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Perry Nuclear Power Plant
Docket No. 50-440
Report of 10 CFR 50.59 Evaluations for 2003 - 2005

Ladies and Gentlemen:

Pursuant to 10 CFR 50.59(d)(2), enclosed is the report of facility changes, tests, and experiments for the Perry Nuclear Power Plant (PNPP). The changes, tests, and experiments reported are for the period from the last submittal dated September 15, 2003 to the present.

Attachment 1 defines the acronyms and format description. Attachment 2 provides the summaries of the 10 CFR 50.59 Evaluations.

If you have questions or require additional information, please contact Mr. Henry L. Hegrat, FirstEnergy Nuclear Operating Company, Fleet Licensing Supervisor, at 330-315-6944.

Very truly yours,

Attachments:
1. Format Description
2. 10 CFR 50.59 Evaluation Summaries

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III
State of Ohio

IE47

FORMAT DESCRIPTION

Evaluation No.:

A sequentially assigned number from one (00001) to the end of the period, preceded by the year, e.g., 03-0025. Note that every number does not necessarily have an Evaluation associated with it.

Source Document:

Common sources of 10 CFR 50.59 Evaluations, with the associated acronym, are listed below.

ECP – Engineering Change Package
NOP – Nuclear Operating Procedure
TM – Temporary Modification
TSB CR – Technical Specification Bases Change Request
TXI – Temporary Test Instruction
USAR CR - Updated Final Safety Analysis Report Change Request

1.1 Activity Description:

A short narrative describing the change.

1.2 Summary of Evaluation:

A summary of the 10 CFR 50.59 Evaluation responses and conclusions associated with the eight (8) criteria contained in 10 CFR 50.59(c)(2).

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

A simple response indicating if the 10 CFR 50.59 Evaluation required the performance of a license amendment pursuant to 10 CFR 50.90.

PERRY NUCLEAR POWER PLANT
10 CFR 50.59 EVALUATION SUMMARY
PURSUANT TO 10 CFR 50.59(d)(2)
2003 – 2005

Evaluation No.: 02-01338

Source Document: USAR CR 04-085

1.1 Activity Description

This activity revises the lighting description for the Emergency Service Water (ESW) Building in USAR Section 9.5.3.2.2. The lighting in the ESW building is not powered from the diesel-backed stub-bus. Therefore, the ESW building is not provided with "essential" lighting. The USAR CR eliminates the term "essential" lighting.

1.2 Summary of Evaluation

The USAR accidents were reviewed with respect to the effects of the proposed change. The ESW system is a key mitigating system for accidents and transients evaluated in the USAR since it provides heat removal to the ultimate heat sink and dilution water for radwaste tank rupture accidents. However, the ESW system and its supplied heat exchangers are not considered initiators of any USAR evaluated event. Furthermore, based on the lack of adverse failure modes or effects, there is no potential that any new failure mechanism could cause the initiation of an USAR described event. None of the accident frequencies were found to be affected by the implementation of the proposed change because the change has no effect on accident initiators/transient.

This evaluation analyzes USAR described design functions that are potentially affected by this USAR change request. Both direct and indirect effects of the proposed change on the design functions were evaluated and, no new failure modes and effects are created by this activity. All potentially affected design functions were found to be satisfactorily performed, such that the likelihood of malfunctions of equipment was not increased. Compliance to applicable ASME Codes, IEEE/NEMA/ANSI standards and equipment qualification requirements is maintained for ESW system components.

The proposed 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 change does not result in increased dose consequences that might impede required actions inside or outside the control room to mitigate the consequences of reactor accidents.

Based on the scope of the proposed change and an evaluation of possible failure effects of the change, no new events of a significance to be considered an accident, nor new malfunctions of a System, Structure, and Component (SSC) important to safety, could be identified.

This evaluation analyzes the effects of the proposed change on the Perry Nuclear Power Plant (PNPP) fission product barriers. No effects were identified.

The change does not involve a change to or departure from a method of evaluation described in the USAR, or used in establishing the Design Basis or Safety Analysis.

The methods of evaluation that support the proposed change are consistent with the methods used in establishing the design bases and the safety analyses documented in the USAR.

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

No.

Evaluation No.: 03-00748

Source Document: Calculation 2.4.6.14, Calculation 2.4.6.15
USAR CR 03-040, TSB CR 03-039

1.1 Activity Description

This activity revises the design basis analysis that establishes the analytical limit for the Turbine Building Leakage Detection System high temperature set point. This analysis is revised in order to establish more accurate modeling of the Turbine Building using a different computer code, GOTHIC. The more accurate model was developed in part to respond to issues associated with the simplicity of the model in the existing analysis.

The major change associated with the revised analysis is the requirement to establish a new main steam leakage limit on which the Turbine Building Leakage Detection System temperature analytical limit and set point are based. However, the Turbine Building Leakage Detection System high temperature isolation set point and analytical limit are not changed by this activity. The new Turbine Building leakage limit is established at 32.9 lbm/sec versus the existing 25 gpm (2.94 lbm/sec).

1.2 Summary of Evaluation

An analysis was performed to determine the Turbine Building temperature assuming a main steam leak of 25 gpm (2.94 lbm/sec). This analysis was performed using the computer code COMPARE. The COMPARE model of the Turbine Building was a simplistic two-volume model that did not specifically account for the effects of heat sinks, ventilation, or seasonal temperature variations. It was determined that this simplistic model should be replaced with a more realistic model. This evaluation addresses the change in the design basis resulting from the new analytical techniques used to evaluate the Turbine Building temperature response to main steam leakage.

This activity impacts the design and licensing bases for the system that has been reviewed against the licensing requirements of the existing leakage detection system as defined in the PNPP USAR and the Safety Evaluation Report (SER). In addition, the change was reviewed against the original system design requirements as defined in various PNPP specific documents and generic industry standards. Analyses have been performed to determine the impact of the change on offsite dose consequences as well as maintaining main steam system integrity. The results of these evaluations indicate that the consequences of the higher leakage limit remain bounded by the current licensing base release from a main steam line break. Additionally, the system has been shown to isolate soon enough to avoid the degradation of the leak to a point where main steam line integrity is jeopardized. Based on the results of these evaluations it has been determined that this activity is acceptable and can be performed within the auspices of 10 CFR 50.59.

Specifically, the revised analytical methods and higher leakage limit do not alter the current function of the leak detection system that isolates the main steam system prior to the leakage degrading to a point where the system integrity is challenged such that a large steam line break would occur. Therefore, there is no increase in the frequency of occurrence of an accident resulting from this change.

The results of this activity are shown to be bounded by the existing acceptance criteria in that system integrity and reactor vessel makeup capability is not challenged by the increased leak rate. The current temperature analytical limit and set point are not changed by this activity and thus the Turbine Building environment remains unchanged by this activity. Therefore, this activity does not increase the likelihood of occurrence of a malfunction of an SSC.

