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L-09-217

10 CFR 50.59(d)(2)

ATTN: Document Control Desk  
U. S. Nuclear Regulatory Commission  
Washington, DC 20555-0001SUBJECT:  
Perry Nuclear Power Plant  
Docket No. 50-440, License No. NPF-58  
Report of Facility Changes, Tests and Experiments

In accordance with 10 CFR 50.59(d)(2), the FirstEnergy Nuclear Operating Company hereby submits the Report of Facility Changes, Tests and Experiments (Attachment) for the Perry Nuclear Power Plant. The report covers the period from the last submittal dated September 7, 2007 to the present.

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,



Mark B. Bezilla

Attachment:  
Perry Nuclear Power Plant, Report of Facility Changes, Tests, and Experiments for the Period September 7, 2007 to the Presentcc: Nuclear Regulatory Commission Region III Administrator  
Nuclear Regulatory Commission Resident Inspector  
Nuclear Reactor Regulation Project ManagerTE47  
NRR

Attachment  
L-09-217

Perry Nuclear Power Plant,  
Report of Facility Changes, Tests, and Experiments for the Period  
September 7, 2007 to the Present  
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Evaluation No: 05-04712, Revision 1

Source Document: Engineering Change Package (ECP) 04-0270-000  
Engineering Change Package (ECP) 04-0270-001  
Engineering Change Package (ECP) 04-0270-003

Title: Alternate Decay Heat Removal (ADHR)

### 1.1 Activity Description

#### Description of Issue

Engineering Change Package (ECP) 04-0270 installs an alternate decay heat removal (ADHR) system. Plant events [loss of emergency service water (ESW) as documented in condition report (CR) 03-05065 and CR 04-2598] have identified that the plant lacks alternatives for decay heat removal in Mode 4. Technical Specification 3.4.10, Cold Shutdown in Mode 4, requires two operable residual heat removal (RHR) shutdown cooling subsystems. With one RHR system inoperable, Technical Specifications require that an alternate system be available for each inoperable shutdown cooling subsystem. Assuming a single failure of an RHR system, an ESW system, or a single standby diesel generator, a gap exists in decay heat removal capability between 200 degrees Fahrenheit when Mode 4 is entered and 150 degrees Fahrenheit where condenser feed and bleed could potentially be used as an alternate decay heat removal system. While the existence of this gap is consistent with the plant's design and licensing basis, plant management has elected to pursue installation of an ADHR system to improve the plant's shutdown safety margin and to provide outage equipment maintenance flexibility.

The intent of the ADHR system is to assure Technical Specification compliance in Mode 4 by providing an additional ADHR option that does not rely upon RHR or ESW. The system has also been designed to operate in Modes 4 and 5 to provide outage flexibility. The ADHR system is designed to be able to remove the decay heat load that exists approximately 24 hours after plant shutdown from 100 percent power.

The ADHR system will include a dedicated heat exchanger and pump, suction and return lines consisting of piping, valves, instrumentation, and appurtenances with tie-ins to the low pressure core spray (LPCS) system, RHR system, and condensate transfer and storage (CTS) system. The cooling water for the ADHR heat exchanger is provided by the service water (SW) system consisting of piping, valves, instrumentation, and appurtenances with tie-ins to the SW system piping downstream of the turbine building closed cooling water (TBCCW) heat exchanger. An ADHR heat exchanger area is created by installing a radiation shield wall on the 599' level of the auxiliary building. This design change includes the addition of two cooling units (water cooled packaged air conditioning units) that will be installed to remove the heat load added by

the installation of the new equipment and components in the LPCS pump room and in the ADHR heat exchanger area. To assure the ADHR system cooling water flow rate requirements are met, SW flow through the TBCCW heat exchanger cannot be modulated, which is the existing TBCCW temperature control method. Therefore, as part of the ADHR design change, the TBCCW system will be modified with the addition of a three-way bypass valve and associated piping, instrumentation and appurtenances to modify system temperature control that will maintain full SW flow through the TBCCW heat exchanger.

#### Description of the Modification

ECP 04-0270 is divided into three supplements. Supplement 002 was voided and the work that was to be contained within that supplement has been added to supplement 001.

Supplement 000 (ECP 04-0270-000, Large Bore SW System Piping) provided the design for the installation of large bore SW system supply and return piping that will provide the cooling water for the ADHR heat exchanger. The supplement 000 design change has been evaluated under 10 CFR 50.59 screen 05-04616.

