January 30, 2001

EA-01-012
EA-01-019

Mr. Robert G. Byram
Senior Vice President, Nuclear
PPL Susquehanna, LLC
Susquehanna Steam Electric Station
2 North Ninth Street
Allentown, PA 18101

SUBJECT: NRC'S SUSQUEHANNA STEAM ELECTRIC STATION INTEGRATED REPORT 05000387/2000-009, 05000388/2000-009

Dear Mr. Byram:

On December 31, 2000, the NRC completed an inspection at the Susquehanna Steam Electric Station. The enclosed report presents the results of that inspection. The results of this inspection were discussed on January 12, 2001, with Mr. B. Shriver and other members of your staff.

This inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. An examination of radwaste transportation and health physics activities was also conducted during this inspection. Within these areas, the inspections consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

The enclosed report also discusses an NRC identified issue that has preliminarily been determined to be of low to moderate safety significance. The matter involves an apparent substantial potential for personnel to sustain external radiation exposures in excess of the 10 CFR 20 occupational radiation exposure limits. It appears that the potential for such an exposure was due to the failure to adequately evaluate the radiological hazards posed by highly radioactive particles encountered between September 2000 through December 14, 2000 during work on irradiated reactor hardware disposal equipment and tools. As described in Section 2OS1 of the attached inspection report, it appears that your organization and program: (1) did not adequately evaluate and characterize the radiation exposure hazards posed by radioactive particles having significantly high activity; and (2) did not establish and implement adequate radiological safety controls to prevent shallow and deep-dose equivalent personnel exposures from exceeding regulatory requirements.

Our review of the significance of this issue took into account the specific characteristics and radiological hazards of the particles involved in the work at Susquehanna. The Occupational Radiation Safety Significance Determination Process (SDP) described in NRC Inspection Manual Chapter (IMC) 0609, Appendix C, indicates that a substantial potential for overexposure attributable to hot particles is outside the scope of the SDP. This portion of the SDP is based on the fact that radioactive hot particles generally cause only localized exposure of the skin, i.e., shallow dose equivalent (SDE); and such skin exposures are not considered risk significant. As noted in the NRC Enforcement Policy, NUREG-1600, Appendix D, Hot Particle Enforcement Policy, a "Hot Particle Exposure" is defined as an occupational dose to the skin resulting from exposure to radiation emitted from the radionuclides in a hot particle on the body or on the clothing of the exposed individual. However, in the subject case at Susquehanna, the radioactivity of the particles encountered was sufficiently high such that the substantial potential for overexposures existed not only for the skin (i.e., SDE), but also for the whole body, i.e., deep dose equivalent (DDE), due to the contribution of beta and gamma radiation that is characteristic of cobalt-60. Such DDE exposures could have caused the annual total effective dose equivalent limit (i.e., 5 rem), set forth in 10 CFR 20.1201, to be exceeded. Accordingly, the potential occupational exposure in this matter was different and more radiologically significant than hot particle exposure as defined by NRC Enforcement Policy. Therefore, the substantial potential for an overexposure attributable to the radioactive particles in this case is within the scope of the SDP.

Accordingly, we applied the Occupational Radiation Safety SDP to establish the safety bearing and importance of this issue. Although no overexposure is known to have occurred, the circumstances in this matter were such that your failure to effectively evaluate the radiological hazard presented by these radioactive particles, relative to DDE exposure, resulted in a condition in which a minor alteration of the exposure circumstances could result in personnel exposure in excess of the regulatory dose limit, i.e., substantial potential for overexposure. Since the matter did not involve a Very High Radiation Area (i.e., an area greater than 500 rem/hour), the issue was determined to have low to moderate safety significance. Accordingly, this matter is being considered as an apparent violation of 10 CFR 20.1501(a), and a preliminary WHITE finding.

Although we believe that we have sufficient information to make a final significance determination relative to this apparent violation, no Notice of Violation is presently being issued for this inspection finding, at this time. In accordance with the current NRC Enforcement Policy, you may provide a written statement of your position on the significance of this finding, including any supporting information. You may also request a Regulatory Conference to present your own assessment and evaluation of this matter for the consideration of the NRC staff. A Regulatory Conference on this matter would be open for public observation. Please contact Mr. John White at (610) 337-5114, within 10 days of the date of this letter, to notify the NRC of your intentions in this matter. If a response is not received within the time specified, excepting a granted extension, we will continue with our significance determination and enforcement decision process. You will be advised by separate correspondence on our final determination in this matter.

The NRC also identified three issues of very low safety significance (Green). Two issues, failure to perform a risk assessment prior to maintenance activities and inadequate alarm response procedures related to residual heat removal service water radiation monitors were determined to involve violations of NRC requirements. These issues were entered into your corrective action program and are discussed in the summary of findings and in the body of the inspection report. These issues involved a violation of NRC requirements, but because of the very low safety significance, the violations were not cited. If you contest these non-cited violation, 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 I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Susquehanna Steam Electric Station.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publically Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (The Public Electronic Reading Room).

If you have any questions please contact Curtis Cowgill of my staff at 610-337-5233.

