Three Mile Island 1
2Q/2008 Plant Inspection Findings

Initiating Events

Mitigating Systems

Significance: G Apr 18, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Stage Equipment Required by Abnormal Operating Procedures
The inspectors identified a NCV of Technical Specification (TS) 6.8.1, which requires that written procedures be implemented as recommended in Appendix A of Regulatory Guide (RG) 1.33, including abnormal operating procedures (AOPs) for loss of service water. Specifically, the AOP for loss of river water was inadequately implemented when equipment required was not staged to support the AOP implementation.

The finding is more than minor because it is associated with the procedure quality attribute of the Mitigating Systems Cornerstone, and the associated cornerstone objective of ensuring the reliability of systems (and personnel) that respond to initiating events to prevent undesirable consequences. Specifically, this finding reduced the reliability of the operators to complete the AOP. This finding was of very low safety significance (Green) because the finding is not a design or qualification deficiency, does not represent a loss of safety function, and does not screen as potentially risk significant due to external hazards. Although the operators would be delayed without the staged hoses, the inspectors concluded that the alternative cooling safety function could be provided to the Nuclear Services Closed Cooling Water (NSCCW) system within the time limit specified by AmerGen's calculations.

The finding has a cross-cutting aspect related to the area of PI&R, corrective action program component, in that, AmerGen identified that the hoses were missing in January 2008, and did not implement CAs to replace the hoses required by the AOP until identified by the inspectors. [P1.(d)] (Section 4OA2.a.3.a)

Inspection Report# : 2008006 *(pdf)*

Significance: G Apr 18, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Include Increased EDG Fuel Oil Consumption Into Design Basis Calculations
The inspectors identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Specifically, AmerGen did not ensure that fuel consumption calculations included the additional fuel needed for allowable emergency diesel generator (EDG) frequency variations of up to 61 Hertz (Hz). The increased fuel consumption was not accurately translated into the TS used to verify operability of the EDGs.

This finding is considered to be more than minor because it is associated with the design control attribute of the Mitigating Systems Cornerstone and the associated cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. This finding was of very low safety significance (Green) because the finding is not a design or qualification deficiency, does not represent a loss of safety function, and does not screen as potentially risk significant due to external hazards.

The issue has a cross-cutting aspect related to the area of PI&R, corrective action program component, in that, AmerGen did not thoroughly evaluate the extent of condition for a previous NRC NCV (reference IR 581933) regarding inadequate design control of EDG loading calculations. Specifically, the cause of the problem, not adequately considering the effect on EDG loading due to operating at the maximum frequency allowed by station procedures, was not resolved for other EDG parameters, such as EDG fuel oil consumption. [P1.(c)] (Section

4OA2.a.3.b)

Inspection Report# : 2008006 *(pdf)*

Significance: G Apr 18, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Meet ASME OM Code Data Collection Requirement for Comprehensive IST
The inspectors identified a NCV of TS 4.2.2 for the failure to implement applicable American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code requirements for comprehensive in-service testing (IST) of the ‘A’ and ‘B’ decay heat removal (DH) pumps. Specifically, AmerGen used differential pressure gauges that did not meet the data collection requirements for instrument accuracy.

This finding is more than minor because it is similar to IMC 0612, Appendix E, example
2C, in that, the issue was repetitive (2005 and 2007 comprehensive tests). Additionally, this finding is associated with the Equipment Performance Attribute of the Mitigating Systems Cornerstone and the associated cornerstone objective of ensuring the reliability of systems that respond to initiating events to prevent undesirable consequences. This finding was of very low safety significance because it involved a qualification deficiency that was confirmed not to result in a loss of operability.

This finding has a cross-cutting aspect in the area of PI&R, corrective actions program component, because AmerGen personnel identified the issue in 2005, but did not take appropriate CAs in a timely manner prior to testing the pumps in 2005 and 2007. [P.1(d)] (Section 4OA2.a.3.c)

Inspection Report# : 2008006 *(pdf)*

Significance: G Apr 18, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Require Emergency Lighting Units (ELUs)
The inspectors identified a NCV of License Condition 2.c(4) and 10 CFR 50, Appendix R, Section III.J, which require that emergency lighting units (ELUs) with at least an eight-hour battery power supply be provided in areas needed for operation of safe shutdown (SSD) equipment and in access and egress routes thereto. Specifically, Fire Hazards Analysis Report (FHAR) Attachment 3-7 specifies a post fire safe shutdown (SSD) action for operators to manually operate valve IC-V-4 within four hours for a fire in fire zone AB-FZ-9, and ELUs were not provided at valve IC-V-4 and portions of the adjacent access and egress routes.

The finding is more than minor because it was associated with the Mitigating Systems Cornerstone attribute of protection against external factors (i.e. fire) and affects the cornerstone objective of ensuring reliability and capability of systems that respond to initiating events. Specifically, the finding adversely affected to some degree the ability to carry out local operator actions required to achieve and maintain a SSD condition following a severe fire. This finding was of very low significance because it involved a low degradation of SSD capability. The conclusion of low degradation was based on the fact that the procedure step in question has a four hour completion time per FHAR Attachment 3-7.