Supplement 001 (ECP 04-0270-001, ADHR System Installation) provides the design for the installation of the ADHR system components and equipment. This supplement contains information for pre-outage and outage work activities, except for the changes needed to the TBCCW system changing the system's temperature control scheme from SW flow controlled to TBCCW flow controlled.

Supplement 003 (ECP 04-0270-003, TBCCW System, Water Temperature Control) provides the design for the new TBCCW system water temperature control scheme. Supplement 003 has been evaluated under 10 CFR 50.59 screen 05-04713.

As identified in 10 CFR 50.59 screen 05-04712, revision 2, numerous design calculations were prepared in support of ECP 04-0270-001. Revision 2 of the screen determined that these calculations screen out from further evaluation. Therefore, these calculations will not be discussed further in this 10 CFR 50.59 evaluation except in the context of supporting other arguments. As such, this 10 CFR 50.59 evaluation has been revised to no longer contain the list of modification related calculations. For a full list of the calculations that support ECP 04-0270-001 refer to screen 05-04712, revision 2. In addition, this change involves Updated Final Safety Analysis Report (UFSAR) change request 05-083. Further, the Technical Specifications Bases are being updated to reflect the ADHR system as another alternate decay heat removal system that can satisfy the Technical Specification requirement to have a decay heat removal method that can function as an alternate to the RHR system.

Installation Standard Specifications ISS-2000 (Piping and Mechanical Equipment Installation), ISS-2300 (Insulation of Plant Systems), and American Society of

Mechanical Engineers (ASME) Design Specifications DSP-E12 (RHR System Piping and Pipe Supports ASME III Division I) and DSP-E21 (LPCS System Piping and Pipe Supports ASME III Division I) are being changed via a Specification Change Notice to reflect the above design changes.

ECP 04-0270-001 may be implemented as a series of partial closures with limiting condition for operation (LCO) reviews of the work order for the piping tie-ins or any other work that can be divisionally implemented to allow work to proceed. Therefore, partial closure is acceptable.

### 1.2 Summary of Evaluation

Based on the analysis performed for this 10 CFR 50.59 evaluation, the effects of supplement 001 to ECP 04-0270 have been reviewed against the causes of all of the UFSAR evaluated accidents. Since the causes are not affected, the frequency of occurrence of any accident previously evaluated in the UFSAR is not increased.

The effects of ECP 04-0270-001 were reviewed against potentially affected UFSAR described design functions. From that review, there is sufficient evidence to conclude that the LPCS and RHR systems will maintain the ability to function as credited in the UFSAR to mitigate design basis events, including earthquakes and tornadoes. This change maintains the redundancy, independence and separation of the RHR and LPCS systems. In addition, the new ADHR system will perform the decay heat removal function as described in the UFSAR when limited to Modes 4 and 5 and 24 hours after shutdown. Based on the evaluations and analyses performed, ECP 04-0270-001 does not affect the likelihood of occurrence of a malfunction of an SSC (structure, system, or component) important to safety previously evaluated in the UFSAR.

Any accidents that rely on the function of the LPCS and RHR systems would be mitigated as evaluated in the UFSAR. This change does not result in any increases in dose release rate or duration, does not create any new radiological release mechanisms or paths, and maintains the effectiveness of SSCs credited to mitigate accident dose. Therefore, ECP 04-0270-001 does not increase the radiological consequences of any accident or malfunction of equipment important to safety evaluated previously in the UFSAR.

ECP 04-0270-001 does not cause a previously evaluated event to become categorized as an accident. This change does not make any events previously categorized in the UFSAR as incredible to be categorized as credible. Based on the analysis performed, the change does not create the potential for the occurrence of any event of such significance that it could be categorized as an accident.

Based on the analysis performed, ECP 04-0270-001 does not create a new potential for common mode failure of the RHR or LPCS systems. The malfunction effects identified in the analysis section are bounded by previously evaluated UFSAR malfunctions.

Based on the analysis performed ECP 04-0270-001 maintains the UFSAR described design functions of the LPCS and RHR systems. Consequently, there are no indirect effects on the design basis limits for any of the fission product barriers.

The analyses that establish the ADHR system design/qualification are consistent with the UFSAR descriptions. Post modification testing will confirm the proper function of the ADHR system. All of the analytical/empirical methods utilized are considered to be consistent with the methods described in the UFSAR that establish the design basis of the RHR decay heat removal function.

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

No.

