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Duane Arnold Energy Center
Docket 50-331
License No. DPR-49

Report of Facility Changes, Tests and Experiments, Fire Plan Changes, and Commitment Changes

In accordance with the requirements of 10 CFR Section 50.59(d)(2), NextEra™ Energy Duane Arnold, LLC, f/k/a FPL Energy Duane Arnold, LLC (hereafter, NextEra Energy Duane Arnold) hereby submits the subject report covering the time period from July 14, 2007 through June 1, 2009. There were no Fire Plan changes or commitment changes during this time period that require reporting. This letter makes no new commitments or changes to any existing commitments. Should you have any questions regarding this matter, please contact Licensing Manager, Steve Catron at (319) 851-7234.

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Enclosures (2)

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DESCRIPTION OF CHANGES

This section contains brief descriptions of plant design changes and procedure changes completed during the period of July 14, 2007 through June 1, 2009, and a summary of the evaluations for the changes, pursuant to the requirements of 10 CFR Section 50.59(d)(2).

00-009 ECP 1737 - Containment Oxygen Analyzer Replacement

Description and Basis of Change

This modification involved the installation of a new containment oxygen analyzer and to clarify the standby operation of the post-accident hydrogen-oxygen (H_2O_2) analyzers. The purpose of the new containment oxygen analyzer is an on-line verification of oxygen concentration to verify the status of containment inertness during normal operation. Previous practice was to have either the 'A' or 'B' H_2O_2 post-accident analyzer running continuously, with the other in standby. The new containment oxygen analyzer allows placing the post-accident containment H_2O_2 analyzers into standby, improving the life span of the post-accident H_2O_2 analyzers by running them less and reducing maintenance costs. The new containment oxygen analyzer is connected to the sample lines in the primary containment air sample system. The oxygen analyzer performs the non-safety function of the Containment Atmosphere Monitoring System portion of the Containment Atmosphere Control System.

Evaluation Summary

The safety function of the Containment Atmosphere Control System that is specific to the post-accident H_2O_2 Analyzers is monitoring the oxygen and hydrogen concentrations in the containment atmosphere post-accident. The non-safety function of the Containment Atmosphere Control System is to maintain the inert concentration of containment and to monitor the radioactivity of the containment atmosphere. The safety functions provided by the post-accident H_2O_2 Analyzers were unaffected by these activities. The containment atmosphere monitoring portion of the Containment Atmosphere Control System is a passive system with no automatic safety actions. Therefore, there were no credible ways of increasing either the probability of occurrence of an accident or the consequences of any of the accidents evaluated in the SAR. The new containment oxygen analyzer is separated from the post-accident H_2O_2 analyzers and provides indication only. There are no credible failures that could increase either the probability of occurrence or the consequences of a malfunction of equipment important to safety as evaluated in the SAR. The installation location of the new containment oxygen analyzer did not introduce any new failure modes for the post-accident H_2O_2 analyzers. There are no credible failures that could create the possibility of an accident not previously evaluated or increase the possibility of malfunction to any equipment important to safety not previously evaluated. There was no possibility of reducing any margin to safety as defined in the basis of any Technical Specification. Based upon this evaluation, prior NRC approval for this change was not required.

06-002 ECP 1748- 'B' Emergency Diesel Generator Governor Replacement

Description and Basis of Change

This modification involved replacement of the 'B' Emergency Diesel Generator (EDG) governor and automation of certain testing functions of the EDG. This change was performed to replace obsolete equipment, thereby improving the EDG reliability, and to minimize unavailability by automating certain functions during slow start testing.