Sincerely,

**/RA/**

A. Randolph Blough, Director
Division of Reactor Projects

Docket Nos. 05000387, 05000388
License Nos. NPF-14, NPF-22

Enclosure: Inspection Report 05000387/2000-009, 05000388/2000-009

cc w/encl:
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ADAMS ACCESSION NUMBER: ML010250469

ADAMS DOCUMENT TITLE: IR 05000387/2000-009, 5000388/2000-009, on 11/12-12/30/2000; PPL Susquehanna, LLC; Susquehanna Steam Electric Station

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| OFFICE | RI/DRP | RI/DRP | RI/ORA | RI/DRS | RI/DRP |
|---|---|---|---|---|---|
| NAME | SHansell | CCowgill | DHolody | JWhite | ABlough |
| DATE | 01/24/01 | 01/26/01 | 01/29/01 | 01/26/01 | 01/30/01 |

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**U.S. NUCLEAR REGULATORY COMMISSION**

REGION I

| | |
|---|---|
| Docket Nos.: | 05000387, 05000388 |
| License Nos.: | NPF-14, NPF-22 |
| Report No.: | 2000-009 |
| Licensee: | PPL Susquehanna, LLC |
| Facility: | Susquehanna Steam Electric Station |
| Location: | Post Office Box 35<br>Berwick, PA 18603 |
| Dates: | November 12, 2000 to December 31, 2000 |
| Inspectors: | S. Hansell, Senior Resident Inspector<br>J. Richmond, Resident Inspector<br>A. Blamey, Resident Inspector<br>J. Noggle, Senior Health Physicist |
| Approved by: | Curtis Cowgill, Chief<br>Projects Branch 4<br>Division of Reactor Projects |

SUMMARY OF FINDINGS

IR 05000387/2000-009, 5000388/2000-009, on 11/12-12/30/2000; PPL Susquehanna, LLC; Susquehanna Steam Electric Station; Units 1&2. Maintenance Risk Assessment and Emergent Work, Operator Workarounds, Access Control to Radiological Significant Areas.

The report covered a seven week period of resident inspection and an announced inspection by a regional senior health physicist inspector.

## A. <u>Inspector Findings</u>

### Cornerstone: Mitigating Systems

- Green. The inspectors identified a Non-Cited Violation for failure to assess risk prior to performing maintenance activities. PPL did not assess the risk of performing a Unit 1 high pressure coolant injection (HPCI) system test concurrent with a test of the “B” emergency diesel generator (EDG). This resulted in the inappropriate removal of the Unit 1 HPCI from service during the EDG testing and an unnecessary increase in risk for the one hour period that both the “B” EDG and Unit 1 HPCI were removed from service for testing.

  This finding was of very low safety significance because HPCI would have been available for vessel injection with minimal operator action. Independent calculation of the increase in core damage frequency associated with removal of both the EDG and HPCI system for one hour determined the risk to be within the very low safety significance band. (section 1R13)

- Green. PPL has not taken timely actions to resolve an issue regarding the ability of the high pressure coolant injection (HPCI) system to respond to a transient in which the main steam line isolation valves close and the reactor does not automatically shut down. Since 1991, PPL documents have recognized: that continued HPCI operation during this transient requires prompt operator action outside of the control room to bypass high suppression pool level signals to prevent HPCI valves from automatically changing the HPCI suction source from the condensate storage tank to the suppression pool; it was unreasonable to expect that the specified prompt operator actions would be reliably completed within the required time; and that this automatic HPCI suction transfer feature should be removed. To date, PPL has not removed this automatic HPCI suction transfer feature.

  A phase 2 significance determination process assessment concluded that this issue was very low safety significance based on the availability of safety relief valves and low pressure injection systems to respond if HPCI failed. (section 1R16.1)

- Green. The inspectors identified a Non-Cited Violation for inadequate alarm response procedures related to residual heat removal service water radiation monitors. The alarm response procedures were inadequate because, although residual heat removal service water process radiation levels are not expected to be high, the expected area background radiation levels during a loss of coolant accident would cause the radiation monitors to alarm and the procedures would then direct the operator to inappropriately shut down the residual heat removal and residual heat removal service water systems when they were required to mitigate the accident.

This finding was determined to be of very low risk significance because sufficient information was available for operators to recognize that these alarms were due to background radiation and not cause the systems to be shutdown. (section 1R16.2)

**Cornerstone: Occupational Radiation Safety**

- Preliminary White. During the period September 2000 through December 14, 2000, workers performed work on irradiated reactor hardware disposal equipment and tools, contaminated with highly radioactive particles on the refueling floor. While PPL took action to evaluate some aspects of the radiological hazards posed by these highly radioactive particles, PPL's organization and program: (1) did not adequately evaluate and characterize the radiation exposure hazards posed by these particles; and (2) did not establish and implement adequate radiological controls to prevent shallow-dose and deep-dose equivalent personnel exposure from exceeding regulatory requirements. While no personnel exposures in excess of 10 CFR 20 occupational limits are known to have occurred, the radiological conditions were such that a minor alteration in exposure circumstances could result in personnel exposure in excess of regulatory limits. Failure to effectively evaluate the radiological hazard as necessary to assure that the regulatory dose limits of 10 CFR 20.1201 were not exceeded is an apparent violation of 10 CFR 20.1501(a). This issue was assessed using the Occupational Safety Significance Determination Process (SDP) described in NRC Inspection Manual Chapter (IMC) 0609, Appendix C, and characterized as a preliminary WHITE finding. (Section 2OS1)

## B. <u>Licensee Identified Violations</u>

Violations of very low safety significance which were identified by PPL have been reviewed by the inspectors. Corrective actions taken or planned by PPL appear reasonable. These violations are listed in section 4OA7 of this report.

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# Report Details

Summary of Plant Status

Susquehanna Steam Electric Station (SSES) Unit 1 began the period at full power and operated at full power for the entire report period except for a planned power reduction to 65% power on December 9, 2000 for maintenance on the 5A Feedwater heater tube side drain. Unit 1 returned to 100% power on December 11, 2000. SSES Unit 2 operated at or near full power for the entire period with exceptions for control rod pattern adjustments, and control rod drive maintenance and testing.

