability determinations. The SROs identified and properly documented a degradation of Auxiliary Building and Fuel Handling Building Ventilation (ABFHV) system flow. The ABFHV flow dropped below the TS minimum value during the routine purge of the Reactor Building.

b. Observations and Findings

On August 7, 1996, control room operators' responded to a plant computer ABFHBV low flow alarm. The ventilation low flow alarm was received after the initiation of the Reactor Building (RB) purge system in preparation for a routine tour and maintenance activities. The ABFHBV exhaust low flow alarm was received when total flow dropped to 99,160 cubic feet per minute (CFM). The shift SROs declared the system inoperable and entered TS # 3.15.3.3b. An event or near miss capture form (ENMCF) was initiated to document the problem and initiate a root cause evaluation for the event.

The inspectors reviewed the ABFHBV control room flow recorder, SRO log, RB purge procedure and associated TSs. The shift SRO entered the applicable TS Limiting Condition for Operation (LCO) when flow dropped below 100,580 CFM. The SRO log included the same ABFHBV out of service time when compared to the flow recorder strip chart readings. The SRO promptly documented the abnormal occurrence with an ENMCF to ensure the problem was evaluated to determine the reason for the flow change. The ventilation flow returned to normal after the RB purge system was secured.

c. Conclusion

An improvement was noted in the area of operability determinations. SROs identified and properly documented a degradation of the ABFHBV system flow. The flow dropped below the Technical Specification minimum value during the routine purge of the Reactor Building. The shift SRO entered the applicable TS Limiting Condition for Operation (LCO) and initiated efforts to troubleshoot the problem.

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. 121455, "Clean and Inspect EF-V-30A Hand Auto Controller."
- Job Order No. 126097, "Troubleshoot the Main Condenser Offgas Radiation Monitor RM-A-15."
- Job Order No. 123250, "Lubricate the Feedwater Control Valves FW-V-17A/B."
- Job Order Nos. 126111 and 126112, "Nuclear River Water Pump NR-P-1C Pump High Vibration and Overhaul."
- Electrical Maintenance Procedure E-1, "Vibration Monitoring for the River Water Pumps."
- Surveillance Procedure 1303-3.1, "Control Rod Movement."
- Surveillance Procedure 1303-5.1, "Reactor Building Emergency Cooling and Isolation System Logic Channel and Component Test."

b. Observations and Findings

The electrical maintenance routine vibration monitoring of safety related equipment resulted in the early detection of a problem related to the nuclear river (NR) water pump, NR-P-1C. The component vibration readings, not required by the inservice test (IST) program, were an example of an excellent initiative by the maintenance department to detect and correct equipment problems before the component fails or becomes inoperable.

The inspector observed a significant portion of the mechanical maintenance pump repair activities. The pump shaft and impeller were removed and replaced with new parts. The location of the pump resulted in very limited work space for the maintenance mechanics to remove the long pump column and shaft sections. The workers excellent attention to detail resulted in the replacement of the pump without impacting the other safety related equipment in the river water screenhouse.

The emergent pump work was prioritized appropriately based on the significance of the safety related cooling water pump with respect to the previously planned work activities. The plant risk associated with the out of service pump was low due to the fact that the degraded pump was not required to be operable by Technical Specifications.

c. Conclusions

The maintenance and surveillance test activities observed during this inspection were performed satisfactorily and demonstrated that the associated systems could perform their design safety functions.

The extra component vibration readings, not required by the IST program, were an example of an excellent initiative by the maintenance department to detect and correct equipment problems before the component fails or becomes inoperable. The mechanical maintenance workers' excellent attention to detail resulted in the replacement of the pump without impacting the other safety related equipment in the river water screenhouse.