Evaluation Summary

The EDG is not an initiator of any accident and therefore does not increase the likelihood of a previously evaluated accident or create the possibility of a different type of accident. Although additional equipment was installed under this change, the failure of any of this additional equipment did not result in more than a minimal increase in the likelihood of a malfunction of equipment important to safety because, in general, the equipment replaced manual operator actions with automatic actions and the new components are at least as reliable as the previous components, which were obsolete and would have been difficult to maintain in the future. This modification did not change the consequences of an EDG failure. Since the likelihood of a malfunction of the EDG was not increased by this change, there was no increase the consequences of any accident. The change did not create the possibility of a malfunction with a different result because the only malfunction that could occur is a failure of the EDG which was already evaluated. No common cause failures were introduced by this change. No Design Basis Limit for a Fission Product Barrier (DBLFPB) was affected by this change because EDG reliability and availability to support front line safety systems that directly protect the DBLFPBs were not degraded by this change. The change did not involve any method of evaluation. Based upon this evaluation, prior NRC approval for this change was not required.

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5059Eval022686 Operation At 1912 MWth With Less Than 5% Excess Feedwater Flow Margin

Description and Basis of Change

This change allowed operation of the plant up to 1912 MWth with less than 5% excess feedwater flow margin previously reserved for response to reactor steady state water level transients. This excess flow margin is referred to as 5% Operational Runout Limit (ORL). The Feedwater System and associated Reactor Water Level Control System have no preventive or mitigative safety design bases function other than its piping integrity requirement. The Feedwater System Power Generation design bases is to provide a reliable supply of feedwater at the temperature, pressure, quality and flow rate as required by the reactor during normal operation. The Feedwater system is a non-safety related system and performs only one "credited" safety design basis function (piping system integrity performs function of Fission Product Barrier). This safety design basis function was not affected by this change. The 5% excess flow margin establishes the Feedwater System's capability to respond to expected reactor water level transients. While the Feedwater System's 5% excess flow margin is utilized during normal steady state operation, routine operation utilizing up to 2% of this margin to provide necessary feedwater flow to achieve 1912 MWth is a change from previous operating practices.

Evaluation Summary

Review of the UFSAR accident and transient analysis revealed that all accidents and transient analyses remain intact and are bounded. No new failure mechanisms were introduced. Equipment was analyzed up to and including 115% rated feedwater flow as described in the UFSAR and the design basis was not changed as a result of this evaluation. Operating with a reduced feedwater excess flow margin does not increase the frequency of occurrence of any accident previously evaluated in the UFSAR. The only credited safety design functions of the Feedwater System are as a pressure boundary and as a fission product barrier. Since this change had no impact on these functions there was no change in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the UFSAR. The Feedwater System's reduced capability to provide make-up to the vessel on decreasing water level transients has the potential to challenge the Reactor Protection System (RPS) and the Emergency Core Cooling Systems more frequently, but does not increase the likelihood of occurrence of a malfunction of these systems. This change did not increase the consequences of any accident or transient previously evaluated in the UFSAR. The Feedwater System is not credited for preventing or mitigating any accidents or operational transients, and operating with a reduced excess feedwater flow capacity does not impact any SSC credited in the UFSAR for mitigating or containing radiological consequences. There were no changes to the consequences of a malfunction of an SSC important to safety previously evaluated, and this activity did not create any possibility for an accident of a different type than previously evaluated. A reduced excess feedwater flow capacity does not create any new failure modes. This activity did not create any new possibilities for malfunction of an SSC important to safety with a different result than any previously evaluated. Reduced excess feedwater flow capacity is bounded by the loss of all feedwater and the trip of a single feedwater pump with respect to initiation of any accident or operational transient. Additionally, feedwater flow capacity has no direct impact on any SSC malfunction. This activity did not result in a DBLFPB being exceeded or altered. All accident and transient conditions in the UFSAR remain intact and bounding as there were no changes in any of the initiating threshold values. The piping system integrity and function as a fission product barrier was not affected by this change. No new or different calculations or methods of evaluation were employed. Based upon this evaluation, this activity did not require prior NRC review and approval.

5059Eval022769 Revision to Evaluation 07- 001- Revision Of Refueling Procedures (RFPs) 110, "Reactor Pressure Vessel Disassembly" and 210, "Reactor Pressure Vessel Reassembly"

Description and Basis of Change

This revision of the evaluation incorporated enhancements documented during a self-assessment. This revision is not related to any physical or procedural changes from that originally evaluated. The changes to RFP 110 and 210 included the use of a new tensioning system for the detensioning and tensioning of the reactor vessel studs using a reduced pass process. In addition, the changes to RFP 210 included increasing the reactor vessel stud elongation acceptance tolerance range. These changes reduced the overall length of refueling outages and the radiation exposure received by the reactor head assembly and disassembly work crew.