## 1. REACTOR SAFETY
**Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness**

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed PPL's preparations for cold weather conditions and performed a plant walkdown for the residual heat removal service water system, emergency service water system, and the ultimate heat sink. The systems were selected because their safety related functions could be affected by cold weather conditions. The inspectors reviewed and evaluated plant condition using NDAP-00-0024, Rev. 2, "Winter Operation Preparations and Severe Weather Operation," OP-116-001, Rev. 23, "RHR Service Water," and OP-054-001, Rev. 19, "Emergency Service Water System."

b. Issues and Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

a. Inspection Scope

The inspectors performed partial system walkdowns to verify system and component alignment and note any discrepancies that would impact system operability. The inspectors verified selected portions of redundant or backup systems/trains while a system was out of service. The inspectors reviewed selected valve positions, electrical power availability, and the general condition of major system components. In addition, the following procedures were reviewed to ensure that written guidance was clear to operate the systems:

OP-054-001, "Emergency Service Water System"
OP-215-001, "Turbine Building Closed Cooling Water System"
ON-215-001, "Loss of Turbine Building Closed Cooling Water"
OP-244-001, "Condensate and Feedwater System"

The walkdowns included the following systems:

- Unit 2 “B” turbine building closed cooling water (TBCCW) while the “A” TBCCW pump was out of service for maintenance
- Division II emergency service water (ESW) system while the Division I ESW system was out of service for maintenance and testing

b. Issues and Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Area Inspection

a. Inspection Scope

The inspectors reviewed PPL’s Fire Protection Review Report, revision 9, dated August 8, 1997, to determine the required fire protection design features, fire area boundaries, and combustible loading requirements for the areas examined during this inspection. The inspectors then performed walkdowns of these plant areas to assess PPL’s control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures. The areas included:

- “C” emergency diesel generator room during painting activities in the room
- Unit 1 emergency switch gear rooms
- Unit 1 standby liquid control system and reactor protection system instrument rack areas, during modification work to fire detection system with preaction system PA-151 out of service
- Emergency service water system pump house during maintenance and testing activities

b. Issues and Findings

No findings of significance were identified.

.2 Station Fire Brigade Performance

a. Inspection Scope

On December 21, 2000, the inspectors observed an unannounced fire brigade drill (Drill Scenario No. 9 - Fire Zone 2-31E) from the control room and locally, to evaluate the readiness of PPL staff to prevent and fight fires. The inspectors reviewed the strategies and information in pre-fire plan FP-213-272, "Pre-Fire Plan for Fire Zone 2-31E," to verify if it was consistent with the fire protection design features, fire area boundaries, and combustible loading assumptions shown in PPL's Fire Protection Review Report. The inspectors observed the fire brigade members don protective clothing, turnout gear, and self-contained breather apparatus, enter the fire area, and utilize the pre-fire plan strategies. The inspectors observed the fire fighting equipment brought to the fire area scene to evaluate whether sufficient equipment was available for the simulated fire. The inspectors evaluated whether the fire hose lines identified in the pre-fire plan were capable of reaching the fire hazzard and whether hose usage was adequately simulated (e.g., laid out without flow constrictions). The inspectors observed fire fighting directions and radio communications between the brigade leader, brigade members, and the control room. The inspectors observed the post drill critique to evaluate if the drill objectives acceptance criteria were satisfied.

b. Issues and Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On December 27, 2000, the inspectors reviewed the licensed operators performance during a degraded electrical grid and loss of off-site power simulator scenario to identify discrepancies and deficiencies in training, and to assess licensed operator performance.

b. Issues and Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed PPL's follow-up actions for selected structure, system, or component (SSC) issues, to assess the effectiveness of PPL's maintenance activities. The inspectors reviewed the performance of selected SSCs to verify that problem identification and resolution of Maintenance Rule related issues had been appropriately monitored, evaluated, and dispositioned in accordance with the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance," and PPL procedure NDAP-QA-0413, "SSES Maintenance Rule Program." In addition, the inspectors reviewed selected SSC classification, performance criteria and goals as listed in PPL analysis EC-RISK-0528, "Risk Significant Systems, Structures, and Components for the Maintenance Rule and Generic Letter 89-10 Components," EC-RISK-1054, "SSC Availability Performance Criteria for the Maintenance Rule," and EC-RISK-1060, "Acceptable Number of Failures for Risk Significant SSCs in the Maintenance Rule." The inspector also reviewed the corrective actions to verify that the actions were reasonable and appropriate. The specific condition reports (CRs) included:

| | |
|---|---|
| CR 286440 | Failure of the Un it 2 “B” control rod drive pump discharge check valve to Close |
| CR297480 | Unit 1 “C” average power range monitor spurious actuation |
| CR 294751 | Unit 1 “B” fuel pool cooling pump tripped |

| | |
|---|---|
| CR 287211 | Unit 2 "B" turbine building closed cooling water pump discharge check valve failure to check, during manual system alignment to swap running pumps |

b. Issues and Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work (71111.13)

a. Inspection Scope

The inspectors reviewed PPL's assessment and management of selected maintenance activities to assess the effectiveness of PPL's risk management for planned and emergent work. The inspectors compared PPL's risk assessments and risk management actions against the requirements of 10 CFR 50.65(a)(4) and the recommendations of NUMARC 93-01 Section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities," dated February 2000, to verify that risk assessments were performed when required and appropriate risk management actions were identified.