Since the revised analytical results have been shown to be bounded by the existing analytical results, there is, therefore, no increase in the consequences of any previously analyzed accident or malfunction of an SSC because of this change.

It has been demonstrated that the revised analytical methodology maintains the main steam line isolation at a point where the leak does not propagate into a large break. In addition, since the current instrument temperature set points are not changed, the resulting Turbine Building temperature and steam environment is not impacted. Therefore, the possibility of new accidents or malfunctions of SSCs is not created by this change.

The revised analytical methodology has been shown to allow the leak detection system to continue to isolate the main steam system. The isolation time ensures that the radiological consequences of the leak remain bounded by the current main steam line break analysis. In addition, the main steam system pressure retaining capability is maintained. That is, the isolation of the main steam system has been shown to occur prior to the crack reaching its critical length, and thus the leak does not propagate into a full steam line break. Therefore, the design basis limit for the fission barrier remains unchanged by this revised methodology.

The methodology for evaluating the Turbine Building steam leakage is not described in the USAR or in any other regulatory document. However, the new method (GOTHIC) was shown to essentially be equivalent to the old method (COMPARE). Therefore, this change is not a departure from any USAR methodology.

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

No.

Evaluation No.: 03-00823

Source Document: USAR CR 03-052

1.1 Activity Description

This activity updates the radiological effects associated with a feedwater line break. The supporting dose calculation was revised to incorporate the Technical Specification limit for continuous operation for the reactor coolant specific activity of$<0.2\mu$Ci/gm I-131 dose equivalent. The activity also notes that the effects do not account for feedwater check valve leakage and refers one to a sensitivity analysis separately approved under License Amendment 105.

1.2 Summary of Evaluation

The proposed activity was initiated due to a revision in the dose calculation associated with the Feedwater Line Break Accident Outside Containment. The evaluation concludes that the change does not increase the frequency of occurrence of an accident previously evaluated or of a malfunction of equipment important to safety. There is no potential for an accident of a different type than previously evaluated or of a malfunction of equipment with a different result. The evaluation concluded that while there was an increase in radiological consequences of an accident, there was no more than a minimal increase in the consequences of an accident or malfunction of equipment. The methodology utilized was not determined to be a departure from what was previously approved as the equations were not changed. Given that it was determined that the increase in consequences was no more than a minimal increase, no license amendment request is considered to be necessary.

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

No.

Evaluation No.: 03-01314

Source Document: USAR CR 03-077

1.1 Activity Description

The calculation that provides the basis for the radiological consequences of the Failure of Air Ejector Lines contained in USAR Table 15.7-10 was revised to correct the filtration efficiency assumption for the Off-Gas Building ventilation system. The filtration efficiency is being revised to 90%. The ventilation unit is categorized as falling under Regulatory Guide 1.140. This guide does not allow a 2" carbon bed to attain a credited efficiency of greater than 90%. The calculation was also revised to incorporate the Technical Specification Section 3.4.8 reactor coolant specific activity for continuous operation of 0.2microCi/gm I-131 dose equivalent.

1.2 Summary of Evaluation

This calculation revision resulted in not more than a minimal increase to the dose in USAR Table 15.7-10. The change does not have any impact on any hardware design, maintenance, or qualification. There are no potential SSC failures or adverse effects resulting from this change. The only change to the plant is the dose consequence of a postulated Air Ejector Line Failure. The evaluation concluded that the change does not result in: an increase in the frequency of occurrence of an accident or malfunction of an SSC important to safety, a more than a minimal increase in the consequences of an evaluated accident or a malfunction of an SSC important to safety, a possibility of an accident of a different type or malfunction of an SSC with a different result than previously evaluated, a design basis limit for a fission product barrier being exceeded or altered, or a departure from a described method of evaluation used in establishing the design bases or in the safety analyses.

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

No.

Evaluation No.: 03-01917

Source Document: ECP 01-5018

1.1 Activity Description

The Testable Rupture Disk (TRD) design installed by this Engineering Change Package (ECP) in the Division 3 Emergency Diesel Generator (EDG) combustion exhaust system represents an upgrade to the TRD design currently installed on the Division 1 and 2 EDG combustion exhaust systems. The TRD is designed to open sufficiently to pass the engine combustion exhaust flow without excessive exhaust backpressure on the engine in the event of a blockage of non safety related combustion exhaust piping/components.

The TRD is designed to open at a pressure setpoint to limit corresponding backpressure measured at the engine turbocharger. The steady state backpressure measured at the turbocharger when exhaust flows through the open TRD with the silencer flow path blocked has been evaluated to be acceptable.

1.2 Summary of Evaluation

The effects of this change have been reviewed against the causes of all of the USAR evaluated accidents. Since the causes are not affected, the frequency of occurrence of any accident previously evaluated in the USAR is not increased.

The effects of this change were reviewed against potentially affected USAR described design functions. From that review, there is sufficient evidence to conclude that the Division 3 EDG will maintain its ability to function as credited in the PNPP USAR to mitigate design basis events, including earthquakes and tornadoes. This change maintains the redundancy, independence and separation of the onsite EDGs. Based on the evaluations and analyses performed, this change does not affect the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the USAR.

Any accidents that rely on the function of the Division 3 EDG would be mitigated as evaluated in the USAR. 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, this change does not increase the radiological consequences of any accident or malfunction of equipment important to safety evaluated previously in the PNPP USAR.

This change does not cause a previously evaluated event to become categorized as an accident. This change does not make any events previously categorized in the USAR 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, this change does not create a new potential for common mode failure of the EDGs. The malfunction effects identified in the analysis section are bound by previously evaluated USAR malfunctions (loss of a single electrical division).

Based on the analysis performed, the change maintains the USAR described design functions of the Division 3 EDG. Consequently, the High Pressure Core Spray (HPCS) system will perform as evaluated previously, and there are no indirect effects on the design basis limits for any of the fission product barriers.

The analyses that establish the TRD design/qualification are consistent with the USAR descriptions. Post modification and periodic surveillance testing will confirm the proper function of the TRD. All of the analytical/empirical methods utilized are considered to be consistent with the methods described in the USAR that establish the design basis of the Division 3 EDG (i.e., Regulatory Guide 1.9, IEEE 308, IEEE 387).

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

No.

Evaluation No.: 04-00103

Source Document: ECP 02-0003

1.1 Activity Description

The refueling platform is used to safely transport fuel and reactor components to and from pool storage and the reactor vessel. The refueling platform control system is considered obsolete and in need of a major upgrade. Equipment aging has compromised reliability to the degree that significant downtime is being encountered more frequently each outage. The current inventory of spare parts is not adequate to support repairs going forward.