The additional weight of the new tensioners affects the tensioner strongback for lifting the tensioners into place for the Reactor Pressure Vessel (RPV) de-tensioning and tensioning activities. A calculation demonstrated that the strongback was adequate "as is" for the heavier lift. The strongback was load tested and inspected per ANSI-N14.6-1978 prior to lifting the additional weight of the new tensioners. In addition, a calculation demonstrated that the dropping of the tensioners on the refuel floor is acceptable for the plant. The calculation assumes a weight of 3000 lbs for each tensioner which bounds the new

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tensioner weight. The changes did not affect the design basis, function, or operation of the tensioner strongback.

Evaluation Summary

The reactor vessel flange/stud connections are part of the reactor pressure vessel (RPV) boundary. This activity did not adversely affect the vessel pressure boundary. While the tensioning and de-tensioning procedure change did permit the reactor vessel studs to have a larger preload stress than had been previously permitted, the closure flange and studs were demonstrated to meet applicable ASME Code stresses and fatigue usage limits, so there was no change to the design basis of the RPV. The additional weight of the new tensioners on the tensioner strongback for lifting the tensioners over the vessel head for tensioning and de-tensioning was evaluated and it was determined the strongback was adequate "as is" for the heavier lift. The strongback was satisfactorily load tested and inspected per ANSI-N14.6-1978 prior to lifting the additional weight of the new tensioners. The changes did not increase the frequency of occurrence of an accident previously evaluated in the UFSAR, and they did not increase the likelihood of occurrence of a malfunction of any SSC important to safety previously evaluated in the UFSAR. The RPV studs have no active role during any hypothesized accident other than to seal the reactor vessel head to the reactor vessel shell. The changes meet the original design and construction standards as applicable to the RPV System. Therefore, the RPV closure flange will remain functional during all accidents considered in the UFSAR. The consequences of an accident previously evaluated in the UFSAR were not increased, and the consequences of a malfunction of a SSC important to safety previously evaluated were not increased. Implementation of the reduced pass detensioning/tensioning process and new elongation acceptance criteria creates no new credible failure modes. Implementation of the new tensioners did not create new credible failure modes. The possibility for an accident of a different type than previously evaluated in the UFSAR was not created, and the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR was not created. The DBLFPB as described in the UFSAR was not exceeded or altered. This change did not result in a departure from the method of evaluation described in the UFSAR used in establishing the design bases or in the safety analysis for the reactor vessel flange, studs or tensioner strongback. No activity requiring prior NRC approval was identified.

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5059Eval028746 ECP 1805 - Reactor High Pressure Scram Pressure Switch Replacement

Description and Basis of Change

This modification replaced the Reactor High Pressure Scram instrumentation input to the Reactor Protection System (RPS). The Barksdale pressure switches were replaced with analog trip instrumentation that consists of a Rosemount pressure transmitter feeding a 4 to 20 mA signal to a Rochester electronic alarm unit. Replacement of these components was required to resolve long standing equipment issues.

Evaluation Summary

This activity did not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated. The Reactor High Pressure Scram function is not an initiator of any accident described in the plant SAR. It is used to mitigate the affects of an accident/transient involving a reactor overpressure condition once the accident/transient has initiated. The likelihood of occurrence of the malfunction was not more than a minimal increase for the following reasons;

1) The replacement equipment was qualified to meet the Seismic, Environmental, and electromagnetic interference/radio-frequency interference (EMI/RFI) requirement for this application. The new equipment was also installed in accordance with approved specifications. These qualifications ensure the equipment will continue to operate properly during all required conditions.

2) Even though this modification installed four pieces of equipment in place of a single pressure switch, equipment reliability was expected to increase. The Barksdale pressure switches had several issues that ultimately resulted in them being considered unreliable. Numerous false alarms and trips signals were being generated by the Barksdale pressure switches. Operating experience at DAEC and industry OE with the Rosemount pressure transmitter and the Rochester alarm unit has been very favorable. There are very few documented failures associated with the replacement equipment.