The inspectors reviewed scheduled and emergent work activities with licensed operators and work coordination personnel to verify that risk management action threshold levels were correctly identified, and that appropriate implementation of risk management actions were performed, in accordance with PPL procedures NDAP-QA-1902, "Maintenance Rule Risk Assessment and Management Program," NDAP-QA-0340, "Protected Equipment Program," PSP-22, "Susquehanna Sentinel Program," and the SSES Team Manual. The inspectors reviewed the assessed risk configuration against the actual plant conditions and any in-progress evolutions or external events to verify that the assessment was accurate, complete, and appropriate for the issue. In addition, the inspectors performed control room and field walkdowns to verify compensatory measures, identified by the risk assessments, were appropriately performed. The specific plant configurations included:

| | |
|---|---|
| December 4, 2000 | Unit 1 high pressure coolant injection (HPCI) system and the “B” emergency diesel generator (EDG) removed from service for planned testing |
| December 11, 2000 | Compensatory actions to protect risk significant equipment, during "C" EDG work window |
| December 12, 2000 | “B" Turbine building closed cooling water (TBCCW) pump discharge check valve degraded and the “A” TBCCW pump removed for planned maintenance |
| December 13, 2000 | Unit 2 TBCCW system alignment following maintenance on the "A" pump ("A" pump in standby, "B" pump running with a degraded check valve) |
| December 18, 2000 | Unit 1 reactor building “B” chiller trip and failure of the “A” chiller to automatically load |

b. Issues and Findings

On December 4, 2000, the NRC identified that PPL did not assess the risk of performing a Unit 1 HPCI system test concurrent with a test of the “B” emergency diesel generator. At 11:36 p.m. on December 3, until 2:14 a.m. on December 4, PPL removed the “B” EDG from service for planned testing. PPL’s plan for the EDG testing, developed in accordance with NDAP-QA-1902, “Maintenance Rule Risk Assessment and Management Program,” recognized the elevated plant risk condition and did not schedule the removal of any additional safety significant equipment from service during this testing. Contrary to PPL’s planned schedule, from 12:30 a.m. until 1:30 a.m. on December 4, PPL removed the Unit 1 HPCI system, a safety significant system, from service to perform SO-152-004, “Quarterly HPCI Valve Exercising.”

10 CFR 50.65 (a)(4) requires, in part, that before performing maintenance activities (including, but not limited to, surveillances, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess the increase in risk that may result from the proposed maintenance activities. Contrary to this, prior to the Unit 1 HPCI test, PPL failed to assess the risk associated with testing the Unit 1 HPCI while the “B” EDG was also being tested. The failure to assess the risk associated with testing of the Unit 1 HPCI while the “B” EDG was also being tested is a finding that is also a violation. This finding is more than minor because it had an actual impact on safety in that it resulted in the inappropriate removal of the Unit 1HPCI system from service during the “B” EDG testing. This finding affects the mitigating system cornerstone. This finding was found to be of very low safety significance (Green) using the Reactor Safety Significance Determination Process because there was no actual loss of the HPCI safety system function and HPCI would have been available for vessel injection with minimal operator action. In addition the inspectors performed an independent calculation of the change in core damage frequency associated with the removal of the HPCI system for 1 hour while the “B” EDG was also removed from service and determined the risk increase to be within the very low safety significance band (< 1E-6). This violation of 10 CFR 50.65 (a)(4) is being treated as a Non-Cited Violation (EA-01-019), consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). This issue is documented in condition report 299792. **(NCV 05000387/2000009-01)**

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events (71111.14)

a. Inspection Scope

On November 21, 2000, the inspectors observed PPL’s response to a “D” emergency diesel generator (EDG) over voltage alarm. PPL declared the “D” EDG inoperable and implemented Technical Specification 3.8.1, “A.C. Sources - Operating.” The inspectors observed PPL maintenance activities (PCWO 297407), control of plant risk, implementation of TS and common cause failure analysis. PPL determined that the alarm resulted from a faulty alarm circuit relay base and that this condition would not have prevented the EDG from performing its required safety functions. The relay base was replaced and the EDG returned to service on November 23, 2000, at 11:12 p.m.

On December 18, 2000, the inspectors reviewed PPL’s response to a Unit 1 reactor building “B” chiller trip and the failure of the “A” chiller to automatically load. Normal drywell cooling was temporarily lost and the air temperature increased to 136.6°

Fahrenheit (F), above the Technical Specification (TS) limit of 135° F. The “A” chiller was manually loaded and drywell temperature returned to 130° F, below the TS value.

b. Issues and Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed selected operability determinations to assess the adequacy of the evaluations, the use and control of compensatory measures, compliance with the Technical Specifications, and the risk significance of the issue. The inspectors verified that the operability determinations were performed as required by procedure NDAP-QA-0703, “Operability Assessments.” The inspectors used the Technical Specifications, Technical Requirements Manual, Final Safety Analysis Report, and associated Design Basis Documents as references. The specific issues reviewed included:

| | |
|---|---|
| CR 299723 | Main steam safety relief valve eductor changed during rebuild without an evaluation |
| CR 301824 & CR 302723 | Unit 1 post accident monitoring indication for primary containment isolation valve position on SV-15782B, drywell hydrogen-oxygen analyzer |
| CR303627 & CR292836 | Unit 2 “C” residual heat removal pump discharge check valve leaks |
| CR 301863 | Unit 2 inadvertent half scram during turbine testing |

b. Issues and Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors reviewed selected equipment issues to determine if the functional capability of the system or human reliability in responding to an event would be affected. This review focused on the operator’s ability to implement abnormal or emergency operating procedures. The specific issues reviewed included:

1. High pressure coolant injection (HPCI) operation after a main steam isolation valve closure transient without an automatic reactor shutdown

2. Post accident residual heat removal (RHR) system operation

b. Issues and Findings

1. <u>HPCI Operation After a Main Steam Isolation Valve Closure Transient Without Automatic Reactor Shutdown</u>

   The inspectors identified that PPL has not taken timely actions to resolve an issue regarding the ability of the high pressure coolant injection (HPCI) system to respond to a transient in which the main steam line isolation valves close and the reactor does not automatically shut down. As described in the Final Safety Analysis Report Section 15.8.8, the plant response to this transient relies on HPCI to maintain reactor water level. Specifically, since 1991 PPL has been aware that continued HPCI operation during this transient requires prompt operator action outside of the control room to bypass high suppression pool level signals to prevent HPCI valves from automatically changing the HPCI suction source from the condensate storage tank to the suppression pool. If the HPCI suction is transferred to the suppression pool, the cooling water for the HPCI lube oil will be heated beyond normal operating temperature and may result in failure of HPCI. In May 1995, 10 CFR 50.59 evaluation for EO-1(2)00-113, (NL-92-020, Rev. 5) "Level/Power Control," and again in June 1997, "The Importance of Properly Treating Human Performance in Probabilistic Risk Assessments," PPL determined that it was unreasonable to expect that the specified prompt operator actions would be reliably completed within the required time. PPL determined in those documents that the feature which causes the automatic change of the HPCI suction from the condensate storage tank to the suppression pool on high suppression pool level should be removed. Currently, PPL has not removed this automatic HPCI suction transfer feature.