The proposed activity will perform a major control system upgrade to the refueling platform that will replace the existing control system and panels with a new Programmable Logic Control (PLC) based control system. The new controls will utilize a touchscreen status monitor for the operator interface versus the existing switches, lights and encoder counters. Beside the control system, the major components that are being replaced are: a) main fuel hoist including a new hoist, motor, brakes and motor drive; b) motors, gearboxes, coupling, and motor drives on the bridge and trolley; c) mechanical limit switches with a redundant encoder-based position indicating system; d) new load cell interfacing with the new control system; e) operator status console with a new operator workstation; and f) provides a new start/stop station.

1.2 Summary of Evaluation

There are three accident scenarios related to fuel handling – Control Rod Withdrawal Error During Refueling and Startup Operations, Inadvertent Loading and Operation with Fuel Assembly in Improper Position, and Fuel Handling Accident Inside Containment. As confirmed by the EMI/RFI testing, the new equipment will not create any EMI/RFI that could adversely impact other control systems. Thus, the proposed design changes do not adversely impact any SSC that could be an initiator of any USAR evaluated events. Thus, the proposed modifications will not adversely impact the frequency of occurrence of any accident.

The completed Failure Modes and Effects Analysis (FMEA), seismic qualification of replaced component attachments, and EMI/RFI testing demonstrates that the proposed modifications do not adversely affect any SSC evaluated or discussed in the USAR. There are no new system interconnections or interactions and the revised intersystem interlocks have been shown to provide an equivalent level of protection to the existing controls. Hence, the proposed activity will not increase the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the USAR.

The new control system is a digital upgrade system. Compliance to industry standards during software development and control of software per site procedures provides assurance of reliable software function. In addition, the new hoist complies with the original vendor specified capacity and safety factors. The FMEA and panel seismic attachments discussed previously assure no unacceptable actions. These factors when combined with the planned in-depth testing provide assurance against unacceptable system interactions. Hence, the proposed activity will not create the possibility for an accident of a different type than any previously evaluated in the USAR.

The refueling platform is not identified as a mitigator of any USAR events. The modifications do not adversely impact any USAR described design functions or control actions. Hence, the proposed activity will not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the USAR.

The proposed activity does not impact the refueling interlocks associated with preventing inadvertent criticality. Nor does the proposed activity involve any of the administrative controls related to the movement of non-fuel items over irradiated fuel inside containment. Hence, the proposed activity does not result in a design basis limit for a fission product barrier as described in the USAR being exceeded or altered.

There are no evaluation methodologies described in the USAR related to the refueling platform, related equipment, or associated components. Thus, this activity does not result in a departure from the method of evaluation described in the USAR used in establishing the design bases or used in the safety analyses.

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

No.

Evaluation No.: 04-00223

Source Document: ECP 03-0253

1.1 Activity Description

The SCRAM Frequency Reduction Project identified a method to prevent potential turbine trips that result in reactor scrams. The Reactor Core Isolation Cooling (RCIC) system consists of a turbine, pump, valves, accessories, and instrumentation designed to assure that sufficient reactor water inventory is maintained in the reactor vessel to permit adequate core cooling to take place. In the event the reactor vessel is isolated, and the feedwater supply is unavailable, the water level will drop due to continued steam generation by decay heat. Once the water level decreases to Level 3 an automatic reactor SCRAM will occur. Upon reaching a predetermined level (Level 2), the RCIC system will automatically initiate. The RCIC initiation logic consists of instrumentation, electronics, relays, and power supplies. Due to RCIC initiation logic component failures, the RCIC system has spuriously initiated on three occasions at PNPP. The RCIC system initiation occurred with the reactor at 100% power and reactor water at the normal operating level. The RCIC system initiation logic provides a turbine trip signal. The turbine trip signal trips both reactor feedwater pump turbines and the main turbine. This results [via Turbine Stop Valve (TSV) or Turbine Combined Intercept Valve (TCIV) fast closure] in a reactor SCRAM. This modification is being installed to prevent a RCIC logic failure from causing a plant shutdown.

A time delay relay will be utilized in the main turbine trip logic associated with the RCIC system initiation. This delay provides time for plant operators to assess plant conditions and secure the RCIC system if not required for safe plant operation. First stage turbine pressure will be utilized to automatically enable/disable the RCIC time delay function. This will ensure that sufficient steam flow is present to ensure all moisture (RCIC spray) is entrained within the steam (no water buildup at low points in the steam lines). A currently installed control switch will be utilized to manually disable the time delay if required for maintenance.

1.2 Summary of Evaluation

The USAR accidents 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 affected by the implementation of the proposed change.

This evaluation analyzes USAR 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 was reviewed. The likelihood of any malfunctions of equipment important to safety are not increased by this change.

This change does not result in increasing previously evaluated release rates, changing a 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 a significance, nor new malfunctions of an 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 USAR.

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.: 04-00301

Source Document: ECP 02-0078

1.1 Activity Description

The proposed activity will replace the existing analog Feedwater Control System (Bailey 7000 series) with an Invensys/Foxboro Intelligent Automation (I/A) Series Digital Feedwater Control System (DFWCS). The new system will minimize single failure point vulnerability internal and external to the DFWCS. The DFWCS will have redundant processors that mirror the functions of each other and if a failure is detected in the primary processor, the redundant processor will be automatically placed in service. The system uses signal validation to detect invalid Inputs and Outputs (I/Os). The system response to invalid I/Os is to use redundant signals if available, and/or to place the system in a safe condition and alert the operator of the abnormal condition.

The system consists of one Engineering Work Station (EWS), two Operator Work Stations (OWSs), two Control Processors (CPs), two Field Communication Modules (FCMs), and redundant hardware for communication networks. The DFWCS is a hierarchical system and utilizes two levels of redundant communication networks. One network is at the field bus level. This network provides for communication between the CPs and the Field Bus Modules (FBMs). The FBMs are the input/output modules. At this level, the communication is facilitated by the redundant FCMs. The FBMs, through the Termination Assemblies (TAs), provide interface with the existing I/Os. The other level of communication takes place on the Node Bus. At this level, the EWS and the OWSs communicate with the CPs.

The DFWCS hardware will be installed in control room panels 1H13-P865D, 1H13-P612A/B, and 1H13-P680. The existing feedwater control Auto/Manual stations will be removed from panel 1H13-P680 and replaced with the OWSs that will be located on panel 1H13-P680. The OWSs will include the following components. Two touch screen 20" LCD flat screen monitors will be mounted on articulating arms and installed on the top of panel 1H13-P680. Each OWS will have its own keyboard that will be stored on a sliding keyboard tray on the underside of panel 1H13-P680. In addition, a new annunciator keyboard will be installed at the location in panel 1H13-P680 where the Auto/Manual stations used to reside. The annunciator keyboard will allow the operator to select various process display screens to view on the flat screen monitors, and has LEDs that communicate alarm conditions. The EWS will be located in the shift manager's office and will include a processor, a keyboard, and a 20" LCD flat panel monitor. A printer for the EWS will be located in the printer area of the control room. Redundant power supplies will be provided for all equipment with the exception of the EWS.