3) The use of analog instrumentation introduced several potential new failure mechanisms which included loss of signal from the transmitter to the alarm unit, signal from the transmitter to the alarm unit fails abnormally high, loss of power supply, loss of EMI filter. However, these new failure mechanisms were bounded by the existing failure analysis for the reactor high pressure logic and no new

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unanalyzed failure modes were created.

By verifying the replacement instrumentation meets the equipment accuracy requirement and time response requirements the analysis for all transients involving pressure changes was validated. Therefore, there was no change to the consequences of an accident as described in the UFSAR. This change did not introduce the possibility of a change in the consequences of a malfunction because the RPS Reactor High Pressure Scram is not an initiator of any accident and no new failure modes were identified that were not bounded by previously evaluated failure modes. This change did not introduce the possibility of a new accident, and it did not introduce the possibility for a malfunction of RPS Reactor High Pressure Scram with a different result because the activity did not introduce a failure mode that was not bounded by those already described in the plant SAR for the RPS. By verifying the replacement instrumentation met the equipment accuracy requirement and time response requirements the analysis for all transients involving pressure changes was validated. This ensured that the design bases limit for a fission product barrier would not be exceeded or altered. The RPS Reactor High Pressure Scram calculation was revised in accordance with the Instrument Setpoint Guide. This Guide meets the UFSAR specified Instrument Setpoint Methodology NEDC-31336 (General Electric Instrument Setpoint Methodology). The Analytical Limit, Allowable Value, and Trip Setpoint used to establish the design bases for the Reactor High Pressure Scram function input to RPS and Low Low Set was not changed. Therefore, this activity did not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases. Based upon this evaluation, prior NRC approval for this change was not required.

5050Eval033022 ECP 1868 - Increased Core Flow To 105%

Description and Basis of Change

This activity allowed an increase in core flow from the rated value of 49 Mlb/hr (100%) to a new maximum value of 51.45 Mlb/hr (105% of rated) at any time during normal plant operation. This allowed an expansion of the operating domain to include an Increased Core Flow (ICF) Region on the power to flow map. The increased core flow capability allows for more efficient operation by providing a larger flow window at rated core power. This window allows for reactivity compensation with fuel burn-up through the operating cycle, ease in achieving desired control rod patterns, and fuel cycle extension (full-power days). ICF operation was evaluated for impact on current equipment operation, plant response to normal and abnormal events, and long-term impacts such as thermal cycle fatigue and flow-induced vibration on components. These evaluations were coordinated with the development of the Core Operating Limits Report (COLR).

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These evaluations identified the key SSCs important to safety that were impacted by ICF operation as: Reactor Recirculation System, Reactor Pressure Vessel (including nozzles, internals and attached piping), and Nuclear Fuel and Primary Containment.

Evaluation Summary

The evaluation did not identify any new failure modes of those SSCs important to safety or an increase in likelihood of those failures, as no equipment is operated outside of its design limits. All SSCs remain within allowable stress and fatigue limits. Fuel operating limits were adjusted in the COLR to accommodate the transient and accident response with ICF. Evaluation of plant accidents did not identify any with consequences more severe than previously evaluated nor were any design basis limits exceeded. Except for the Reactor Internal Pressure Difference (RIPD) evaluations for the steam dryer, all the other evaluations were performed with the same methods of evaluation as previously used. Because the upgraded method for calculating the steam dryer RIPD gave a more conservative result than the previous method described in the UFSAR, it did not constitute a departure from a previously approved method. No new design basis limits were created as a result of this activity. Plant events/accidents are not made more likely by ICF, as no equipment is operated outside of its design limits. No new or different accidents or malfunctions of SSCs important to safety were identified. Consequently, the implementation of the ICF domain did not require prior NRC review.