   The issue of not removing the automatic HPCI suction transfer feature is more than minor because it involves a credible impact on safety in that if the main steam line isolation valves close and the reactor does not automatically shut down, HPCI may not be able to maintain reactor water level. This issue affects the mitigating system cornerstone and was reviewed using the Reactor Safety Significance Determination Process (SDP). A phase 2 assessment was performed because the safety function of HPCI was assumed to have been lost. The phase 2 SDP review concluded that this issue was of very low safety significance (Green) based on the availability of safety relief valves and low pressure injection systems to respond to the HPCI failure. No violations of NRC requirements were identified because this is beyond the licensing basis for HPCI.

2. <u>Post Accident Residual Heat Removal (RHR) System Operation</u>

   On November 17, 2000, the inspectors identified that the alarm response procedures for the RHRSW radiation monitors, AR-1(2)09-001F01, Revision 21and AR-1(2)13-001F01, Revision 19, were inadequate in that they would have required the operator to shut down the residual heat removal (RHR) and residual heat removal service water (RHRSW) systems during a loss of coolant accident (LOCA). During a LOCA, although the RHRSW process radiations levels are not expected to be high, the background area radiation levels around these monitors would result in a high RHRSW radiation alarm and require the operators to shut down the RHR and RHRSW systems. This issue is a violation and is more than minor because, if the procedures are not corrected, the inadequate procedures would become a more significant concern in that they would potentially cause inappropriate shutdown of RHR pumps needed to assure adequate core cooling during a LOCA. This issue affects the mitigating systems cornerstone and was determined to be of very low risk significance (Green) by the Reactor Safety Significance Determination Process. This conclusion was based on the recognition that the simultaneous actuation of both alarms following a LOCA, without RHRSW sample flow, would be sufficient information for operators to recognize that these alarms were due to background radiation and not cause a loss of safety function of the RHR and RHRSW systems. The inadequate procedure is considered to be a violation of Technical Specification 5.4.1 which requires written procedures to be established and implemented in accordance with Regulatory Guide 1.33 which requires procedures for alarm conditions. This violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the NRC Enforcement Policy issued May 1 2000 (65 FR 25368). This violation is in the corrective action program as CR 305722. **(NCV 05000387, 388/2000-009-02)**

The inspectors also determined that PPL failed to identify these inadequate alarm response procedures for RHRSW radiation monitors. Specifically PPL's review of procedures related to RHRSW radiation monitor response during a LOCA did not identify inappropriate procedure actions. On April 1999, PPL determined that during a LOCA the RHRSW radiation monitor could alarm from the high background radiation, not high RHRSW radiation. PPL subsequently reviewed system procedures to determine if any procedure changes were required. PPL's procedure review did not identify these inadequate procedures.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors observed post-maintenance testing activities and reviewed the PPL test data. The inspectors verified the test success criteria addressed in the procedures was in compliance with Technical Specification requirements. The specific issues reviewed included:

| | |
|---|---|
| PCWO 286565 | Unit 2 “B” turbine building closed cooling water pump discharge check valve repair |
| PCWO 253127 | “C” emergency diesel generator intake / exhaust valve lock nut torque check and jacket water flush |
| PCWO 302256 | Unit 1 reactor building chilled water "B" condenser pump motor and breaker failure |

b. Issues and Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the applicable Technical Specifications and observed the performance of selected portions of surveillance tests. The test result were reviewed to verify that the tested systems and components were capable of performing their safety functions. The tests observed or reviewed include:

| | |
|---|---|
| SO-258-001 | Weekly manual scram control switch functional check |
| SO-216-003 | Unit 2 residual heat removal service water quarterly flow verification test |
| SO-054-A03 | Quarterly Emergency Service Water Flow Verification Loop A |
| SO-104-001 | Monthly Bus 1A201, 1A202, 1A203, 1A204, and OB565 Degraded Voltage Channel Functional Test |

b. Issues and Findings

No findings of significance were identified.

## 2. RADIATION SAFETY

### Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

Radiological safety controls associated with a waste consolidation and shipping of irradiated reactor hardware project were reviewed. The review included: observation of control of radioactive particles, attendance at an ALARA pre-job meeting for removal of a stellite roller punch tool, and discussions with applicable radiation protection staff members. The inspection also involved the review of documents, including applicable refueling floor radiation work permits, radiation surveys and personnel dose evaluations due to exposure to radioactive particles. Applicable procedures with respect to radioactive particle controls and resulting skin dose assessments were also reviewed. In addition, ten condition reports associated with radioactive particle events since September 2000, were reviewed including CR No. 289959, dated October 12, 2000, that included a Level 1 root cause analysis of radiological controls.

b. <u>Issues and Findings</u>

The inspector identified that during the period September 2000 through December 2000, PPL failed to adequately evaluate the radiological hazards associated with irradiated reactor hardware disposal equipment and tools that were contaminated with highly radioactive particles (i.e., very small particles, principally cobalt-60, containing millicurie levels of radioactivity). Such highly radioactive particles were capable of causing shallow-dose equivalent (SDE) and deep-dose equivalent (DDE) radiation exposures in excess of the 10CFR20 occupational exposure limits if not properly and effectively controlled. Accordingly, the occupational exposure potential from these radioactive particles was distinctly different than “hot particle” exposure as described in the NRC’s Enforcement Policy, NUREG-1600, Appendix D, in which “hot particle” exposure is only defined as occupational exposure to the skin, i,e, SDE. Contrary to the requirements of 10 CFR 20.1501(a), PPL failed to adequately evaluate the radiological hazards associated with these radioactive particles to ensure compliance with the requirements of 10 CFR 20.1201 occupational dose limits.