The analog feedwater control system is being replaced with a digital system to improve the reliability of the control system. The analog system is obsolete, cannot be upgraded, and is no longer supported by its manufacturer (Bailey Controls). Thus, the system is increasingly more prone to cause plant challenges as well as being more difficult and expensive to maintain. Installation of the new digital feedwater control system will reduce the likelihood of plant transients and reactor scrams initiated from feedwater level controller failures. It will also reduce the operator workload and challenges that result from the limited automation provided by the existing system. The overall purpose is to obtain improved nuclear safety by minimizing plant challenges.

1.2 Summary of Evaluation

Installation of the new DFWCS improves the reliability of the feedwater 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 new DFWCS minimizes failures or malfunctions of the DFWCS, which in turn minimizes other SSC failures or malfunctions since adverse effects will not be propagated from the DFWCS to other interfacing systems. The new system also minimizes the occurrence of USAR 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 Feedwater Controller Failure – Maximum Demand and Loss of all Feedwater events is not increased.

The licensing basis does not rely on the feedwater control system for mitigation of accidents or malfunctions. Consequently, installation of a new digital feedwater 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 USAR 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 DFWCS and due to its EMI/RFI qualifications that prevent any adverse interactions with other systems. Protection of the DFWCS 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. The flat screen displays mounted on the articulating arms on panel 1H13-P680 are qualified to prevent fall-down in a seismic event and thus a malfunction with a different result cannot occur during a seismic event.

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.: 04-00316

Source Document: ECP 02-0212

1.1 Activity Description

A bypass to the reactor water Level 3 scram will be provided by 4 keylock control switches (1C71A-S10A-D) located on panels 1H13P0691-P0694 (one control switch per panel). The switches are type GE CR2940 key removable in 'NORMAL' position. The key is not removable in the 'BYPASS' position. These control switches will be positioned to the 'NORMAL' position during plant operation. These control switches will have no affect on the plant when positioned to 'NORMAL'. If any of these switches are taken to the 'BYPASS' position, annunciator 1H13P680-7A C2 will alarm to alert the operator of the switch position. The 4 control switches will have no affect on Reactor Protection System (RPS - C71) logic circuits unless the mode switch is in the 'SHUTDOWN' position. A bypass of the reactor vessel low water level trip is provided when the keylock switches in the 'BYPASS' positions and the mode switch in the 'SHUTDOWN' position. The interlock with the mode switch will ensure that the reactor is in the shutdown condition prior to bypassing the reactor water Level 3 scram. The RPS reactor water Level 3 scram function is required during plant power operation.

1.2 Summary of Evaluation

The USAR accidents 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 affected by the implementation of the proposed change. This evaluation analyzes USAR 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 was 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 USAR.

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.: 04-00335

Source Document: ECP 02-0022

1.1 Activity Description

The Inclined Fuel Transfer System (IFTS) is being upgraded by this modification package. The objectives of this upgrade are to improve equipment reliability, speed refueling and reduce operating labor requirements. Also, many of the components currently used in the equipment are now obsolete, making spare parts difficult to obtain. PNPP's IFTS is being modified and upgraded to ensure reliable operation during refueling outages and to enhance operation with state-of-the-art components.

The upgrades include replacement of the existing PLC based control system and panels with a new PLC based control system. The new controls will utilize a touchscreen status monitor for the operator interface versus the existing switches, lights and encoder counters. New operating stations are added to the fuel handling bridge and the refueling bridge to allow automatic transfer initiation and monitoring of IFTS from each bridge. The winch motor and electric brake are being replaced with equivalent replacement parts. The winch load weighing system is being upgraded and the carrier position sensing system is revised to minimize the use of proximity sensors in favor of redundant encoder based positions. The winch protective cover is being replaced with a cover having improved maintenance access. Personnel access controls into IFTS maintenance rooms are being revised to facilitate testing and maintenance while providing the required level of radiation protection for very high radiation areas. In addition, one electrical penetration power supply is revised to provide protection of the penetration per Regulatory Guide 1.63. As part of the IFTS upgrade, two unused Visual Area Monitoring consoles are being deleted for better IFTS control panel access.

1.2 Summary of Evaluation

There are two accident scenarios related to fuel handling – Fuel Handling Accident Outside Containment and Fuel Handling Accident Inside Containment. The IFTS is not an initiator of any of these accidents. The revised protective interlocks between IFTS and the fuel handling and refueling bridges continue to provide assurance against possible collisions that could damage the masts or their loads. In addition, the IFTS tube that forms part of the containment fission product barrier is not altered by the proposed modification. Thus, the proposed design changes do not adversely impact any SSC that could be an initiator of any USAR evaluated events. Therefore, the proposed modifications will not adversely impact the probability of occurrence or the consequences of any accident.

The completed Failure Modes and Effects Analysis (FMEA), seismic qualification of new panel attachments, and EMI/RFI testing demonstrates that the proposed modifications do not adversely affect any SSC evaluated or discussed in the USAR. There are no new system interconnections or interactions and the revised intersystem interlocks provide an equivalent level of protection to the existing controls. Hence, the proposed activity will not increase the likelihood of occurrence or the consequences of a malfunction of an SSC important to safety previously evaluated in the USAR.

The new control system is a replacement PLC system that is more software based than the existing system. Compliance to industry standards during software development and control of software per site procedures provides assurance of reliable software function. In addition, new winch parts comply with the original vendor specified capacity and safety factors. The FMEA and panel seismic attachments discussed previously assure no unacceptable actions. These factors when combined with the planned in-depth testing provide assurance against unacceptable system interactions. Hence, the proposed activity will not create the possibility for an accident of a different type than any previously evaluated in the USAR.

The IFTS is not identified as an initiator or mitigator of any USAR events. The modifications do not alter or adversely impact any USAR described design functions or control actions. The proposed changes will not result in any new or altered system interactions. The replacement control panels are seismically attached to prevent any impact to other SSCs important to safety. Comprehensive testing prior to first use assures the reliability of the new software/logic. Furthermore, as identified in USAR Table 9.1-4, fuel transfer system components are classified as non-essential and non-safety related. Thus, the proposed IFTS control system upgrade does not involve SSCs that are safety related, required to safely shut down the plant during or following a design basis accident, perform a safety related function, or are relied upon for any safety system operation. Hence, the proposed activity will not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the USAR.

As previously stated this modification does not alter the IFTS tube and blank and does not create any new system actions that could adversely affect the integrity of this barrier. Hence, the proposed activity does not result in a design basis limit for a fission product barrier as described in the USAR being exceeded or altered.