III. Engineering

E1 Conduct of Engineering (37551)

E1.1 TMI Response to the Crystal River Decay Heat Pump Minimum Flow Issue (37551)

a. Inspection Scope

A generic issue was raised at the Crystal River (CR) nuclear power plant related to the decay heat (DH) pump design minimum flow capabilities. Based on the most recent concerns TMI and CR have initiated a DHP minimum flow test with at an independent test facility. The spare TMI DH pump was selected to perform a simulated minimum flow test that would be representative of the post small break loss of coolant accident (SBLOCA) conditions. As of October 21, 1996, the pump had run for twelve days without any unexpected problems.

b. Observations and Findings

TMI's two low pressure injection pumps are manufactured by Worthington and are similar to the CR design. A difference is the minimum flow: at TMI the flow is 125 gallons per minute (GPM) compared to the 80 GPM at CR. The pumps are single stage centrifugal pumps rated at 3000 GPM at 151 pounds per square inch (psig). According to the TMI updated final safety analysis (UFSAR) Chapter 6.1-4, the pumps can operate indefinitely at

shutoff head with the 125 GPM minimum flowrate. Also, in response to Bulletin No. 88-04, TMI noted that the operation of the pumps in the low flow mode would last no more than six hours under actual ECCS operation. TMI engineering is currently involved with this issue through the Babcock and Wilcox owners group and have attended the recent meetings in NRC Headquarters about the Bulletin No. 88-04 issue.

The inspectors reviewed the decay heat system section of the UFSAR, emergency operating procedures (EOPs), and the plant review group (PRG) evaluation of the issue. The TMI DH pumps start on low reactor coolant system (RCS) pressure at 1600 psig and again at 500 psig, and a RB pressure of 4 psig. The TMI EOPs have the operators secure the DH pumps if they started at 1600 psig RCS pressure OR 4 psig RB pressure and the event is a SBLOCA that results in sufficient high pressure injection (HPI) flow and a slow drop in RCS pressure. The pumps would receive an automatic signal to start when RCS pressure dropped to 500 psig and begin injecting at approximately 250 psig.

All control room operators were familiar with the CR issue and understood how the TMI DH pumps were designed. The operators' were very knowledgeable about the DH pump automatic start signals and the expected system response to a SBLOCA. They had an excellent working knowledge of the SBLOCA EOPs and understood the operational limitations for the DH pumps. The EOPs provided clear direction for the operators about the DH pump starting and stopping requirements.

The PRG evaluation of the issue concluded that the DH pumps were operable. The determination was based upon the past operating history of the pumps and the fact that the TMI pumps have been maintained in very good condition. The TMI pumps do have an intermittent vibration condition that Engineering is trying to resolve. The PRG also proposed four alternative flowpaths to maximize the DH system flow if the plant experienced a SBLOCA and the RCS suction path was unavailable for long term decay heat removal operation. Management's decision to perform the pump minimum flow test showed their willingness to completely resolve this potential safety issue.

c. Conclusions

The TMI engineering response to the Crystal River decay heat pump minimum flow concern was comprehensive, thorough, and demonstrated management's commitment and perseverance to resolve the generic safety issue.

Licensed operators were very knowledgeable about the decay heat pump generic minimum flow concerns. The operators' had an excellent working knowledge of the small break loss of coolant accident emergency operating procedure. The SBLOCA EOP provided clear and concise written directions for the operators.

E1.2 Operability of the Decay River Water System (37551)

A potential safety concern was raised by engineering related to the decay river (DR) cooling water flow measurement. The decay heat river water flow data could be unconservative due to the location of the annubar flow instrument near a pipe elbow. If the DR system flow indicated higher than actual flow, then the measured flow could be

less than the Technical Specification requirement. Increasing flow above current levels raises issues with emergency diesel loading. Engineering management recognized the potential safety significance of the concern and convened a plant review group (PRG) meeting to evaluate the system operability and if the issue met the reportability criteria in administrative procedure AP-1044. A material nonconformance report (MNCR) was submitted to document the potential safety concern and track the recommended corrective actions.

