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ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Perry Nuclear Power Plant
Docket No. 50-440

Subject: Report of Facility Changes, Tests and Experiments

In accordance with 10 CFR 50.59(d)(2), the Report of Facility Changes, Tests, and Experiments for the Perry Nuclear Power Plant is provided. The report covers the period from the last submittal dated September 9, 2005 to the present.

Attachment 1 provides the summaries of the 10 CFR 50.59 Evaluations.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 761-6071.

Sincerely,

Barry S. Allen

Attachment: Perry Nuclear Power Plant, Report of Facility Changes, Tests, and Experiments for the period September 9, 2005 to the present

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

IE47

NRR

PERRY NUCLEAR POWER PLANT
Report of Facility Changes, Tests, and Experiments for the period September 9, 2005 to the present

Evaluation No: 05-05649

Source Document: Engineering Change Package (ECP) 04-0032

Title: Installation of 3-position spring-return-to-normal control switches for the Target Rock solenoid valves in the M51 system (Containment Combustible Gas Control)

1.1 Activity Description

Description of Issue

The Target Rock solenoid valves in the M51 system have dual solenoid coils for opening and closing operations. The current control switch configuration requires that one of the solenoid coils be energized at all times. This results in a reduced qualified life due to the ohmic heat load created by the coils during the energized state. As a result, the solenoids on several of the valves are replaced every refuel outage.

Description of the Modification

This modification will install 3-position spring-return-to-normal control switches for open/close control of Target Rock valves 1M51F0210A/B, 1M51F0220A/B, 1M51F0230A/B, 1M51F0240A/B, 1M51F0250A/B, 1M51F0260A/B and 1M51F0270A/B. The new control switches will replace the existing 2-position maintained - contact General Electric CR2940 control switches.

This modification will install two relays in the control circuits for the 1M51F0250A/B solenoid valves. This will allow the valves to operate correctly with the newly installed 3-position control switches. New single coil solenoid valves will be installed for 1M51F0250A/B.

The relays installed by this ECP will cause the coils of the solenoid valves to energize when the control switches are taken to the open position. The coils will then stay energized until the control switches are taken to the closed position. This will leave the solenoid valves energized during M51 system operation, but will allow the solenoids to be de-energized during normal plant operations in which the M51 system is not operational.

This modification is being made to extend the qualification life of the Target Rock solenoid coils in the Combustible Gas Control System by leaving them in the de-energized state during normal plant operations. This change does not impact the operation of the Combustible Gas Control System. These valves are used to isolate containment during accident conditions and also allow for the re-opening of the appropriate valves for system operation following an accident. This modification only impacts the Hydrogen Analysis subsystem of the M51 system. There is no effect on the

Combustible Gas Mixing, Hydrogen Recombiner and the Backup Hydrogen Purge systems as a result of this modification.

This modification affects Perry Nuclear Power Plant's (PNPP) compliance with respect to Regulatory Guide 1.11. Regulatory Guide 1.11 states that isolation valves on instrument sensing lines will fail in the "as is" condition. The change will result in isolation valves 1M51F0250A/B failing in the closed position.

1.2 Summary of Evaluation

The Updated Final Safety Analysis Report (UFSAR) accidents and events were reviewed with respect to the effects of the proposed design change in the analysis section of this evaluation. None of the accident frequencies were found to be adversely affected by the implementation of the proposed change. This evaluation analyzes UFSAR described design functions that are potentially affected by the proposed design change. Both direct and indirect effects of the proposed changes on design functions were evaluated. Potentially affected design functions were found to be satisfactorily performed. Considerations of the potential for equipment failure and an increased likelihood of a malfunction associated with the design change were reviewed. The likelihood of any malfunctions of equipment important to safety is not increased by this change. This change does not result in increasing previously evaluated release rates, changing release duration, establishing a new release mechanism or path, or changing mitigation effectiveness. The proposed changes do not result in increased dose consequences that might impede required actions inside or outside the control room to mitigate the consequences of nuclear accidents. Based on the scope of the proposed changes and the evaluation of possible failure effects of the proposed changes, no new events of significance, nor new malfunctions of any Systems, Structures and Components (SSC) important to safety, could be identified. This evaluation analyzed the effects of the proposed changes on the PNPP fission product barriers. No effects were identified. The methods of evaluation that support the proposed design change are consistent with the methods utilized in establishing the design bases and the safety analyses documented in the UFSAR.