There are no evaluation methodologies described in the USAR related to the IFTS or its components. Therefore, this activity does not result in a departure from the method of evaluation described in the USAR used in establishing the design bases or used in the safety analyses.

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

No.

Evaluation No.: 04-00336

Source Document: ECP 02-0023

1.1 Activity Description

The fuel handling platform rides on rails and covers all of the various fuel handling and storage pools. It is used to transport fuel and other components to and from the Fuel Handling Building fuel pools. It has 2 hoists: a main fuel hoist used with a telescoping fuel grapple and a monorail mounted hoist.

The fuel handling platform control system is now considered obsolete and is in need of a major upgrade. The current inventory of spare parts will not be adequate to support the platform and platform operations could be affected. Equipment aging has compromised reliability to the degree that significant downtime is being encountered more frequently each outage.

The proposed activity will perform a major control system upgrade to the fuel handling platform that will replace the existing control system and panels with a new PLC based control system. The new controls will utilize a touchscreen status monitor for the operator interface versus the existing switches, lights and relay logic. Beside the control system, the major components that are being replaced are: a) main fuel hoist with a new hoist, motor, brakes and motor drive; b) motors, gearboxes, coupling, and motor drives on the bridge and trolley; c) mechanical limit switches; d) new load cell; e) operator status console with a new operator workstation; f) provides a new start/stop station; g) access ladders and; h) bridge lighting.

1.2 Summary of Evaluation

There are two accident scenarios related to fuel handling platform - Inadvertent Loading and Operation with Fuel Assembly in Improper Position and Fuel Handling Accident Outside Containment. As detailed in the impact evaluation, the fuel handling platform is not an initiator of any of these accidents. Installation of the new control system improves the reliability of the fuel handling platform. The new control 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) a successful Factory Acceptance Test (FAT), and (4) qualification to the intent of Regulatory Guide 1.180, Revision 1 in regard to EMI/RFI. As confirmed by the EMI/RFI testing and FAT, the new equipment will not create any significant EMI/RFI that could adversely impact other control systems. The addition of automatic move capability with the increased level of protective interlocks over operator manual control will reduce potential failures. Thus, the proposed design changes do not adversely impact any SSC that could be an initiator of any USAR evaluated events. Thus, the proposed modifications will not adversely impact frequency of occurrence of any accident.

The completed Failure Modes and Effects Analysis (FMEA), seismic qualification of replaced component attachments, and EMI/RFI testing demonstrates that the proposed modifications do not adversely affect any SSC evaluated or discussed in the USAR. There are no new system interconnections or interactions and the revised intersystem interlocks have been shown to provide an equivalent level of protection to the existing controls. Hence, the proposed activity will not increase the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the USAR.

The new control system is a digital upgrade system. Compliance to industry standards during software development and control of software per site procedures provides assurance of reliable software function. In addition, new platform parts comply with the original vendor specified capacity and safety factors. The FMEA and panel seismic attachments discussed previously assure no unacceptable actions. These factors when combined with the planned in-depth testing provide assurance against unacceptable system interactions. Hence, the proposed activity will not create the possibility for an accident of a different type than any previously evaluated in the USAR.

The fuel handling platform is not identified as an initiator or mitigator of any USAR events. The modifications do not alter or adversely impact any USAR described design functions or control actions. The replacement components are seismically attached to prevent any impact to other SSCs important to safety. Hence, the proposed activity will not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the USAR.

The proposed activity does not impact the refueling interlocks associated with preventing inadvertent criticality. Nor does the proposed activity involve any of the administrative controls related to the movement of non-fuel items over irradiated fuel inside containment. Hence, the proposed activity does not result in a design basis limit for a fission product barrier as described in the USAR being exceeded or altered.

The USAR has methodology associated with the evaluation of the dropping of a fuel bundle or irradiated component for the Fuel Handling Accident Outside Containment. The proposed changes do not alter any parameters such as maximum height of fuel over other fuel that could be used for this analysis in the USAR. This change complies with all applicable codes and standards; therefore, this activity does not result in a departure from the method of evaluation described in the USAR used in establishing the design bases or used in the safety analyses.

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

No.

Evaluation No.: 04-00690

Source Document: TXI-365, TM 04-0006

1.1 Activity Description

This temporary modification and contingency procedure is being implemented to provide an alternate means for reactor decay heat removal as defense in depth during the planned evolution to repair the Division 2 Emergency Service Water pump. The alternate decay heat removal method involves connecting the Fire Service Water (P54) System to the Division 2 loop of the Emergency Service Water (P45) System via temporary piping through the Unit 2 Emergency Closed Cooling Water (P42) System piping. The other heat loads on the Division 2 ESW loop will be isolated, such that cooling water is provided to the Division 2 Residual Heat Removal System (RHR) heat exchangers only. The TXI procedure authorizes opening locked closed isolation valves between P54 and the P45 system to maximize P45 water flow to the RHR heat exchanger for decay heat removal. The function of the alternate decay heat removal system is to remove sufficient decay heat from the reactor to maintain the reactor in the cold shutdown mode (Mode 4).

1.2 Summary of Evaluation

The USAR accidents 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 affected by the implementation of the proposed change.

This evaluation analyzes USAR described design functions that are potentially affected by the proposed design change. All potentially affected design functions were found to be satisfactorily performed, such that no increase in the likelihood of any malfunctions of SSCs important to safety were identified.

This change does not result in increasing previously evaluated release rates, changing a release duration, establishing a new release mechanism or path, or changing mitigation effectiveness. The proposed change does not result in increased dose consequences that might impede required actions inside or outside the control room to mitigate the consequences of reactor accidents.

Based on the scope of the proposed change, and the evaluation of possible failure effects of the proposed change, no new events of a significance to be considered an accident, nor new malfunctions of an SSC important to safety, could be identified.

This evaluation analyzes the effects of the proposed change on the PNPP fission product barriers. No effects were identified.

The proposed TM does not change or affect any of the methods used in establishing the design bases and the safety analyses documented in the USAR

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

No.

Evaluation No.: 04-00762

Source Document: USAR CR 04-035

1.1 Activity Description

The proposed activity updates USAR Sections 3.6.2.1.7 and 5.2.4.9 to allow the use of risk-informed methodology in determining the number of augmented piping weld inspections in the Break Exclusion Region (BER).

1.2 Summary of Evaluation

The proposed activity allows the use of an alternate method for determining the number of augmented piping inspections required to meet the criteria of USAR Sections 3.6.2.1.7 and 5.2.4.9. These USAR sections are based upon the criteria contained in section 3.6.2 of the Standard Review Plan, Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping, and specifically Branch Technical Position MEB 3-1, Postulated Rupture Locations in Fluid System Piping Inside And Outside Containment. The proposed activity implements a methodology approved by the NRC for this intended application and as such is not a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analysis.

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

No.