Evaluation No: 06-01428

Source Document: Engineering Change Package (ECP) 04-0049-002  
Engineering Change Package (ECP) 04-0049-003

Title: Division 1 Standby Diesel Generator (SDG) Governor Modification

### 1.1 Activity Description

#### Description of Issue

ECP 04-0049-002 and ECP 04-0049-003 makes the following changes to the Division 1 standby diesel generator (SDG) speed control system: 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.

The Woodward 2301A speed control is an analog speed controller which will perform the same functions as the Woodward 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.

#### Description of the Modification

ECP 04-0049-002 – Division 1 Pre-Outage/Outage Package, includes installation of the new speed control panel and installation of the conduit. (A portion of this package will be installed during the outage).

ECP 04-0049-003 – Division 1 Outage Package, includes all remaining Division 1 work for implementation of the ECP; that is, removal of old components, installation and termination of field cables in existing panels, modification of the flywheel, flywheel guard, installation of the magnetic speed pickup and bracket. This work scope must be performed in a refueling outage or forced outage of sufficient duration to support both implementation and testing activities.

The objective of these modifications is to eliminate parts obsolescence issues with the existing governor components.

The 2301A speed control system provides the capability of starting the SDG in a slow speed mode. This design change includes the necessary hardware and operator interface (Slow/Fast selector switch) to implement the slow start capability. However, due to outage constraints, sufficient time is not available to fully test the control logic

and 2301A dynamic response necessary to consider the feature operational. Therefore, it has been determined not to make this option available for use at the present. Use of the slow start feature will be blocked administratively by making the Slow/Fast selector switch a keylock switch with the switch locked in the Fast position. The control logic and 2301A slow start dynamics; that is, start fuel limit, associated with the slow start circuit is bypassed and passive as long as the Slow/Fast selector switch is in the Fast position. The following discussions regarding the functionality of the slow start feature are maintained since the feature will be operational in the future whenever the control logic and 2301A are tested to ensure their correct operation and setup. Prior to utilizing the slow start feature, this evaluation will be revised or another 10 CFR 50.59 review will be completed to ensure all design and licensing requirements are fully met.

## 1.2 Summary of Evaluation

The Division 1 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 1 SDG remains unchanged. The controls, control circuitry and method of operation for the Division 1 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 Updated Final Safety Analysis Report (UFSAR) 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 1 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 diesel generators (DGs). 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 motor operated potentiometer (MOP) and Woodward 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 (structure, system, or component) 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: 08-00688

Source Document: Engineering Change Package (ECP) 08-0099-001

Title: Temporary Jumper to Disable Flow Switch (1N62N0102A) – Low Flow Signal

### 1.1 Activity Description

#### Description of Issue

This Temporary Modification (TM) Engineering Change Package (ECP) temporarily defeats the low steam flow isolation signal to the last stage of steam jet air ejector (SJAE) "A" isolation air operated valves (AOVs) 1N62F0140A and 1N62F0170A. The trip signal from flow switch 1N62N0102A to AOV solenoid valves 1N62F0141A and 1N62F0171A will be jumpered out. This is to ensure that SJAE "A" remains on-line in case the flow switch activates its low steam flow trip due to a flow switch malfunction.

However, flow switch 1N62N0102A is currently indicating off-scale high due to the effects of a probable steam leak at the flange location of 1N62N0100A, its associated flow element (that is, orifice). As such, Operations personnel have lost flow switch 1N62N0102A as a reliable indication but will be able to use the following alternate means to validate proper SJAE "A" operation:

- 1N11R0406 main steam supply pressure to steam jet air ejectors (control room panel 1H13P0870),
- 1N11R0400 main steam supply pressure to steam jet air ejectors (turbine building 605' elevation panel 1H51P1007),
- 1N11R0405 main steam supply pressure regulating valve position controller air signal pressure (turbine building 605' elevation panel 1H51P1007), or
- 1N64R0620 off-gas flow (control room panel 1H13P0845)

The low steam flow trip signal from the last stage SJAE "A" flow switch 1N62N0102A is being temporarily defeated. This trip signal isolates AOVs 1N62F0140A and 1N62F0170A when steam flow to the last stage of SJAE "A" reaches the low flow trip setpoint of 7,440 pounds/hour steam.