In conclusion, the proposed activity does not meet any of the criteria in paragraph (c)(2) of 10 CFR 50.59 and therefore the evaluation of the proposed changes has determined that a license amendment is not required.

1.3 Is a license amendment required prior to implementation of the change?

No

Evaluation No: 05-06738

Source Document: ECP 04-0345

Title: Deletion of the automatic isolation from differential temperature instruments

1.1 Activity Description

This ECP will delete the automatic isolation signals from differential temperature instruments in the main steam line tunnel, the reactor water cleanup rooms and the residual heat removal areas. The differential temperature signals will provide an alarm function only.

This design change is limited to the E31 Leak Detection system.

1.2 Summary of Evaluation

The UFSAR accidents were reviewed with respect to the effects of the proposed design change. None of the accident frequencies were found to be affected by the implementation of the proposed change. This evaluation analyzes UFSAR described design functions that are potentially affected by the proposed design change. Both direct and indirect effects of the proposed changes on design functions were evaluated. Potentially affected design functions were found to be not impacted or changed. Consideration of the potential for equipment failure and an increased likelihood of a malfunction associated with the design change were reviewed. The likelihood of any malfunctions of equipment important to safety is not increased by this change. This change does not result in increasing previously evaluated release rates, changing release duration, establishing a new release mechanism or path, or changing mitigation effectiveness. The proposed changes do not result in increased dose consequences that might impede required actions inside or outside the control room to mitigate the consequences of nuclear accidents. Based on the scope of the proposed changes and the evaluation of possible failure effects of the proposed changes, no new events of significance, nor new malfunctions of any SSC important to safety, could be identified. This evaluation analyzed the effects of the proposed changes on the PNPP fission product barriers. No effects were identified. The methods of evaluation that support the proposed design change are consistent with the methods utilized in establishing the design bases and the safety analyses documented in the UFSAR.

In conclusion, the proposed activity does not meet any of the criteria in paragraph (c)(2) of 10 CFR 50.59 and therefore the evaluation of the proposed changes has determined that a license amendment is not required.

1.3 Is a license amendment required prior to implementation of the change?

No

Evaluation No: 06-00185

Source Document: ECP 05-0229

Title: Division 1 and 2 Diesel Generator Bus Under/Degraded Voltage Start Logic Modification

1.1 Activity Description

This modification is relocating the diesel generator bus under/degraded voltage start signal to the LOCA diesel generator start logic. Additionally, bus under/degraded voltage seal-in relays are required to lock-in the diesel start signal. The seal-in relays are required due to the bus under/degraded voltage relays de-energizing after the bus is re-powered by the diesel generator.

An issue was identified that involves the inability of the diesel generators to re-start either manually or upon receipt of a bus under/degraded voltage signal during shutdown from surveillance testing. However, by relocating the bus under/degraded voltage diesel generator start signal to the LOCA diesel generator start logic, the issue no longer exists.

As a result of relocating the diesel generator bus under/degraded voltage start signal the diesel generator operation will be altered as follows:

During a bus under/degraded voltage diesel generator start signal, the associated diesel generator start signal will override testing activities, specifically the 2-minute period after diesel generator shutdown.

During a bus under/degraded voltage diesel generator start event, the associated diesel generator non-essential trips will be bypassed by the newly installed seal-in relays.

If both diesel generator air start headers decrease to 150 psig, the associated diesel generator under/degraded voltage start signal will be blocked. This will ensure that sufficient air remains for an additional manual start attempt. The under/degraded voltage diesel generator air start logic will be identical to the existing LOCA air start logic.