The finding has a cross-cutting aspect in the area of PI&R, Self and Independent Assessments Component, because AmerGen did not take the appropriate corrective actions to address this issue commensurate with its safety significance. Specifically, as part of an extent of condition review for missing ELUs identified by a fire safe shutdown self assessment conducted in 2003, AmerGen identifed that emergency lighting was needed at valve, IC-V-4, to meet the requirements of 10 CFR 50, Appendix R, but has not evaluated and corrected the issue in a timely manner. [P.3(c)] (Section 4OA2.c.3)

Inspection Report# : 2008006 *(pdf)*

Significance: SL-IV Dec 31, 2007
Identified By: NRC

Item Type: NCV NonCited Violation
Deficient Control of Plant Staff Overtime
The inspectors identified a Severity Level IV NCV of TS 6.8.1.j for not properly implementing and maintaining procedures for controlling plant staff work hours of personnel performing safety-related activities. Procedure LS-AA-119, Overtime Controls, was deficient in that it permitted the plant manager to authorize work-hour deviations for routine refueling outage activities. Consequently, the plant manager authorized over 700 personnel to work greater than 72 hours and up to 84 hours per 7-day period for routine outage support activities during the TMI refueling outage (1R17), which exceeded the TS requirements. The affected workers included reactor operators, senior reactor operators, auxiliary operators, health physicists, key maintenance personnel, emergency response organization members, reactor engineers supporting reactivity manipulations and fuel handling, and engineering and professional personnel performing safety-related work. The licensee entered this issue into their corrective action program (IR 713257).

Changing implementing procedure, LS-AA-119, so that it no longer complies with the facility technical specification for controlling the plant staff overtime has the potential to impact the NRC's ability to perform its regulatory function. The violation affected the Mitigating Systems cornerstone and is more than minor because, if left uncorrected, the excessive work hours would increase the likelihood of human errors during refueling outage activities and response to plant events. This violation is characterized as a Severity Level IV in accordance with the NRC Enforcement Policy. This issue has a cross-cutting aspect in the area of Human Performance, because procedure LS-AA-119 did not provide adequate instructions to provide reasonable assurance that station management would properly control overtime for plant staff performing safety-related functions [H.2(c)].

Inspection Report# : 2007005 *(pdf)*

Significance: G Dec 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Deficient Control of Reactor Coolant System Vent Area During Mid-Loop Operation
The inspectors identified a NCV of TS 6.8.1a for failure to maintain the appropriate reactor coolant system (RCS) vent area required by station procedures during mid-loop operation. The reduced vent area degraded operators' ability to add water to the RCS in the event of a loss of decay heat removal (DHR) and caused reactor vessel level indication to be inaccurate. Corrective actions included operator crew briefings, communications between radiological protection and operations personnel, and initiation of IRs 698486 and 705000.

This issue affected the configuration control attribute of the Mitigating Systems cornerstone and was more than minor because it affected the availability of water from the borated water storage tank (BWST) to the RCS in the event of a loss of decay heat removal and caused reactor vessel level indication to be inaccurate. The inspectors determined that although design margin was reduced, the RCS gravity feed and bleed from the BWST function remained operable. The inspectors also concluded that level indication remained sufficient to alert operators if a significant change occurred which would warrant operator actions. Therefore a Phase 2 quantitative assessment was not required and the issue had very low safety significance. The finding has a cross-cutting aspect in the area of Human Performance, because work control during removal of the High Efficiency Particulate Air (HEPA) fans and ventilation hoses from the once through steam generator (OTSG) handholes was deficient. Operators and Radiation Protection (RP) technicians did not appropriately coordinate work activities to ensure the required RCS vent area was maintained when technicians established radiological postings for the OTSG handhole area [H.3(b)].

Inspection Report# : 2007005 *(pdf)*

Significance: G Dec 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Deficient Procedure Causes Failure to Perform Surveillance Testing of Valves in Accordance with ASME OM Code
The inspectors identified an NCV of TS 4.2.2 for failure to test eight safety-related valves in accordance with American Society of Mechanical Engineers Operations & Maintenance (ASME OM) Code requirements. Procedure OP-TM-211-211 contained no procedural steps to verify and document local position indication for eight safety-related make-up system valves. Plant staff revised the procedure and successfully tested the valves prior to the

completion of this inspection period.

The finding is more than minor because it affected the procedure quality attribute of the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This finding was determined to be of very low safety significance since the condition did not involve an actual loss of safety function. This finding has a cross-cutting aspect in the area of Human Performance, because TMI did not ensure complete and accurate procedures were available for testing eight safety-related make-up system valves [H.2(c)].

Inspection Report# : 2007005 *(pdf)*

Significance: G Dec 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Deficient Control of Temporary Ventilation During Mid-Loop Operation Leads to Unusual Event
A self-revealing NCV of TS 6.8.1.a was identified for failure to properly coordinate maintenance and operational activities associated with installing the OTSG primary lower manways during mid-loop operation. Installation of the OTSG lower manway cover, while temporary ventilation fans were exhausting air from the OTSG handhole RCS vent, caused an unexpected drop in reactor vessel level indication and declaration of an Unusual Event emergency. Corrective actions included removing the ventilation fans and ventilation hose from the OTSG handholes, restricting the use of the fans, and initiating IRs 698291, 698693, and 699314.