5059Eval034202 Revision To Evaluation 01-025 - ECP 1628 - Reactor Building Ventilation Shaft and Control Building Air Intake Radiation Monitors Replacement

Description and Basis of Change

During a NRC assessment of the NextEra Energy Duane Arnold evaluation process, several comments/improvements to this evaluation were provided. This revision incorporated the comments into the evaluation. Also, a Yokogawa series DX100 recorder was installed in place of RR7606A(B). The evaluation was updated to reflect the correct recorder model installation.

This modification replaced analog radiation monitoring systems for the Reactor Building Vent Shaft (RBVS) and the Control Building Air Intake (CBAI) with digital radiation monitoring systems. The instruments were obsolete and required upgrade. The new Sorrento radiation monitors were implemented using the guidance of NRC Generic Letter 95-02, which informs licensee of the NRC's position on the use of EPRI Report TR-102348; "Guideline on Licensing Digital

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Upgrades," as acceptable guidance for determining when an analog-to-digital replacement can be performed without prior NRC approval under 10 CFR 50.59.

Evaluation Summary

The new detectors and monitors for RBVS and CBAI radiation monitoring systems by themselves can not cause an accident. The design functions of RBVS and CBAI radiation monitoring systems were not altered by this activity. The probabilities of occurrence of an accident evaluated previously in the UFSAR were not adversely affected by this modification. This activity did not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR. This digital upgrade did not introduce any new failure modes. The software common mode failure is addressed by the use of a Structured Software Development, and a watchdog circuit, which places the unit in an alarm/trip condition upon a non-self-evident failure (lockup) of the processor or the software. All new components were procured safety related, Class-1E, and were seismically installed. The non-safety related circuits were appropriately isolated from the safety related circuits with the use of Class-1E isolators. Therefore, this activity did not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety. The function of RBVS and CBAI radiation monitoring systems to monitor radioactivity levels and initiate appropriate response was unchanged by this modification. Therefore, this activity did not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR. The SSC important to safety associated with the RBVS radiation monitoring system is primary and secondary containment isolation instrumentation, valves and dampers. The requirement of automatic closure of appropriate primary and secondary containment isolation valves and dampers to limit fission product release to the environment upon detection of abnormal radiation level in RBVS exhaust was not adversely impacted by this modification. For CBAI radiation monitoring system, the SSCs important to safety are Standby Filter Unit (SFU) and associated dampers. The requirement of automatic initiation of SFU and dampers upon detection of abnormal radiation level in CBAI plenum was not changed by this modification. This activity did not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR. This activity did not create a possibility for an accident of a different type than any previously evaluated in the UFSAR. The digital upgrade recommendations of EPRI reports TR-106439 and TR-102348 were observed. Common mode software issues were evaluated. This activity did not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR, and it did not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered. This modification did not involve a change to

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any method of evaluation. No activity requiring prior NRC approval was identified by this evaluation.

5059Eval036196 ECP 1830 - Recirculation System Motor Generator Scoop Tube Positioner Replacement

Description and Basis of Change

The Recirculation System Motor-Generator (MG) Scoop Tube Positioner utilized a Bailey actuator that was obsolete, and spare replacement parts were becoming increasingly difficult to obtain. In addition, three main issues affected the actuator; Scoop Tube Lockups, Spurious Activation of the Deviation Relay, and Inadvertent Recirculation Speed Changes/Runbacks.

This modification replaced the degraded, obsolete equipment, and improved plant performance by reducing the number failures with the scoop tube. This modification included the following changes:

- Replaced the Moore 352 Controller with a newer Siemens-Moore 353 Controller, and included an Action Instruments Signal Isolator for the command signal.

- Removed the MG Set Scoop Tube Deviation Relay, and Deviation Meter.

- Provided a comparison of the controller output to the actuator position feedback so if the controller output deviates much greater than the position feedback, the scoop tube lock relay de-energizes.

- Control signal cables were modified to have the associated shield wire tied to ground.

- Replaced the Bailey Positioner for the MG Set Fluid Drive with a Jordan Controls Actuator and associated electronic controls.