Evaluation No.: 04-00891

Source Document: ECP 03-0281, ECP 03-0281-01, ECP 03-0281-02

1.1 Activity Description

The proposed activities remove the strainers from the six Combined Intermediate Valves (CIVs) to reduce the pressure drop across the CIVs to allow more steam flow into the three Low Pressure (LP) turbines, resulting in increased power output of the turbine generator. This will provide a commercial benefit via the increased plant electrical output.

The purpose of the CIV strainers is to capture foreign material that may come loose from the upstream piping or moisture separator reheater during operation and prevent it from entering and damaging the LP turbines. Inspection of the strainers since Refueling Outage 1 has revealed no foreign material accumulation. Permanent removal of the strainers is considered by the Original Equipment Manufacturer (GE) to be acceptable to eliminate the related pressure drop and increase generator output. The pressure drop in each CIV that is caused by the restriction of a clean strainer is estimated to be about 2 psig, based on CIV strainer removal results observed at another facility. This is within the range of differential pressure variations that currently exist across all six CIVs at PNPP, as indicated by plant computer data. Removal of all six CIV strainers will increase turbine generator output by 1 to 2 MW. As per GE, removal of the strainers can be done at different times, although it is recommended that they be removed in pairs from each LP turbine. Strainer removal in this manner will help maintain a balanced pressure drop and consistent steam flow on each side of a given LP turbine hood.

1.2 Summary of Evaluation

The relevant USAR accidents (i.e., Turbine Trip and remotely associated Steam System Piping Break Outside Containment) 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 affected by the implementation of the proposed change (i.e., removal of the CIV strainers). The USAR described design functions that are potentially affected by the proposed change were evaluated. Both direct and indirect effects of the proposed change on component and system design functions were evaluated. Potentially affected design functions were found to be unaltered. Consideration of the potential for equipment failure and an increased likelihood of a malfunction associated with the design change was 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. The effect of the proposed change on the fission product barriers was evaluated and 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 USAR.

Based this Evaluation, removal of the CIV strainers does not meet any of the (c)(2) criteria of 10 CFR 50.59 that would require a license amendment.

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

No.

Evaluation No.: 04-01035

Source Document: TXI-0321

1.1 Activity Description

TXI-321 "NobleChem™ Metals Addition" was successfully performed in February 2001, during the first two days of Refueling Outage 8. Subsequently questions concerning the application of NobleChem™ [also known as Noble Metal Chemical Addition (NMCA) or noble metal]) and its effect on nuclear fuel Peak Cladding Temperature (PCT) during a LOCA arose. This 10 CFR 50.59 Evaluation addresses this aspect of NobleChem™ and supercedes PNPP Safety Evaluation 01-0007.

1.2 Summary of Evaluation

The addition of NobleChem™ to the reactor coolant system is being performed to prevent future or arrest existing Intergranular Stress Corrosion Cracking (IGSCC). This change was analyzed and will not affect SSCs by adverse interactions with treated materials. Potential issues such as fit, adhesion, heat transfer and corrosion are addressed. Accident and Anticipated Operating Transient (AOT) analyses are also not affected, with detailed analysis performed for effect on Peak Cladding Temperature (PCT) in the LOCA and post-LOCA environment. This analysis concluded NobleChem™ does not change the threshold or reaction rate of the zirconium/water reaction and as such, does not affect the PCT defined in the LOCA analysis.

No changes or effects were identified to the licensing documents. A minimal increase in reactor coolant pressure boundary failure was noted due to a slightly increased pipe wall thinning rate as a result of changed metal surface chemistry conditions after NobleChem™ application. This should be offset by the reduction in probability of a LOCA due to IGSCC in the post-NobleChem™ environment.

As the NobleChem™ treatment is has minimal effect to the material it contacts, there was no increase in the consequences of an accident. No new or changed equipment interactions were identified as a result of NobleChem™ treatment. Plant equipment will continue to be operated and function as designed. No new failure modes and effects will occur as a result of NobleChem™ treatment. As all treated equipment will continue to function as designed, there is no possibility for a malfunction of an SSC important to safety with a different result than that identified in the USAR.

No effects were identified in regard to pressure and temperature rating of SSCs. All current evaluation methods described in the USAR remain unchanged. Compliance with design standards, codes, and modeling will assure no increase in malfunctions.

Therefore this activity does not require a License Amendment per paragraph (c)(2) of 10 CFR 50.59.

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

No.

Evaluation No.: 04-01200

Source Document: ECP 04-00169

1.1 Activity Description

This proposed activity modifies the existing Testable Rupture Disk (TRD) design that is installed on the combustion exhaust system for the Division 1 and 2 Emergency Diesel Generators (EDG). The basic design and function of the TRD will not change. The existing TRD assembly consists of a hinged steel disk that seats over the 30-inch diameter exhaust relief flange that is part of the exhaust system piping. The disk is held closed by a latch assembly that consists of a pivoting counterweight.

The upgraded TRD design implemented includes the following changes. The slotted holes for the hinge pins are replaced with round holes. GRAPHALLOY bushings with hardened steel pins will be installed at the hinge pins and latch mechanism pivot. Hard facing is applied to the contact surfaces of the latching mechanism. Larger lugs will be installed at the pivot points to enhance capability to withstand a seismic event. The maximum angle of disk opening is increased to 45 degrees.

These enhancements to the original design are intended to minimize the binding/repeatability problems that are present in the existing TRD design, and to reduce combustion exhaust backpressure when the TRD is open.

1.2 Summary of Evaluation

The effects of this change have been reviewed against the causes of all of the USAR evaluated accidents. Since the causes are not affected, the frequency of occurrence of any accident previously evaluated in the USAR is not increased.

The effects of this change were reviewed against potentially affected USAR described design functions. From that review, there is sufficient evidence to conclude that the Division 1 and 2 EDGs will maintain their ability to function as credited in the PNPP USAR to mitigate design basis events, including earthquakes and tornadoes. This change maintains the redundancy, independence and separation of the onsite EDGs. Based on the evaluations and analyses performed, this change does not affect the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the USAR.

Any accidents that rely on the function of the Division 1 and 2 EDGs would be mitigated as evaluated in the USAR. 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, this change does not increase the radiological consequences of any accident or malfunction of equipment important to safety evaluated previously in the PNPP USAR.

This change does not cause a previously evaluated event to become categorized as an accident. This change does not make any events previously categorized in the USAR 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, this change does not create a new potential for common mode failure of the EDG. The malfunction effects identified in the analysis section are bound by previously evaluated USAR malfunctions (loss of a single electrical division).

Based on the analysis performed, the change maintains the USAR described design functions of the Division 1 and 2 EDGs. Consequently, the Division 1 and 2 EDGs and their combustion exhaust systems will perform as evaluated previously, and there are no indirect effects on the design basis limits for any of the fission product barriers.