The PRG evaluated the issue on August 20, 1996, and determined that a concern may exist under the worst case conditions assumed in the updated final safety analysis report (UFSAR). The "as found" DR system data that was presented to the PRG showed that the heat exchanger fouling, strainer differential pressure (DP), river water level and river water temperature did not exceed the design limits. Considering the above, the PRG determined that the decay river system was operable and the issue was not reportable. The Shift Engineer (SE) was instructed to monitor and log the DR plant conditions to assure that the system does not approach the safety analysis assumptions.

The inspectors verified that the SEs understood the DR safety concern and that the DR data was properly monitored and logged. In addition, the associated DR parameter computer alarm setpoints were adjusted to alert the control room personnel before the system deign limits were exceeded. The PRG meeting conducted a detailed review of the potential safety concern and recommended appropriate actions to ensure the DR system operability would be considered if a key design parameter exceeds a value used for the safety analysis calculations. The PRG also assigned engineering an action item to determine the proper flow alignment during the routine system flow test.

In summary, engineering management promptly addressed a potential design concern related to the safety related decay river cooling water system. The immediate and long term system operability concerns were addressed by a detailed plant review group and engineering evaluation.

IV. Plant Support

R2 Status of RP&C Facilities and Equipment

R2.1 (Opened VIO, 50-289/96-06-01) Loss of Controls For a High Radiation Area (92904)

a. Inspection Scope

The inspectors reviewed a licensee-identified incident that occurred on August 7, 1996, involving a barrier to a posted high radiation area (HRA) in the Auxiliary Building (AB) to determine the effectiveness of the licensee's root cause investigation and proposed corrective actions. Postings on the barrier were appropriate and cautioned workers that the barrier was required by TS. Actual dose rates in the area varied from 300 to 500 millirem per hour on contact and 90 mrem at 30 centimeters. The barrier was estimated out of position for approximately three hours. The highest radiation dose received by a plant worker was approximately 6 millirem on the day of the moved barrier.

b. Observations and Findings

The HRA and contamination postings around the AB 'B' emergency safeguards vault area were changed on dayshift to support the plant preservation activities. The activities included the floor scabbling and painting work in the 'B' decay heat removal and building spray vault areas. The radiological controls technician requested that the contaminated area and high radiation area postings be extended to the boundary of the entire area to be scabbled. The expanded boundaries provided a more convenient working area for the preservation activities. The group radiological control supervisor (GRCS) approved the posting change in the 'B' emergency safeguards vault area. After completion of the floor scabbling work activities at approximately 2:30 p.m., the preservation crew left for the day with the high radiation area gate in the proper position.

At approximately 5:00 p.m. a second dayshift GRCS inspected the expanded postings surrounding the 'B' emergency safeguards vault area. The GRCS specifically remembered looking at the two high radiation area swing gates around the work area access points and determined that the posting and gates were properly positioned. At approximately 8:00 p.m., the high radiation area swing gate at the west access point step off pad was found to be moved 180 degrees and did not provide a barrier to the area, as required by the TMI Technical Specification 6.12.1.a. An Auxiliary Operator (AO) noticed the gate open and informed the in plant shift foreman (SF). The SF immediately notified the duty GRCS of the situation. The GRCS and SF photographed the HRA gate in the as found condition, the gate was returned to its proper position, and a second photograph was taken to capture the HRA barrier condition. TMI management was informed of the incident, a radiation survey was performed of the AB area, and an event near miss and capture form was initiated to document the problem.

On August 8, 1996, TMI initiated an investigation to determine the cause of the incident. The investigative team consisted of a representative from nuclear safety assessment (NSA), security, operations, maintenance, and radiological controls departments. Based on the GRCS verification and AO recognition of the HRA barrier, the Team determined that the high radiation barrier was propped open between 5:00 p.m. and 8:00 p.m. on August 7th. Computer printouts were obtained that listed all of the personnel signed on an RWP (18 workers) and all of the personnel that key carded through the AB vital area doors (43 workers) between the hours of 5:00 p.m. and 8:00 p.m. The investigation team conducted individual interviews with the personnel that accessed the AB. In addition, plant walkdowns of the 'B' emergency safeguards vault area were performed with the GRCS that inspected the area at 5:00 p.m. and the AO and SF that found the gate propped open at 8:00 p.m. Nobody that was in the AB on August 7th admitted to moving or knowing who moved the HRA gate during the three hour time period.