While PPL had conducted surveys, was aware of the presence of radioactive particles in the work place, and effected some degree of radiological control to address the perceived radiological hazards, their evaluation did not consider all of the exposure consequences. Specifically, PPL failed to recognize that some particles encountered were so radioactive as to pose a significant shallow dose and deep dose equivalent radiological exposure hazard in certain circumstances; and, subsequently failed to establish effective personnel exposure controls necessary to assure that occupational exposure limits would not be exceeded.

During the period of the work, PPL set up a containment tent on the refueling floor to provide an enclosure for working on tools and equipment. PPL’s radiological controls for the work on December 6, 2000, required protective clothing and established that individuals would be surveyed at 15 minute intervals to determine if radioactive particles resided on their person. On December 6, 2000, a decontamination worker was scrubbing material off an advanced crusher/shearer (ACS) machine stand inside the enclosure. When the individual was surveyed at a 15 minute interval, PPL determined that a highly radioactive particle resided on the individual’s right boot. Subsequently, PPL determined that the radioactive particle contained about 1.4 millicuries of Cobalt-60. A conservative dose assessment performed by PPL indicated the worker sustained a 17 rem shallow-dose equivalent (SDE) exposure which was within the 10 CFR 20.1201 shallow and extremity dose limit of 50 rem.

Prior to December 6, 2000, PPL had encountered other highly radioactive particles in the work area. On September 9, 2000, a radioactive hot particle caused an unplanned exposure of 12 rem, SDE (i.e., skin exposure) to the right forearm of a worker, an exposure that was within the occupational limits specified in 10 CFR 20.1201. At that time, PPL initiated a condition report and had the opportunity to conduct a thorough review of the exposure event, the circumstances surrounding it, and the adequacy of radiological controls (e.g., protective clothing, stay-times, and survey requirements). However, no actions were immediately taken and work activities were allowed to continue without any change in radiological controls.

On October 12, 2000, following discovery of a 75 millicurie particle on the refueling floor, a Level 1 condition report was initiated and an event review team (ERT) was assigned to evaluate the condition. While a more thorough assessment was conducted, the inspector determined that ERT analysis failed to recognize that PPL's radiological surveys of highly radioactive particles relied on the use of conventional survey instruments which had the potential to lead to underestimation of the radiological consequences and hazards posed by such particles. Following the ERT's recommendation to establish a more conservative radiological control protocol, the staff reduced the stay-time interval to 15 minutes, but without effective evaluation of the possible exposure hazards posed by these highly radioactive particles, and consequently, did not provide reasonable assurance that deep dose equivalent (DDE) and shallow dose equivalent (SDE) exposure to workers would not exceed the 10 CFR 20 occupational dose limits.

Subsequently, other highly radioactive hot particles were discovered on November 28 (two hot particles, 20.6 millicurie each) and December 4, 2000 (18.8 millicurie) in areas accessible to workers. While these discoveries provided opportunities to re-evaluate the radiological hazard potential and re-assess the adequacy of radiological controls, radiological survey and personnel exposure monitoring activities, PPL continued to implement only the established radiological controls, and failed to recognize or consider any other radiological implication or exposure hazard posed by the recurring presence of such highly radioactive particles.

The 1.4 millicurie hot particle which caused the 17 rem (SDE) dose to the worker on December 6, 2000, was capable of causing a SDE dose to the worker in excess of the 10 CFR 20 occupational limits (50 rem) well within PPL's 15 minute survey interval if it had been located on a less protected portion of the body. It was only fortuitous that the dose to the worker was limited in this circumstance, due to radioactive hot particle residing on the outside of the worker's boot. Notwithstanding, other radioactive particles encountered by PPL on October 12, 2000 (75 millicurie); November 28, 2000 (two 20.6 millicurie particles) and December 4, 2000 (18.8 millicurie) had the potential to exceed both the SDE and DDE dose limits within several minutes if located on a portion of the whole body, a potential that could occur with only a minor change in the exposure circumstances.

Accordingly, the radioactivity exhibited by these particles was sufficient to cause personnel exposure in excess of the regulatory requirements relative to either shallow or deep dose equivalent well within the 15 minute stay-time interval established by PPL's hot particle control procedure. It was fortuitous and not by design that overexposure did not occur. PPL's failure to effectively evaluate and address this condition, as required by 10 CFR 20.1501(a), constitutes a substantial potential to exceed occupational dose limits specified in 10 CFR 20.1201.

The Occupational Radiation Safety Significance Determination Process described in NRC Inspection Manual Chapter (IMC) 0609, Appendix C, indicates that a substantial potential for overexposure attributable to hot particles is outside the scope of the SDP. However, this portion of the SDP is based on the position that hot particle exposure means an occupational dose to the skin, i.e., shallow dose equivalent (SDE); and such skin exposures are not considered risk significant. As noted in the NRC Enforcement

Policy, NUREG-1600, Appendix D, Hot Particle Enforcement Policy, a “Hot particle exposure” is defined as an occupational dose to the skin resulting from exposure to radiation emitted from the radionuclides in a hot particle on the body or on the clothing of the exposed individual. In this case, the radioactivity of the particles encountered was sufficiently high, such that the substantial potential for overexposures from the particles existed not only for the skin (i.e., SDE), but also for the whole body, i.e., deep dose equivalent (DDE) due to the contribution of beta and gamma radiation that is characteristic of cobalt-60. Such DDE exposures could have caused the annual total effective dose equivalent limit (i.e., 5 rem) set forth in 10 CFR 20.1201 to be exceeded. As such, the substantial potential for an overexposure attributable to the highly radioactive particles in this case is considered within the scope of the SDP.