- Removed the feedback function of the MG Set Generator Tachometer from the Recirculation Flow Control System Controller. This changed the Recirculation Speed Control scheme from closed loop to open loop.

- Set the Mechanical Stops for the Scoop Tube Positioner from 102.5% to 107%. Procedural controls were put in place to ensure that the core power and core flow limits of the Extended Power Uprate evaluations of Single Loop Operation (SLO) would not be exceeded.

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Evaluation Summary

Based upon the changes made there is no greater frequency of occurrence of an accident evaluated in the UFSAR. Previously analyzed failure modes were not affected by this modification. The recirculation pumps are not considered essential for safe plant shutdown under either normal or abnormal operations. The Reactor Recirculation System (RRS) is not required to operate after a design basis accident. The RRS is designed to meet safety design bases related to margins for fuel temperatures, maintaining pressure integrity of the primary coolant boundary, and piping configuration provisions for re-flooding of the RPV for a pipe break accident. None of the safety design bases were affected by this modification. Tripping the recirculation pumps is used to mitigate the consequences of a common mode failure to scram when needed. Recirculation pump trip is also used to improve thermal margin for limiting thermal transients near the end of core life. This modification did not affect the Recirculation Pump Trip logic. There was not more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR. The new controllers are located in the same location and orientation as the previous devices and were mounted to meet seismic 2 over 1 requirements, so there was no increase in the consequences of a SSC malfunction. This modification did not affect the pump start and stop logic, and it did not affect reactor pressure boundaries. No new types of accidents or malfunctions were created. The change in runback rate did not impact the safety analysis, as the assumed conditions are still bounding with the changed runback rate. No DBLFPB was exceeded due to this modification, and this activity did not result in a departure from a method of evaluation. No activity requiring prior NRC approval was identified.

5059Eval048254 Revision 2 to Evaluation 06-001 - ECP 1720 - Steam Leak Detection System Temperature Monitoring Riley Module Replacement

Description and Basis of Change

This evaluation was revised to incorporate information identified during a 50.59 self assessment. This modification replaced the instrument and control functions of the Riley/Panalarm modules of the Steam Leak Detection System (SLDS) Temperature Monitoring System, that provides indication from the High Pressure Coolant Injection System (HPCI), Residual Heat Removal System (RHR), Reactor Core Isolation Cooling System (RCIC) and Reactor Water Cleanup System (RWCU) area temperatures and differential temperatures to the main control room, with devices that perform the same functions, but are readily available, less expensive, use more modern technology, and have vendor support. This modification also replaced the slide link disconnect terminal blocks

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for the associated thermocouples with more readily available, less expensive, and more durable equivalents.

Evaluation Summary

There are no previously evaluated accidents that can be caused or initiated by this activity. A failure of the Steam Leak Detection System could cause or prevent the isolation of HPCI, RCIC, or RWCU, however, this is not considered an accident. This activity did not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated. The new instrumentation meets the required plant environmental and seismic envelopes. The failure of the network of the plant process computer cannot affect the recorder in a way that has not been analyzed. The electrical loads are increased as the recorders draw more current than the Riley modules. This increase is evaluated as minimal within the analyzed design parameters of the supporting AC and DC systems. The increase in heat load of the control room HVAC is evaluated as minimal and within the analyzed design parameters of the supporting HVAC systems. This modification maintains and meets the requirements for separation, independence, and grounding. There is reasonable assurance that the dependability of the system is sufficient and that the likelihood of a common mode failure is significantly below that of any single, active failure. This activity does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated. This system indirectly contributes to accident mitigation by isolating steam leaks outside of the primary containment and thus reducing the amount of released radiation in the associated area in the instance of a leak. The SLDS is an auxiliary system required to support the mitigation of the unacceptable consequences of a Loss of Coolant Accident (LOCA) inside containment, a LOCA outside containment, and a Pressure Regulator Failure Open. This change did not result in more than a minimal increase in the consequences of an accident previously evaluated, and it did not result in more than a minimal increase in the consequences of a malfunction of a SSC important to safety previously evaluated. Although the change did affect the SLDS which supports mitigation of accidents, the results of a malfunction of the SLDS did not change as a result of this activity. This change was bounded by existing failure analyses and assessments of system-level failure modes. No new types of system-level failure modes that could cause a different type of accident were identified. This change involved combining separate functions into one digital device; however, a failure does not create a result unbounded by the results of malfunctions previously considered. This upgrade did not involve a change to any element of the analytical methods described in the UFSAR which are used to demonstrate the design meets the design basis or that the safety analysis is acceptable. This change did not involve use of a method of evaluation not already approved by

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the NRC. No activity requiring prior NRC approval was identified by this evaluation.