This issue affected the configuration control attribute of the Mitigating Systems cornerstone and was more than minor because this equipment lineup error affected the accuracy of reactor vessel level instrument indication during mid-loop operations, a high risk evolution. The inspectors determined that although all four reactor vessel level instruments were affected, their collective level indications, trends, and alarms provided sufficient information to alert operators in the event of a loss of inventory. Therefore a Phase 2 quantitative assessment was not required and the issue was of very low safety significance. The finding has a cross-cutting aspect in the area of Human Performance, because the installation and removal of temporary OTSG ventilation during installation of the OTSG lower primary manway were not appropriately coordinated to ensure the operational impact on reactor vessel level indication while in mid-loop operation was understood. Consequently reactor vessel level indication was inaccurate and not understood by operations personnel while the plant was in an elevated shutdown risk condition [H.3 (b)].

Inspection Report# : 2007005 *(pdf)*

Barrier Integrity

Significance: G Dec 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Ineffective Corrective Actions for Failure to Maintain Structural Design Clearances Inside Reactor Building Containment
The inspectors identified a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, for ineffective corrective actions to a previously identified NCV regarding a failure to maintain structural design clearances inside the reactor building. This violation involves several permanently installed structures inside the containment that did not meet the required separation distance to the containment liner. The inadequate structural clearance increased the likelihood of damage to the safety-related containment liner during a postulated seismic event. Corrective actions included evaluation of the specific conditions [Issue Reports (IR) 694026, 700592, and 700679] and initiation of actions to move the elevator support structure away from the containment liner during the 2009 refueling outage.

This finding is more than minor because it impacted the configuration control attribute of the Barrier Integrity cornerstone objective to ensure the containment barrier protects the public from radionuclide releases. The containment design parameter for clearance between structures and the containment liner was not maintained. The finding is of very low safety significance because the issue did not involve an actual open pathway in the physical integrity of the containment. This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, because engineering inspections performed as corrective actions to a previous NCV did not thoroughly evaluate the containment liner for additional clearance deficiencies [P.1(c)].

Inspection Report# : 2007005 *(pdf)*

Significance: G Dec 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Damaged Control Rod Assembly Due to Deficient Fuel Handling Practices
A self-revealing NCV of Technical Specification (TS) 6.8.1.c was identified for failure to properly implement procedures to safely move control rod assemblies (CRAs) within the spent fuel pool (SFP). Fuel handling operators did not monitor the control mast load cell during CRA movement activities and did not verify the transit path (including the control mast) was clear of obstruction prior to bridge or trolley movement. These human performance deficiencies resulted in a damaged CRA and had the potential to damage the affected fuel assembly (FA) cladding fission product barrier. Corrective actions included verifying proper CRA handling equipment operation, increased personnel and supervisory oversight for all FA or CRA moves, event lesson learned briefings, and various procedure revisions to strengthen verification requirements.

The issue was more than minor because it affected the human performance attribute of the Barrier Integrity cornerstone objective to ensure the fuel cladding barrier protects the public from radionuclide release. The CRA was damaged and another CRA had to be selected for core reload. However, the inspectors determined the affected FA fuel clad barrier was not damaged and that containment controls were unaffected. Therefore a Phase 2 quantitative assessment was not required and the issue screened to Green (very low safety significance). The finding has a cross-cutting aspect in the area of Human Performance, because fuel handling operations personnel did not follow procedure requirements for safely moving CRAs in the SFP [H.4(b)].

Inspection Report# : 2007005 *(pdf)*

Significance: G Dec 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Damaged Fuel Assembly Due to Deficient Fuel Handling Practices
A self-revealing NCV of TS 6.8.1.c was identified for failure to properly establish and implement procedures to safely reload fuel into the reactor vessel core. Fuel handling operators proceeded to insert a FA without properly verifying the fuel movement could be safely accomplished. The FA became hung up on a cable, was damaged, and required replacement and core redesign for cycle 18 operation. Corrective actions included core redesign, a stand-down and event briefing for all refueling personnel, procedure revisions, redesign of the shoehorn cables, a root cause evaluation of fuel handling errors, and additional cameras and viewing monitors to further improve visibility during core reload.

The issue was more than minor because it affected the human performance attribute of the Barrier Integrity cornerstone objective to ensure the fuel cladding design barrier protects the public from radionuclide release. The FA was damaged and another FA was selected for core reload, requiring core redesign analysis. However, the inspectors determined the affected FA fuel clad barrier remained intact and that containment controls were unaffected. Therefore a Phase 2 quantitative assessment was not required and the issue screened to Green (very low safety significance). The finding has a cross-cutting aspect in the area of Human Performance, because fuel handling personnel proceeded ahead in the face of uncertainty, without stopping to restore proper visibility [H.4(a)].

Inspection Report# : 2007005 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

Miscellaneous

Last modified : August 29, 2008