Applying the SDP, the inspector determined that this matter did not constitute an ALARA finding; and confirmed that no overexposure is known to have occurred. Notwithstanding, the circumstances in this matter were such that PPL’s failure to effectively evaluate the radiological hazard presented by these radioactive particles relative to DDE exposure resulted in a condition in which a minor alteration of the exposure circumstances could result in personnel exposure in excess of the regulatory dose limit, i.e., substantial potential for overexposure. Since the matter did not involve a Very High Radiation Area (i.e., an area greater than 500 rem/hour), the issue was determined to have low to moderate safety significance. Accordingly, this matter is being considered as an apparent violation of 10 CFR 20.1501(a), and a preliminary WHITE finding (EA-01-012). **(AV 05000387, 388/2000009-03)**

**Cornerstone: Public Radiation Safety**

2PS2 Radioactive Material Processing and Transportation (71122.02)

a. Inspection Scope

The inspector walked down radwaste equipment spaces, the radwaste process area, the low level radioactive waste storage facility, and the spent fuel storage installation. Material condition of operating and abandoned radwaste equipment, radwaste storage areas and radioactive waste storage inventories were evaluated.

b. Issues and Findings

No significant findings or issues were identified.

## 4. OTHER ACTIVITIES

4OA3 Event Follow-up (71153)

.1 (Closed) LER 05000388/00-003-00 Inadvertent Engineered Safety Feature Actuation Caused by Loss of Reactor Protection System Power (RPS) Supply

On July 4, 2000, the Unit 2 “B” reactor protection system (RPS) power was lost due to the failure of the RPS “B” motor-generator voltage regulator card. The failure resulted in a RPS “B” half scram and corresponding containment isolations. PPL replaced the voltage regulator card and reestablished the “B” motor-generator set as the normal power source for the “B” RPS system. The LER was reviewed and no findings of significance were identified. This issue was documented in condition report CR 269440. This LER is closed.

.2 (Closed) LER 05000387/00-009-00 Primary Containment Isolation Valves not Checked per Surveillance Requirements

On August 10, 2000, PPL identified that 87 Unit 1 and 85 Unit 2 primary containment isolation valves (PCIVs) had not been tested as part of monthly TS Surveillance Requirement 3.6.1.3.2. PPL determined the cause of the PCIVs not being included in the surveillance was an unclear definition of containment boundary components. All of the valves were subsequently tested, with no identified leakage. Additional corrective actions, completed or planned, included revising the associated surveillance procedure, and clarifying the wording in the TS Bases. No new issues were identified in this review. This is a minor violation not subject to formal enforcement. This issue was documented in condition report CR 276714. This LER is closed.

.3 (Closed) LER 05000387/00-011-00 Missed Surveillance Requirement for Post Accident Monitoring Instrumentation Valve Position Indication Function

On October 20, 2000, PPL identified that monthly channel checks required by Technical Specifications for primary containment isolation valve position indications, associated with post accident monitoring instrumentation, had not been performed on Unit 1 from October 1998 to October 2000. These channel checks had not been required prior to October 1998, and had been overlooked during PPL's conversion from the previous standard Technical Specifications to the current Improved Technical Specifications, in October 1998. PPL determined the cause of the event was personnel error, in that an individual failed to include the corrective actions from a previous similar event into the corrective action program tracking system. PPL revised the surveillance procedures and the channel checks were completed satisfactorily. No new issues were identified in this review. This is a minor violation not subject to formal enforcement This issue was documented in condition report CR 291538. This LER is closed.

.4 (Closed) LER 05000388/00-004-00 Technical Specification Interpretation Incorrect - Operation Prohibited by Technical Specifications

On July 17, 2000, PPL identified that one primary containment isolation valve (PCIV) was inoperable and the associated TS Limiting Condition for Operation was not entered. Specifically, from April 11 to 14, 2000, one of the PCIVs in a hydrogen/oxygen (H2O2) analyzer penetration was inoperable, and the penetration was not isolated, as required by the Unit 2 TS Section 3.6.1.3. PPL determined the cause to be unclear wording in the Final Safety Analysis Report for the design basis for the H2O2 analyzer penetration and a non-conservative TS Interpretation (TSI) for the associated section. Corrective actions included a revision to the specific TSI, a review of all the existing TSIs for non-conservative direction, and a plan to eliminate all TSIs.

This issue is more than minor because it had a credible impact on safety in that if the redundant valve in the penetration did not close on a containment isolation signal, containment integrity would not be assured. This finding affects the Barrier Integrity Cornerstone and was considered to have very low safety significance (green) using the Significance Determination Process because, the likelihood of an accident leading to core damage was not affected, the probability of early primary containment failure was negligible, and the redundant isolation valve remained operable during this event. This PPL identified violation is discussed in Section 4AO7. This issue was documented in PPL's corrective action program as CR 272262. This LER is closed.