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Tests and Experiments

This section contains a brief description of Tests completed during the period from July 14, 2007 through June 1, 2009.

5059EVAL033022 Special Test Procedure (SpTP) 213 - Increased Core Flow and Power Ascension Test To Greater Than 1880 MWth

Description and Basis of Change

This SpTP was performed for testing each loop of the Reactor Recirculation System. The test was performed prior to exceeding 49.0 Mlb/hr and increased one loop with the other loop at reduced flow. This test allowed identification of Reactor Recirculation System limiting components. Various reactor recirculation equipment parameters were monitored and utilized for implementation and provided adequate monitoring to assume the equipment is operable in the safety analysis. Prior to implementation of increased core flow (ICF), this special test provided increased monitoring of the plant equipment as core flow was increased in small increments to ensure that the plant equipment operated within expected ranges.

Evaluation Summary

The evaluation did not identify any new failure modes of those SSCs important to safety or an increase in likelihood of those failures, as no equipment was operated outside of its design limits. All SSCs remained within allowable stress and fatigue limits. Evaluation of plant accidents did not identify any with consequences more severe than previously evaluated nor were any design basis limits exceeded. A departure from a previously approved method of evaluation was not identified. No new design basis limits were created as a result of this activity. No new or different accidents or malfunctions of SSCs important to safety were identified. No activity requiring prior NRC approval was identified by this evaluation.

01-009 SpTP 214 - Pressure Regulator Dynamic Tuning

Description and Basis of Change

The Power Uprate Project (PUP) increased the output of DAEC from 1658 MWth to 1912 MWth. This special test verified the tuning of the Electro-hydraulic Control (EHC) parameters and demonstrated the EHC System response to pressure transients and regulator failure was acceptable during power ascension. The information obtained from performing SpTP 214 verified instrument settings

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and proved operability. This testing was required as a result of the PUP Task Report (TR) 1005 "Startup Test Specifications," and the Turbine Control System modification. This Special Test coordinated all activities, such that they would not create new types of events and ensured that additional data was recorded above that normally documented during routine operation. This test used permanently installed plant monitoring equipment/instrumentation as well as "non intrusive" recording/data gathering techniques. The turbine generator Original Equipment Manufacturer (OEM), General Electric (GE) issued a Service Information Letter (SIL) 589, Revision 1 in February 1996 for the Main Turbine EHC System. This SIL contained EHC tuning and adjustments essential for proper operation and transient response. PUP Project TR 1005 was issued to generate Startup Test Specifications for Nuclear Steam Supply System and Balance of Plant System tests necessary for the confirmation of acceptable plant performance for operation at uprated power levels to 1912 MWth. The TR recommended confirmation of the dynamic tuning parameters for the system. These tuning parameters are stated in SIL 589, Revision 1. SpTP 214 demonstrated proper transient operation of the Main Turbine Pressure Regulation System as referenced in the SIL.

Evaluation Summary

This SpTP demonstrated proper transient operation of the Main Turbine Pressure Regulation System. This test did not increase the probability or consequences of any accident because no systems designed to mitigate any accident were affected, and the EHC testing that was performed did not affect any system that initiates any evaluated accident. This test did not increase the probability or the consequences of a malfunction of equipment important to safety because no equipment important to safety was adjusted or tested under this procedure. This test did not create the possibility of a different type of accident or malfunction because the Turbine Control System cannot create a different type of accident than those already evaluated and this system is not taken credit for in the mitigation of any accident or transient. No activity requiring prior NRC review was identified.