The design function of the last stage air ejector is to supply sufficient steam to the off-gas system to maintain the hydrogen concentration downstream at less than 4 percent by volume. Flow element 1N62N0100A is provided to measure steam flow to the last stage SJAES. Normally energized solenoid valves 1N62F0141A and F0171A will de-energize and close isolation AOVs 1N62F0140A and 1N62F0170A on a low steam flow trip signal. These AOVs are to remain closed until proper steam flow has been established (reset value of 8,340 pounds/hour steam). The flow element, flow switch, isolation AOVs and their solenoid valves are non-safety related, non-seismic classified

components. The control wiring is non-safety and non-safety jumpers will be used to defeat the automatic isolation function of the flow switch.

The TM ECP is being implemented in conjunction with troubleshooting and temporary repair activities associated with a steam leak identified on steam flow element 1N62N0100A. The steam flow indication has pegged high (>10,000 pounds/hour) and the reliability of the flow switch may be reduced due to the steam leak and subsequent troubleshooting activities (insulation removal, sealant injection, etc.).

The installation of the proposed jumper will prevent the automatic isolation of SJAE suction valves 1N62F0140A and 1N62F0170A during low steam flow to SJAE "A".

#### Description of the Modification

This TM ECP temporarily defeats the low steam flow isolation signal to the last stage of SJAE "A" isolation AOVs 1N62F0140A and 1N62F0170A. The trip signal from flow switch 1N62N0102A to AOV solenoid valves 1N62F0141A and 1N62F0171A will be jumpered out. This is to ensure that SJAE "A" remains on-line in case the flow switch activates its low steam flow trip due to a flow switch malfunction.

#### 1.2 Summary of Evaluation

This TM ECP temporarily defeats the low steam flow isolation signal to the last stage of SJAE "A" isolation AOVs 1N62F0140A and 1N62F0170A. The trip signal from flow switch 1N62N0102A to AOV solenoid valves 1N62F0141A and 1N62F0171A will be jumpered out. This is to ensure that SJAE "A" remains on-line in case the flow switch activates its low steam flow trip due to a flow switch malfunction.

The TM ECP is being implemented in conjunction with troubleshooting and temporary repair activities associated with a steam leak identified on steam flow element 1N62N0100A. The steam flow indication has pegged high (>10,000 pounds/hour) and the reliability of the flow switch may be reduced due to the steam leak and subsequent troubleshooting activities (insulation removal, sealant injection, etc.)

Accordingly, the installation of the proposed jumper per the TM will prevent the automatic isolation of SJAE suction valves 1N62F0140A and 1N62F0170A during low steam flow conditions (process) to SJAE "A".

The evaluation concluded that sufficient alarms currently in place will alert the operators that there may exist a low dilution steam flow to SJAE "A" and place the unit in a safe condition in the event that steam flow cannot be re-established. The evaluation credits several alarm response instructions alerting operators that a low dilution steam flow exists. This temporary modification is expected to be in place while troubleshooting and temporary repair activities associated with a steam leak identified on steam flow element 1N62N0100A are being performed.

The evaluation was performed assuming that an occurrence of low dilution steam flow occurred while the jumpers were in place. Various alarms are provided in the control room which would result from a potential low dilution steam flow occurrence. It is concluded that the actions to close the suction valves would be realized by these alarms. The conclusion was that the plant would not experience a transient worse than evaluated in the Updated Final Safety Analysis Report (UFSAR) for a loss of condenser vacuum. Failure of the steam jet air ejector piping per UFSAR 15.7.1.3 is considered a limiting fault and the temporary modification does not increase the frequency of this event. Additionally, the event which could cause a gross failure of the off-gas system is a seismic event. The equipment and piping are designed to contain any hydrogen – oxygen detonation. A detonation is not considered as a possible failure mode.

The proposed activity cannot increase consequences of an accident or a malfunction of equipment, nor will it create the possibility of an accident of a different type or malfunction of an SSC (structure, system, or component) with a different result. The activity cannot impact fission product barriers nor is it considered a departure of a methodology used in establishing the design basis.

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

No.

Evaluation No: 09-01526

Source Document: Procedure SVI-G33-T9131

Title: Type C Local Leak Rate Test of 1G33 Penetration P131

### 1.1 Activity Description

#### Description of Issue

The scope of this activity is limited to the changes that deviate from Safety Evaluation 97-0079, specifically, permitting reactor water cleanup (RWCU) suction valve 1G33F0101 to remain in the open position during the surveillance test while the freeze seal is in place. As an alternative measure, an operator will be dedicated to manually close RWCU suction valve 1G33F0101 if the freeze seal ruptures the pipe. In the event that the valve cannot be manually manipulated due to mechanical failure or that environmental conditions do not permit local access (that is, break flow or spray makes the valve inaccessible), a pipe crimping tool will be pre-staged on the pipe that can be utilized to effectively crimp the 3-inch diameter pipe minimizing any leakage from the system.