1.2 Summary of Evaluation

The UFSAR accidents and events were reviewed with respect to the effects of the proposed design change in the analysis section of this evaluation. None of the accident frequencies were found to be adversely affected by the implementation of the proposed change. This evaluation analyzes UFSAR described design functions that are potentially affected by the proposed design change. Both direct and indirect effects of the proposed changes on design functions were evaluated. Potentially affected design

functions were found to be satisfactorily performed. Consideration of the potential for equipment failure and an increased likelihood of a malfunction associated with the design change were reviewed. The likelihood of any malfunctions of equipment important to safety is not increased by this change. This change does not result in increasing previously evaluated release rates, changing release duration, establishing a new release mechanism or path, or changing mitigation effectiveness. The proposed changes do not result in increased dose consequences that might impede required actions inside or outside the control room to mitigate the consequences of nuclear accidents. Based on the scope of the proposed changes and the evaluation of possible failure effects of the proposed changes, no new events of significance, nor new malfunctions of any SSC important to safety, could be identified. This evaluation analyzed the effects of the proposed changes on the PNPP fission product barriers. No effects were identified. The methods of evaluation that support the proposed design change are consistent with the methods utilized in establishing the design bases and the safety analyses documented in the UFSAR.

In conclusion, the proposed activity does not meet any of the criteria in paragraph (c)(2) of 10 CFR 50.59 and therefore the evaluation of the proposed changes has determined that a license amendment is not required.

<u>1.3 Is a license amendment required prior to implementation of the change?</u>

No

Evaluation No: 06-00228

Source Document: ECP 04-0049-01

Title: Standby Diesel Generator (SDG) Governor Replacement

1.1 Activity Description

ECP 04-0049 replaces the Standby Diesel Generator (SDG) speed control system components as follows: the Woodward EGA analog speed control and associated resistor box are replaced with a Woodward 2301A Reverse Acting Dual Dynamics analog speed control; the Woodward EGB-35C hydraulic actuator is replaced by a Woodward EGB-35P actuator; and the Motor Operated Potentiometer (MOP) is replaced by a Woodward Digital Reference Unit (DRU). A new magnetic pickup speed sensor is installed to provide a high frequency speed signal to the new 2301A speed control system. The Woodward 2301A speed control is an analog speed controller which will perform the same functions as the EGA analog speed control, as well as allowing engine slow starts. The new hydraulic actuator and DRU provide the same functions as the components which they replace.

There are two ECP supplements per diesel generator to implement the changes; the scopes of the packages are as follows:

ECP 04-0049-00 - Division 2 Non-Outage Package: All work involving modification of the flywheel, flywheel guard, installation of the magnetic speed pickup and bracket, installation of new speed control panel, installation of conduit, and field routing of cables (not terminations in existing panels) is not required to be performed during a plant outage but does require a system outage for part of this work. The only post-modification testing required following completion of this work scope to restore the SDG to OPERABLE status is to perform the normal monthly surveillance test in accordance with SVI-R43-T1318. Portions of ECP 04-0049-00 may be performed during the plant outage as required based on time and parts availability.

ECP 04-0049-01 - Division 2 Outage Package: All remaining work for implementation of the ECP, i.e., removal of old components, termination of field cables in existing panels, etc., including post-modification testing will be included in this package and in a refueling outage or forced outage of sufficient duration.

The objective of these modifications is to eliminate parts obsolescence issues with the existing speed control components. The existing Woodward EGA speed control and MOP used on the SDGs have become obsolete and are no longer manufactured. The Woodward 2301A speed control and DRU will be available for the foreseeable future. Additionally, the NRC has recognized accelerated engine aging degradation associated with fast starting emergency diesel generators. As a result, the NRC recommends that the engines be slow started and loaded in accordance with the vendor recommendations for all test purposes other than the refueling outage Loss-Of-Offsite

Power tests and once per six month "fast start" tests. With the 2301A, a "slow start" capability is added for the Division 2 SDG. This evaluation is only for the Division 2 SDG ECPs and associated supporting documents.

1.2 Summary of Evaluation

The Division 2 SDG will be fully capable of performing its design functions subsequent to the proposed changes. The design function and method of operation for the Division 2 SDG remains unchanged. The controls, control circuitry and method of operation for the Division 2 SDG are not adversely impacted by the proposed change.