The analyses that establish the TRD design/qualification are consistent with the USAR descriptions. Post modification and periodic surveillance testing will confirm the proper function of the TRD. All of the analytical/empirical methods utilized are considered to be consistent with the methods described in the USAR that establish the design basis of the Division 1 and 2 EDGs (i.e., Regulatory Guide 1.9, IEEE 308; IEEE 387).

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

No.

Evaluation No.: 04-01378

Source Document: ECP 04-0092, ECP 04-0092-01

1.1 Activity Description

The proposed activities will remove the vibration trip sensors from the Division 1 and 2 Emergency Diesel Generators (EDGs). The change will remove four vibration trip sensors from each diesel engine, two from the engine block and one from each turbocharger. Electrical leads will be lifted and removed from the pressure switches that sense the diesel engine vibration trip signal. The vibration trip alarm pressure switches will be spared in place. Setpoints for the vibration trip alarm pressure switch will be deleted/cancelled. The diesel engine vibration trip annunciator windows will be blanked out on the engine control panels, and on the main control room panels.

1.2 Summary of Evaluation

The effects of this design change on the PNPP USAR accident analysis was evaluated and a technical basis has been established to conclude the proposed change does not increase any USAR accident frequencies.

The review performed regarding the effects of this design change on USAR described design functions establishes evidence to conclude that the affected USAR described design functions will be performed as evaluated previously. Based on that conclusion, this change cannot involve an increase to the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated.

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, this change does not increase the radiological consequences of any accident or malfunction of equipment important to safety evaluated previously in the PNPP USAR.

This change does not compromise any of the USAR described design functions associated with the Division 1 and 2 EDGs. The change does not create the potential for the occurrence of any event of such significance that it could be categorized as an accident.

The removal of the EDG vibration trip function does not affect the internal missile hazard protection basis for the EDGs. This change does not create a new potential for common mode failure of the EDGs. Since the change does not change an event previously considered to be incredible to credible, no possibility of a malfunction with a different result is created.

Since this change is limited in scope to the Division 1 and 2 EDG protective trip controls it can not directly affect a fission product barrier. Furthermore, the ECCS and containment heat removal systems will perform as evaluated previously, and there are no indirect effects on the design basis limits for any of the fission product barriers. Since the scope of this change is being implemented consistent with design basis codes and standards, any calculations performed to support this change are not considered to involve revising or replacing any USAR described evaluation methodology or elements of methodologies that would be used in establishing the design bases or used in the safety analyses.

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

No.

Evaluation No.: 05-01304

Source Document: ECP 05-0028, ECP 05-0028-01

1.1 Activity Description

The proposed activities install nominal 32-second time delay relays for valves 1G33F0028, 1G33F0034, 1G33F0053 and 1G33F0054. The time delay allows the 1G33F0001 and 1G33F0004 valves to be the first Reactor Water Cleanup (RWCU) containment isolation valves to close during a RWCU isolation signal. This reduces the differential pressure that other RWCU isolation valves 1G33F0028, 1G33F0034, 1G33F0053 and 1G33F0054 must close against during their closing operation.

1.2 Summary of Evaluation

The USAR accidents 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 affected by the implementation of the proposed change. This evaluation analyzes USAR 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 was reviewed. The likelihood of any malfunctions of equipment important to safety is not more than minimally increased by this change. This change does not result in a more than minimal increase to previously evaluated release rates or release durations. This change does not establish a new release mechanism or path and there is no change to mitigation effectiveness. Accidents are mitigated as described in the USAR. Therefore, dose consequences are not affected. 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. The proposed activity will not result in a more than minimal increase in the consequences of a malfunction of an SSC important to safety. 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 adverse 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 USAR.

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

No.

Evaluation No.: 05-01539

Source Document: ECP 05-0032

1.1 Activity Description

The proposed activity installs insulation in the Emergency Diesel Generator (EDG) exhaust missile barrier and removes the Division 1, 2, and 3 Testable Rupture Discs (blowoff hatches). The existing plant design includes Testable Rupture Discs (TRD) for a safety-related exhaust flow path in the event of the failure of the EDG exhaust silencer, which are non-safety-related. The TRDs are essentially calibrated flappers that are designed to open based on engine backpressure (i.e., if an exhaust silencer is blocked, the associated TRD will open). The TRDs provide an exhaust flow path into the EDG missile barrier. The exhaust gas then flows through the missile barrier and through several doorways and Unit 2 construction openings into the atmosphere.

The EDG exhaust missile barrier was never analyzed for the elevated temperatures that could be experienced during EDG operation with exhaust discharging through the TRD. As discussed in USAR Section 3.8.3.3.7, the maximum normal operation temperature of the concrete is limited to 150°F and the maximum accident temperature of the concrete is limited to 350°F. To resolve the elevated concrete temperature as a result of exhaust discharging through the TRD, the inside of the EDG exhaust missile barrier will be insulated with a high efficiency insulation board. As a result of a history of reliability issues with the TRDs, the TRDs will be removed. This will result in the EDG missile barrier becoming a normal exhaust path in parallel with the existing silencers.

1.2 Summary of Evaluation

The effects of this change have been reviewed against the initiators of all of the USAR evaluated accidents. Since the initiators are not affected, the frequency of occurrence of any accident previously evaluated in the USAR is not increased.

The effects of this change were reviewed against potentially affected USAR described design functions. From that review, there is sufficient evidence to conclude that the Division 1, 2, and 3 EDGs will maintain their ability to function as credited in the PNPP USAR to mitigate design basis events. This change maintains the redundancy, independence and separation of the onsite EDGs. Based on the evaluations and analyses performed, this change does not affect the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the USAR.

Any accidents that rely on the function of the Division 1, 2, and 3 EDGs would be mitigated as evaluated in the USAR. 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, this change does not increase the radiological consequences of any accident or malfunction of equipment important to safety evaluated previously in the PNPP USAR.

This change does not cause a previously evaluated event to become categorized as an accident. This change does not make any events previously categorized in the USAR 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, this change does not create a new potential for common mode failure of the EDGs. The seismic, structural, thermal, and tornado loads etc. have been analyzed to prevent the insulation composite or support structure from becoming an internal missile or potential debris that could block the exhaust plenum. The stainless steel sheet and insulation are supported from multiple locations with elements designed for the conditions of service and are within accepted design limits. The insulation composite and support structure uses a robust design meant to withstand all plant accident conditions and is not considered to fail. The malfunction effects identified in the analysis section are bounded by previously evaluated USAR malfunctions (loss of a single electrical division).

Based on the analysis performed, the change maintains the USAR described design functions of the Division 1, 2, and 3 EDGs. Consequently, the EDGs and their combustion exhaust system will perform as evaluated previously, and there are no indirect effects on the design basis limits for any of the fission product barriers.

All of the analytical/empirical methods utilized are considered to be consistent with the methods described in the USAR that establish the design basis of the EDG and the Diesel Generator Building.