The licensee conducted an investigation and critique of this event and determined that one root cause that had contributed to this event was a lack of attention to detail by personnel working in the room. Since the time of this event, the licensee has further investigated and analyzed the failures and implemented some corrective actions. These actions included providing a review of the event to all operations personnel, reviewing the corrective actions taken as a result of previous similar incidents, evaluating all HRA postings and barriers for "user friendliness", and coordination of feedback from employees

on the condition and ease of use for HRA barriers.

The inspectors interviewed plant workers independently to determine the facts about the incident. The inspectors determined that the employees understood their responsibilities for ensuring that radiological controls and barriers were in place when they entered and exited a radiological area. Similar to the GPUN investigation, nobody admitted that they moved or knew who moved the HRA gate on August 7, 1996. It appears that there were three probable reasons for the moved barrier: 1) the barrier made it difficult to remove protective clothing at the contaminated and HRA boundary; 2) a plant worker moved the barrier to transport equipment into or out of the contaminated/HRA; and 3) an unknown person intentionally opened the HRA swing gate.

The inspectors also reviewed the maintenance activities performed in the AB on August 7th. The maintenance tasks that were performed in the AB did not require the workers to pass through the HRA boundary on the 281 foot elevation. A review of the personnel exposure on the day of the moved the barrier revealed that the maximum individual dose was 6 millirem.

The site investigation team performed a detailed and thorough review of the incident. The Team was assembled promptly after the facts about the issue were known. Management recognized the significance of the repeat HRA barrier problems and assigned the most knowledgeable personnel onsite to perform the root cause investigation. The investigation was initiated immediately after the Team was formed in an attempt to increase the probability that the person and reason for the moved barrier would be determined. As part of the interview process, personnel were asked for suggestions and recommendations to improve the TMI radiological controls program with respect to high radiation area posting and control. The following is a summary of the comments:

- The vast majority of suggestions by both radiological controls and non-radiological controls personnel were to eliminate where possible, or reduce to a minimum, overly conservative high radiation area postings within TMI. The incident described in the event capture form involved a high radiation area posting that was in a 2 mr/hr radiation field.

- Use of "turnstiles", instead of swing gates, for high radiation area barriers would ensure that a physical barricade is in place at all times.

- Installation of devices similar to limit switches on the swing gates that would cause a light or buzzer to energize when the gate was open.

- Placement of video cameras at high radiation area gate locations.

- Increase TMI worker sensitivity to company and regulatory concerns with repeated high radiation area barrier violations, through short training discussions, videotapes, or shop meetings.

c. Conclusions

The immediate corrective actions were comprehensive. Because of the repeat problems related to the control of high radiation barriers, this latest example is considered a violation of the TMI Unit 1 TS 6.12.1.a. **(VIO 50-289/96-06-01).**

The site investigation team performed a detailed and thorough review of the incident. The team was assembled promptly after the facts about the issue were known. Plant Management support was meaningful and focused significant resources on the incident to resolve the recent repeat work problems.

P3.1 Emergency Plan Procedures and Documentation (71750)

The Region I emergency preparedness inspector performed an in-office review of the revisions to the TMI emergency plan implementing procedures. Based on your determinations that the changes did not decrease the overall effectiveness of your emergency plan, and that it continues to meet the standards of 10 CFR 50.47(b) and the requirements of Appendix E to Part 50, NRC approval is not required. Our initial review of these changes indicates them to be in accordance with 10 CFR 50.54(q). Implementation of these changes will be subject to inspection to confirm that they have not decreased the overall effectiveness of your emergency plan. A list of the specific revisions that were reviewed is included below. The inspector concluded that the revisions did not reduce the effectiveness of the E-Plan and were acceptable.