The SDG system does not in itself initiate any currently evaluated accidents; the failure of both SDGs are included in the Station Blackout (SBO) evaluated in UFSAR Section Appendix 15H. However, this accident is initiated by Loss of Offsite Power followed by the failure of the SDGs. This is a unique case in that it takes the failure of both SDGs following a Loss of Offsite Power for this event to occur. The changes proposed for the Division 2 SDG speed control system will not adversely impact any of the assumptions used in determining the coping duration for an SBO, including the reliability of any of the SDGs. The changes implemented by this change do not create any situations that could initiate a new accident. Consequently, the frequency of occurrence of the currently evaluated accidents does not increase and the possibility of initiating accidents of a different type is not created. Other accidents evaluated in the UFSAR credit the operation of the SDGs for mitigation of the events but are not caused by the failure of an SDG. The mitigation effectiveness of the affected safety systems remains unchanged, and therefore, the radiological consequences from accidents and malfunctions do not increase. The affected safety system will be fully capable of providing a reliable, safety-related power source to accident mitigation systems, and therefore, the fission product barriers will not be adversely affected. The proposed change utilizes accepted analytical methods and does not conflict with any UFSAR described methodologies. The change does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design basis or in the safety analysis. The basis for the proposed change is the obsolescence of the MOP and EGA speed control system. The design function and the manner in which the system supports the design and license basis remain unchanged. The proposed change does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

In conclusion, the proposed change does not meet any of the criteria in paragraph (c)(2) of 10 CFR 50.59 and therefore the evaluation of the proposed change determined that a license amendment is not required.

1.3 Is a license amendment required prior to implementation of the change?

No

Evaluation No: 06-00680

Source Document: ECP 03-0404

Title: Installation of a Digital Reactor Feed Pump Turbine Speed Control System

1.1 Activity Description

ECP 03-0404 replaces the existing General Electric MDT-20 electronic governor system for the Reactor Feed Pump Turbine (RFPT) Control System with a new Invensys/Foxboro I/A Series Digital Reactor Feed Pump Turbine Speed Control System (DRFPTSCS).

The new DRFPTSCS is integrated into the Digital Feedwater Control System (DFWCS). An Invensys/Foxboro I/A Series digital system replaced the previous Bailey Feedwater Control System. The digital system is fault tolerant, minimizing single failure point vulnerability internal and external to the DFWCS. The system includes redundancy in its power sources to meet this requirement. System redundancy is accomplished by a secondary control processor that waits in stand-by mode, mirroring the operation of the primary, and is automatically placed in service when failure of the primary is detected. The DFWCS includes signal validation features, so as to be able to detect invalid inputs, outputs and other signals. When a fault is detected, the standby equipment takes over without control interruption. The DFWCS includes redundancy in all control processors, critical I/O modules, power supplies and bus communication devices between controllers and I/O modules. The DRFPTSCS has these same features in regard to system redundancy.

The DRFPTSCS includes the use of Woodward Servo Position Controllers (SPCs) to position the operating cylinder which mechanically positions the low pressure and high pressure steam valve torque arms. The DRFPTSCS has its own I/O racks with their own power sources. The DRFPTSCS has its own redundant processors, which share the same rack as the DFWCS processors and uses the same redundant power sources. The DRFPTSCS also shares the operator workstations with the DFWCS.

Each RFPT has a pair of Woodward Servo Position Controllers (SPCs) which are configured in a cascade to provide position control for the low pressure and high pressure control valves. Each of the SPCs have alarm and status indications which are available to the operator through the I/A system. The operator has the ability to reset the SPC's alarms via the Woodward Operating Cylinder Controller interface overlay and the Woodward Pilot Valve controller interface overlay. Through the Woodward SPC interface overlays on the I/A system workstations, the operators also have the ability to shutdown and reset the SPCs.