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

No.

Evaluation No.: 05-03796

Source Document: NOP-ER-3201

1.1 Activity Description

PNPP is committed to Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants", Rev 2, March 1978, for the design and testing of the Emergency Safety Feature (ESF) ventilation system air filtration and adsorption units. Regulatory Guide 1.52, Regulatory Positions C.5.c and C.5.d, indicate that in-place testing of the HEPA filters and the activated carbon adsorber sections should be performed following painting, fire or chemical release in any ventilation zone communicating with the system. Additionally, Position C.6.b references Table 2 of the Regulatory Guide, which indicates that a laboratory test for a representative sample of the activated carbon be performed following painting, fire or chemical release in any ventilation zone communicating with the system.

For the non-ESF ventilation systems, PNPP is committed to Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants", Rev 0, March 1978. Regulatory Guide 1.140, Regulatory Positions C.5.c and C.5.d, also indicates that in-place testing of the HEPA filters and the activated carbon adsorber sections should be performed following painting, fire or chemical release in any ventilation zone communicating with the system in such a manner that the HEPA filters or charcoal adsorbers could become adversely affected by the fumes, chemicals or foreign materials. Additionally, Position C.6.b references Table 2 of the Regulatory Guide, which indicates that a laboratory test for a representative sample of the activated carbon be performed following painting, fire or chemical release in any ventilation zone communicating with the system in such a manner that the HEPA filters or charcoal adsorbers could become adversely affected by the fumes, chemicals or foreign materials.

Even though the frequency for testing the ESF and normal ventilation system filtration and adsorption units are delineated in the USAR and Technical Specifications (ESF units only) by reference to the applicable Regulatory Guides, no clear definition of what constitutes "conditions which could have an adverse affect on the filters" has been previously documented and the terms "painting", "fires", "chemical releases", and "communicating" have not been defined. Nuclear Operating Procedure NOP-ER-3201, "Control of Carbon Filter Contaminants", has been developed to document the basis for the interpretation of Regulatory Guide 1.52, Regulatory Positions C.5.c and C.5.d and C.6.b and Regulatory Guide 1.140, Regulatory Positions C.5.c and C.5.d and C.6.b with regard to the threshold for exposure to these contaminants that would trigger the required testing and defines the terms "painting", "fires", "chemical releases", and "communicating". NOP-ER-3201 references "A Study on the Effects of Coating Operation on Radioiodine Removing Adsorbents" (W.P. Freeman and J.C. Enneking, Nuclear Consulting Services, Inc., 21st DOE/NRC Nuclear Air Cleaning Conference) that shows a negligible carbon degradation occurs up to approximately 10% by weight of Volatile Organic Compound content. NOP-ER-3201 states that for conservatism, a carbon sample shall be removed for analysis at 5% and each 2.5% increment thereafter by weight of the carbon filters. These values will be utilized at PNPP to determine when to test the filtration units. Additionally, NOP-ER-3201 defines the terms "painting", "fire", "and "chemical release", and other terms such as: Volatile Organic Compound (VOC) planned VOC release activity, unplanned VOC release activity, and contaminant.

1.2 Summary of Evaluation

The effects of this activity have been reviewed against the initiators of all of the USAR evaluated accidents. Since the initiators are not affected, the frequency of occurrence of any accident previously evaluated in the USAR is not increased.

The effects of this activity were reviewed against potentially affected USAR described design functions. From that review, there is sufficient evidence to conclude that the ESF filtration and adsorption units will maintain their ability to function as credited in the PNPP USAR to mitigate design basis events. Likewise, the impacts on the normal ventilation system filtration and adsorption units were evaluated and it was determined that their design functions have not been adversely impacted. This activity does not impact the redundancy, independence or separation of the ESF filtration and adsorption units. Based on the evaluations and analyses performed, this activity does not affect the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the USAR (the normal ventilation systems are not classified as important to safety).

Excessive "pre-loading" of contaminants such as VOCs on the carbon adsorption sites of an ESF filtration system could reduce the efficiency of the filter unit which in turn would result in an increase in the consequences on an accident. Industry testing has shown that the activated carbon can be saturated with VOCs up to 10% by weight without reducing performance to an unacceptable level. For PNPP, the effect is even further reduced since the credited carbon adsorption penetration (i.e., inefficiency) has been eliminated for the Annulus Exhaust Gas Treatment System (AEGTS) and the Fuel Handling Area Exhaust System (FHAES) (i.e., assume that all activity passes though the filter). For the Control Room Emergency Recirculation System (CRERS), the credited penetration has been reduced by a factor of 10 (inefficiencies have been reduced from 5% to 50%). As such, there is adequate margin to ensure that the presence of 10% VOCs by weight will not affect the credited performance. Even though the accident analyses have been revised with either no credit (AEGTS or FHAES) or 50% efficiency (CRERS), the systems are maintained and tested, consistent with the industry requirements for high performance safety-related atmospheric cleanup systems as required by the Technical Specifications. Any accidents that rely on the function of the ESF filtration and adsorption units would be mitigated as evaluated in the USAR. This activity does not result in any increases in the calculated 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, this activity does not increase the radiological consequences of any accident or malfunction of equipment important to safety evaluated previously in the PNPP USAR.

This activity does not cause a previously evaluated event to become categorized as an accident. This activity does not make any events previously categorized in the USAR as incredible to be categorized as credible. Based on the analysis performed, the activity 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, this activity does not create a new potential for common mode failure of the ESF filtration and adsorption units. The malfunction effects identified in the analysis section are bounded by previously evaluated USAR malfunctions.

Based on the analysis performed, the activity maintains the USAR described design functions of the ESF filtration and adsorption units. Consequently, the ESF filtration and adsorption units will perform as evaluated previously, and there are no indirect effects on the design basis limits for any of the fission product barriers. Additionally, the activity maintains the USAR described design functions of the normal ventilation systems' filtration and adsorption units. Consequently, the normal ventilation systems' filtration and adsorption units will perform as evaluated previously in the USAR.

All of the analytical/empirical methods utilized are considered to be consistent with the methods described in the USAR that establish the design basis of the ESF filtration and adsorption units. Regulatory Guide 1.52, Revision 2 provides a "methodology" for ensuring that the activated carbon filter units perform as credited following design basis accidents. The interpretation of the requirements of Regulatory Guide 1.52, Revision 2 relative to testing could be construed as a change to a methodology, however, the NRC has stated in a Letter to Entergy Operations dated September 11, 1997 that licensees can interpret the subject requirements on a sound and conservative technical basis. The combination of the industry testing and the minimal performance credited at PNPP provides this basis and as such, NOP-ER-3201 does not result in a departure from a method of evaluation described in the USAR. No revision to the USAR or any other licensing basis document is required as a result of this activity.

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

No.