## 4OA4 Cross Cutting Issues

### .1 Problem Identification and Resolution

The inspectors identified inconsistences in implementation of PPL's problem identification and resolution program. Specifically, the inspectors noted two examples in which PPL had the opportunity but failed to identify problems. While PPL had conducted surveys and was aware of the presence of radioactive hot particles in the work place, PPL failed to recognize that some hot particles encountered were so radioactive as to pose a significant shallow dose and deep dose equivalent radiological exposure hazard in certain circumstances (2OS1). During procedure reviews PPL failed to identify problems associated with alarm response procedures for the residual heat removal service water radiation monitors (1R16.2). In addition, the inspectors identified that PPL had not taken timely actions to resolve an longstanding issue regarding the ability of the high pressure coolant injection system to respond to a transient in which the main steam line isolation valves close and the reactor does not automatically shut down. (1R16.1).

## 4OA6 Meetings

### .1 Exit Meeting Summary

On January 12, 2001, the resident inspectors presented the inspection results to Mr. B. Shriver and other members of your staff who acknowledged the findings.

On January 25, 2001, the resident inspectors and a senior health physicist presented the inspection results of the occupational radiation safety inspection to Mr. B. Shriver and other members of your staff who acknowledged the findings.

The inspectors asked PPL whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee Identified Violations

The following findings of very low safety significance were identified by PPL and are violations of NRC requirements which meet Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as Non-Cited Violations (NCVs).

| NCV Tracking Number | Requirement Licensee Failed to Meet |
|---|---|
| 05000388/2000009-04 | Technical Specification 3.6.1.3, required that a primary containment penetration be isolated within 4 hours, if the associated primary containment isolation valve (PCIV) was not operable. Contrary to this, from April 11 to 14, 2000, a PCIV for a Unit 2 hydrogen-oxygen analyzer was not operable, and the penetration was not isolated. |
| 05000387, 388/2000009-05 | 10 CFR 20.1501(a)(1), requires that surveys be made to comply with the regulations in 10 CFR Part 20 including 10 CFR 20.1902(b) for posting of high radiation areas (defined as an area greater than 100 mR/hr at 30 centimeters(cm)). On November 12, 2000, a shipping cask had not been surveyed properly and, as a result, an area measuring 700 mR/hr at 30 centimeters was undetected and constituted an unposted high radiation area. This event is documented in Condition Report No. 297422. |

## ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

| | | |
|---|---|---|
| 05000387, 388/2000009-03 | AV | Failure to Conduct an Adequate Radiological Survey in Accordance With 10 CFR 20.1501 (section 2OS1) |

Opened and Closed

| | | |
|---|---|---|
| 05000387/2000009-01 | NCV | Failure to Perform a Risk Assessment Prior to Planned Maintenance Activities (section 1R13) |
| 05000387, 388/2000009-02 | NCV | Post Accident Residual Heat Removal System Operation (section 1R16.2 ) |
| 05000388/2000009-04 | NCV | Technical Specification Interpretation Incorrect - Operation Prohibited by Technical Specifications (section 4OA7 ) |
| 05000387, 388/2000-009-05 | NCV | Failure to Post a High Radiation Area. (section 4OA7 ) |

Closed

| | | |
|---|---|---|
| 05000387/00-009-00 | LER | Primary Containment Isolation Valves not Checked per Surveillance Requirements (section 4OA3.2 ) |
| 05000387/00-011-00 | LER | Missed Surveillance Requirement for Post Accident Monitoring Instrumentation Valve Position Indication Function (section 4OA3.3 ) |
| 05000388/00-003-00 | LER | Inadvertent Engineered Safety Function Actuation Caused by Loss of Reactor Protection System Power Supply (section 4OA3.1) |
| 05000388/00-004-00 | LER | Technical Specification Interpretation Incorrect - Operation Prohibited by Technical Specifications (section 4OA3.4 ) |

## LIST OF ACRONYMS USED

| | |
|---|---|
| AC/S | Advanced Crusher/Shearer |
| ALARA | As Low As is Reasonably Achievable |
| ARM | Area Radiation Monitor |
| ATWS | Anticipated Transient Without Scram (failure of an automatic reactor shutdown) |
| CCTV | Closed Circuit Television Camera |
| CFR | Code of Federal Regulations |
| CR | Condition Report |
| CST | Condensate Storage Tank |
| EDG | Emergency Diesel Generator |
| ERO | Emergency Response Organization |
| ERT | [SSES] Event Review Team |
| ESW | Emergency Service Water |
| F | Fahrenheit |
| FR | Federal Register |
| FSAR | [SSES] Final Safety Analysis Report |
| FPRR | [SSES] Fire Protection Review Report |
| H2O2 | Hydrogen / Oxygen Monitor |
| HP | Health Physics |
| HPCI | High Pressure Coolant Injection |
| IDS | Intrusion Detection System |
| LER | Licensee Event Report |
| LOCA | Loss of Coolant Accident |
| MSIV | Main Steam Isolation Valve |
| NCV | Non-Cited Violation |
| NRC | Nuclear Regulatory Commission |
| NSSS | Nuclear Steam Supply System |
| PASS | Post Accident Sampling System |
| PCIV | Primary Containment Isolation Valve |
| PI | Performance Indicator |
| PPL | PPL Susquehanna, LLC |
| QA | Quality Assurance |
| RHR | Residual Heat Removal |
| RHRSW | Residual Heat Removal Service Water |
| SCBA | Self-Contained Breathing Apparatus |
| SDE | Shallow dose equivalent |
| SGTS | Standby Gas Treatment System |
| SRV | Safety Relief Valves |
| SSC | Structure, System, or Component |
| SSES | Susquehanna Steam Electric Station |
| TBCCW | Turbine Building Closed Cooling Water |
| TIP | Transverse In-Core Probe |
| TS | Technical Specification |
| TSI | Technical Specification Interpretation |

## ATTACHMENT 1

## NRC's REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

| **Reactor Safety** | **Radiation Safety** | **Safeguards** |
|---|---|---|
| • Initiating Events<br>• Mitigating Systems<br>• Barrier Integrity<br>• Emergency Preparedness | • Occupational<br>• Public | • Physical Protection |

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.