1.2 Summary of Evaluation:

Installation of the new DRFPTSCS improves the reliability of the Reactor Feedpump Turbine Speed control system. The system reliability is based on several factors including (1) a highly dependable system software developed in accordance with industry standards, (2) self diagnostics and internal fault tolerance, (3) redundancy in its control processor, redundancy of critical inputs/outputs and its handling of input and output signals, and redundancy in its power supplies, (4) qualification to the intent of Regulatory Guide 1.180, Revision 1 in regard to EMI/RFI, and (5) Mean Time Between Failure (MTBF) rates of system components that are significantly better than the analog system. This system reliability minimizes failures or malfunctions of the DRFPTSCS, which in turn minimizes other SSC failures or malfunctions since adverse effects will not be propagated from the DRFPTSCS to other interfacing systems. This reliability also minimizes the occurrence of UFSAR described initiating events. Given this high degree of reliability, the frequency or likelihood of occurrence of previously evaluated accidents and SSC malfunctions is not increased. In particular, the frequency of occurrence of the Reactor Feedpump Turbine Speed Control Failure - Maximum Demand and Loss of all Feedwater events is not increased.

The licensing basis does not rely on the Reactor Feedpump Turbine Speed Control System for mitigation of accidents or malfunctions. Consequently, installation of a new digital Reactor Feedpump Turbine Speed Control System that performs the same basic functions as the originally installed system and also does not perform consequence mitigation functions, cannot increase the radiological consequences of accidents or malfunctions. The non-safety related Level 8 high reactor vessel water level trip that is processed by the feedwater control system and which trips the main turbine and turbine feed pumps will be retained. Several UFSAR Chapter 15 events take credit for this trip in the evaluation of the incident scenario. Since the trip will be retained and the plant response to these events will be maintained within the original acceptance criteria, the outcome of these events will not change, and thus the radiological consequences of the events will not increase. Further, the unchanged sequence of events in these transients guarantees that the fission product barrier performance does not change from its current state, and thus design basis limits on fission product barriers will not be exceeded.

Accidents of a different type or malfunctions of SSCs with a different result will not occur due to the high degree of reliability attributed to the DRFPTSCS and due to its EMI/RFI qualifications that prevent any adverse interactions with other systems. Protection of the DRFPTSCS from induced voltages is provided and therefore system malfunctions from induced voltages that could create accidents of a different type or malfunctions with a different result are prevented. The high degree of software dependability ensures that the possibility of a common cause or common mode software failure is sufficiently low and therefore considered unlikely, and thus accidents of a different type or malfunctions with a different result will not be created. Consequently, malfunctions with a different result cannot be created.

In conclusion, the proposed activity does not meet any of the criteria in paragraph (c)(2) of 10 CFR 50.59 and therefore the evaluation of the proposed change determined that a license amendment is not required.

1.3 Is a license amendment required prior to implementation of the change?

No

Evaluation No: 06-03003

Source Document: ECP 05-0186

Title: Replacement of Auxiliary Platform 1F15E0005

1.1 Activity Description

The existing Auxiliary Platform will be replaced by a new 360 Degree Auxiliary Platform for service work on the refuel floor. The replacement is designed to provide for increased efficiency on the refuel floor and reduce critical path time associated with fuel movements, In-Vessel Visual Inspections and other vessel servicing. The primary advantage of the new design is the ability to perform fuel movements with the Refuel Bridge concurrent with vessel service work from the 360 Degree Platform. A portion of the existing auxiliary platform is the detachable vessel platform and its associated circular track system that is designed to be utilized at the reactor vessel flange for vessel inspections when the reactor cavity is dry. The new 360 Degree Auxiliary Platform does not have this vessel platform or its rail system.