EMERGENCY PLAN AND IMPLEMENTING PROCEDURES REVIEWED

Document	Document Title	Revision
EPIP-TMI-.01	Emergency Classification and Basis	4
EPIP-TMI-.07	Activation of the RAC	2
EPIP-TMI-.16	Contaminated Injuries	4
EPIP-TMI-.27	Emergency Operations Facility	7
EPIP-TMI-.28	Activation of the TSC	6
TEP-ADM-1300.02	Emergency Preparedness Training	1
TEP-ADM-1300.04	Administration of the TMI Initial Response and Emergency Support Organization Duty Roster	1

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 October 15, 1996, 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.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

J. Knubel, Vice President, TMI
*M. Ross, Director, Operations and Maintenance
L. Noll, Plant Operations Director
R. Maag, Plant Maintenance Director
D. Etheridge, Radiological Controls/Occupational Safety Director
J. Schork, Regulatory Affairs
J. Wetmore, Manager, Regulatory Affairs
D. Hosking, NSA Manager
G. Skillman, Technical Functions Site Director
P. Walsh, Engineering Director
R. Hess, Training Manager
* senior licensee manager present at exit meeting on October 15, 1996.

NRC

J. Norris, TMI Project Manager, NRR

INSPECTION PROCEDURES USED

IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 37551:	Onsite Engineering
IP 71750:	Plant Support Activities
IP 92904:	Followup - Plant Support

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-289/96-06-01, "Loss of Controls For a High Radiation Area" (VIO).

Closed

None

Updated

None

LIST OF ACRONYMS USED

AB	Auxiliary Building
AEOF	Annex to the Emergency Operations Facility
ALARA	As low As Reasonably Achievable
ASME	American Society of Mechanical Engineers
CDF	Core Damage Frequency
CEDE	Committed Effective Dose Equivalent
CR	Control Room
CFR	Code of Federal Regulations
DBD	Design Basis Documents
ECCS	Emergency Core Cooling System
ED	Emergency Director
EDG	Emergency Diesel Generator
EFW	Emergency Feedwater
EOF	Emergency Operations Facility
ENMCF	Event or Near Miss Capture Form
EPIP	Emergency Plan and Implementing Procedure
ESF	Engineered Safety Feature
HEPA	High Efficiency Particulate
HRA	High Radiation Area
IFI	Inspection Followup Item
IPE	Individual Plant Evaluation
IR	Inspection Report
IST	Inservice Testing Program
JO	Job Order
JPM	Job Performance Measure
LCO	Limiting Condition of Operation
LER	Licensee Event Report
MNCR	Material Nonconformance Report
MSA	Mine Safety Appliance
NCV	Non-Cited Violation
NI	Nuclear Instrument
NRC	Nuclear Regulatory Commission
NSA	Nuclear Safety Assessment
NVLAP	Nuclear Voluntary Laboratory Accreditation Program
ODCM	Offsite Dose Calculation Manual
OSC	Operations Support Center
PAS	Post Accident Sample
PCR	Procedure Change Request
PPB	Part per Billion
PPM	Part per Million
PRA	Probabilistic Risk Assessment
PRG	Plant Review Group
QV	Quality Verification
RAC	Radiological Assessment Coordinator
RCA	Radiological Control Area
RCS	Reactor Coolant System

RP	Radiation Protection
RSP	Remote Shutdown Panel
RWP	Radiation Work Permits
SALP	Systematic Assessment of Licensee Performance
SF	Shift Foreman
SRO	Senior Reactor Operator
SS	Shift Supervisor
TI	Temporary Instruction
TLD	Thermoluminescent Dosimeter
TS	Technical Specification
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VIO	Violation