#### Description of the Modification

This was a change to a procedure.

### 1.2 Summary of Evaluation

The change to SVI-G33-T9131, Type C Local Leak Rate Test of 1G33 Penetration P131, permits RWCU suction valve 1G33F0101 to remain open during the installation of a freeze seal during the local leak rate testing of RWCU containment isolation valves 1G33F0001 and 1G33F0004. The evaluation considered valve closure or crimping of the pipe in the unlikely event that the pipe ruptures while the freeze plug is established.

The evaluation concluded that either the closure of the valve or crimping of the pipe by dedicated personnel will not cause excessive leakage from the reactor vessel. Any leakage past the known leaking valves will easily be made up with emergency core cooling systems.

The evaluation concluded that in the event of a pipe rupture at the freeze seal location, redundant methods of closing off the pipe from the reactor to the break exists such that nuclear fuel will be continuously covered and cooled and that reactor inventory can be maintained.

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

No.

Evaluation No: 09-01528

Source Document: Engineering Change Package (ECP) 09-0244-001  
Engineering Change Package (ECP) 09-0244-002

Title: Evaluation of N1B Recirculation Outlet Nozzle Plug Installation to Facilitate the Rework of the 1E12F0010 Isolation Valve

### 1.1 Activity Description

#### Description of Issue

As documented in Condition Reports (CR) 09-57011 and 09-56938, the residual heat removal (RHR) system shutdown cooling shutoff valve 1E12F0010 stem appears to have separated from the valve disc. The valve is currently in the closed position and will require disassembly to rework. This valve is in the RHR system shutdown cooling (SDC) line, which takes suction from the reactor recirculation system reactor pressure vessel (RPV) suction line. The shutoff valve 1E12F0010 is the first isolation valve off of the RPV. To support maintenance of this valve, the normal configuration of the reactor recirculation system will be altered by the installation of a safety related, seismically designed plug in the N1B nozzle. Temporary Modification (TM) Engineering Change Package (ECP) 09-0244-001 will be initiated to evaluate the installation of a safety related, seismically qualified, N1B reactor nozzle plug as the reactor recirculation system isolation to prevent drain down of the water from the RPV. This activity will be performed in Perry's Refueling Outage Number 12 (RFO12) with the RPV core fully loaded with fuel. During this time, the plant will be in Mode 5 with the reactor cavity flooded and the fuel pool cooling and cleanup system providing decay heat removal. In addition, this activity will be considered as an Operation with Potential to Drain the Reactor Vessel (OPDRV) with appropriate Technical Specification actions in place. Emergency core cooling systems (ECCS) will be available for inventory control. The N1B plug will be a clearance boundary for an in-service system, the RPV, and a maintenance boundary for protection of plant maintenance personnel.

#### Description of the Modification

TM ECP 09-0244-001 and TM ECP 09-0244-002 will be initiated to evaluate the installation and removal of a safety related, seismically qualified, N1B reactor nozzle plug as the reactor recirculation system isolation to prevent drain down of the water from the RPV.

### 1.2 Summary of Evaluation

The scope of this evaluation is the use of and removal of the safety related, seismically qualified RPV N1B reactor nozzle plug completed within TM ECP 09-0244-001 and TM ECP 09-0244-002.

This evaluation addresses the use of the N1B reactor nozzle plug that was supplied as safety related, and has American Society of Mechanical Engineers (ASME) Code certifications for materials, welding and non-destructive examination. However, the plug does not meet all of the applicable requirements of the ASME Code, Section III, Subsection NC (Class 2) as required for equipment design for fuel containment pools as specified in the Updated Final Safety Analysis Report (UFSAR). This plug will be used as an isolation boundary to allow maintenance on the RHR system shutdown cooling shutoff valve 1E12F0010. This maintenance is needed to rework this valve because the stem appears to have separated from the valve disc.

The evaluation concluded that the use of this safety related, seismically qualified plug will not cause a leakage failure from the reactor vessel that could not be mitigated by the ECCS such that nuclear fuel will be continuously covered and cooled and that reactor inventory can be maintained.

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

No.