1.2 Summary of Evaluation

The auxiliary platform is not considered an initiator for any UFSAR described accident or event; however, the proposed use of the platform concurrent with fuel moves could impact the Fuel Handling Accident inside containment. The auxiliary platform is designed as a work station to perform work over the upper containment pools including reactor vessel servicing. The new platform includes a 1000 pound capacity chain hoist which is limited to 500 pound loads administratively. This administrative limit combined with the hoist's limited horizontal reach assures that the hoist will not be lifting loads over spent fuel while doing service work and cannot lift spent fuel. The initial installation and subsequent setup of the new auxiliary platform into its various configurations will require the lifting of loads including heavy loads either over or near spent fuel. The site procedures for all load lifts will be followed and will satisfy NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants," requirements. Furthermore, the combination of existing refuel platform electronic controls and a dedicated spotter will provide assurance that the potential for collisions of the Refuel Platform Mast with the auxiliary platform are minimized. In addition, operation of the platform bridge remains a manual function even though the motor drive system utilizes an electronic programmable drive. Based on the above, the assembly and use of the new platform will not result in more than a minimal increase in the frequency of occurrence of the Fuel Handling Accident inside containment.

The new platform has been re-classified from Safety Related to Non-Safety Related, Augmented Quality. The re-classification is consistent with the definition of equipment that is not safety related since the only safety function of the platform is passive, to remain intact during and following a seismic event in order to prevent falldown on fuel or safety related equipment. The increased quality requirements associated with the

Augmented classification, including design, material and fabrication controls, assure the completed platform will satisfy all design requirements. Both the existing and the new platforms have been analytically qualified in accordance with Seismic Category I requirements such that the potential for a falldown of the structure during a seismic event is not credible. The software associated with the new drive system is limited to control of drive motor parameters and is not utilized for any automatic control action. The potential for electromagnetic interference or radio frequency interactions with other plant equipment is minimized by the design of the controls. Therefore, the proposed modification will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety that was previously evaluated in the UFSAR.

The movement of all loads required for installation and normal operation and storage locations have been evaluated to insure against unwanted interactions with other equipment. In addition, the new platform is designed and built to industry standards to assure reliability. Furthermore, adequate shielding between workers in the platform troughs/carriages and spent fuel will be maintained. Assuming a malfunction of the Refuel Platform that would place spent fuel near personnel working on the Auxiliary Platform, radiation levels would be low enough to allow adequate time for personnel to exit the platform without receiving unacceptable radiation doses. Thus, the proposed platform will not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety that was previously evaluated in the UFSAR and will not create the possibility of an accident of a different type than any previously evaluated in the UFSAR.

Based on the evaluation performed, the proposed design change and associated documentation does not meet any of the criteria in paragraph (c)(2) of 10 CFR 50.59. Therefore, the proposed activity does not require a license amendment.

<u>1.3 Is a license amendment required prior to implementation of the change?</u>

No

Evaluation No: 07-00644

Source Document: Calculation FM-012

Title: OPRM Device Settings and Setpoints - Cycle 12 changes

1.1 Activity Description

The purpose of this evaluation is to evaluate the change in methodology used to perform the Reload 11 / Cycle 12 core stability analysis. In the development of the new Delta Over Initial Critical Power Ratio versus Oscillation Magnitude Curve (DIVOM Curve), computer codes PANACEA Version 11 and TRACG Version 04 were used.

NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996, references the use of computer codes PANACEA Version 10 and TRACG Version 02 for the development of the DIVOM Curve.

Calculation FM-012, OPRM Device Settings and Setpoints was revised and updated for the Cycle 12 core design and stability analysis. This update documents the revision to the Perry DIVOM Curve.

1.2 Summary of Evaluation

General Electric completed a detailed evaluation between TRACG Version 02 coupled with PANACEA Version 10 as compared to TRACG Version 04 coupled with PANACEA Version 11. This analysis is contained in GE report GE-NE-0000-052-5690-Revision 0, "TRACG04 DIVOM 10 CFR 50.59 Evaluation Basis," April 2006.

This analysis shows the results are essentially the same for TRACG02 with PANACEA 10 and TRACG04 with PANACEA 11. Use of a different methodology that has essentially the same results is allowed under 10 CFR 50.59 rules as interpreted by NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation" Section 4.3.8.

Therefore, the use of TRACG04 with PANACEA 11 in calculating the plant-specific DIVOM does not constitute a departure of a method of evaluation as described in the UFSAR.

1.3 Is a license amendment required prior to implementation of the change?

No