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SAFETY EVALUATION SUMMARY REPORT 1992

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SAFETY EVALUATION SUMMARY REPORT

1992

Docket No. 50-341
License No. NPF-43

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SAFETY EVALUATION SUMMARY REPORT

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AS-BUILT NOTICES

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0117 UFSAR Revision No. 6

Reference Document: ABN 12969-1 Section(s) 9.2

Table(s) 9.2-3

Figure Change [ ] Yes [X] No

Title of Change: Reactor Building Closed Cooling Water (RBCCW) and Emergency Equipment Cooling Water (EECW) System Pressure and Temperature Changes

SUMMARY:

This evaluation supports the following UFSAR changes:

1. The EECW makeup tank operating pressure has been changed from 8 psig to 32 psig. This change reflects the current operating practice and ensures that the EECW system can be adequately filled and vented.

2. The EECW makeup tank design temperature has been increased from 120 ^{O}F. to 140 ^{O}F. This change reflects the tank metal temperature that would exist based on postulated accident environmental profiles.

3. The UFSAR text has been revised to indicate that the EECW system operates at a higher pressure than the emergency equipment service water (EESW) system. The UFSAR originally stated that the EECW pressure was less than the EESW system pressure and that this precluded any radioactive water leakage to the environment.

The increases in EECW operating pressure and EECW makeup tank design temperature are within the design parameters of the system and the resultant component stresses are within ASME Code allowable stresses. There is no impact on the cooling capacity of the EECW and EESW systems or the operation of the plant process radiation monitoring and sampling systems. Operating the EECW system at a higher pressure than the EESW system increases the possibility of the EECW system leaking into the residual heat removal (RHR) reservoir. However, radioactive contamination of the RHR reservoir would require multiple failures (a contaminated reactor building closed cooling water (RBCCW) inleakage failure and an EECW heat exchanger tube leak) in the RBCCW system. The operators would be alerted to the existence of gross leakage by various temperature alarms associated with the RBCCW system and cooling loads; relief valve actuations; and the RBCCW, EECW, and EESW radiation monitors. These systems also have sampling capabilities. Therefore, the potential for contaminating the RHR reservoir is insignificant.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0126

UFSAR Revision No. 6

Reference Document: ABN 12808-1

Section(s) 9.4; 9.5

Table(s) 3.11-4

Figure Change [ ] Yes [X] No

Title of Change: Residual Heat Removal (RHR) Complex Divisions I and II Fuel Oil Storage Room Ventilation System Maximum Room Temperature Change

SUMMARY:

This evaluation changed the RHR complex divisions I and II fuel oil storage room ventilation system maximum room temperature stated in the UFSAR from 104 ^{o}F to 125 ^{o}F. Deviation Event Report DER 89-0861 changed the design calculation assumed outdoor air temperature from 91 ^{o}F to 95 ^{o}F. As a result, the calculated maximum room temperatures for the CO2 storage tank rooms and the fuel oil storage tank rooms exceed the original maximum room temperature previously stated in the UFSAR. The design calculation indicates that the higher temperatures will be seen only when any emergency diesel generator is operating and the outside temperature is 95 ^{o}F.

This change does not affect the design or function of safety systems, structures, or components. The affected equipment environmental qualifications envelope the increased calculated heat load and, therefore, ensure the operability of safety related equipment and components under the worst environmental conditions.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0079 UFSAR Revision No. 6

Reference Document: ABN 12064-1 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Emergency Diesel Generator (EDG) Lube Oil Storage Tank Quality and Seismic Classification Upgrade

SUMMARY:

This change upgraded the quality and seismic classifications for certain EDG system components to Seismic I, QA I on system diagrams, UFSAR figures, isometric drawings, instrument installation drawings, and the component database (CECO). These components include the EDG lube oil storage tank, its instrument piping and valves, and the feed line from the tank to the EDGs. Previously, the design drawings and the UFSAR figures did not depict the Seismic I, QA I levels for the EDG lube oil system as described in UFSAR Subsection 9.5.7. In addition, the UFSAR did not specify a QA I classification for the lube oil tank as required by Fermi 2 procedure FIP-CM1-03, "Identifying QA Level I and QA Level IM Structures, Systems, and Equipment". This procedure requires a QA I classification for components that have a passive essential function for pressure boundary integrity.

These classifications are consistent with the classification of other EDG safety related components. The classifications support the UFSAR assumption that the EDGs will be available for all postulated accidents and will perform as expected.

MINOR ABN'S

The following As-Built Notices (ABNs) resulted in UFSAR drawing or text changes. These changes were reviewed for potential safety consequences. Because the changes were minor and were made to reflect as-built plant conditions, a summary for each was not prepared. The ABNs and their associated safety evaluations have been listed for reference.

Safety Evaluation No.:	91-0087	Figure Change
Implementation Document:	ABN 12557-1	
Safety Evaluation No.:	91-0124	Figure Change
Implementation Document:	ABN 12609-1	
Safety Evaluation No.:	92-0054	Figure Change
Implementation Document:	ABN 13389-1	
Safety Evaluation No.:	92-0067	Figure Change
Implementation Document:	ABN 13369-1	

END OF ABN SECTION

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ENGINEERING DESIGN PACKAGES

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 87-0284 REV1 UFSAR Revision No. 6

Reference Document: EDP 7385 Section(s) N/A

Table(s) 5.2-16; 7.3-1

Figure Change [ ] Yes [X] No

Title of Change: High Pressure Coolant Injection (HPCI) Pump Suction Pressure Switch Replacement

SUMMARY:

This modification replaced the original Barksdale Model D2H-M80SS HPCI pump suction pressure switch E41N031 with a Static-O-Ring model 4N6-E45-NX-C1A-TTX6 pressure switch. The original switch pressure rating was not adequate to sustain HPCI pump suction line pressure transients. The new switch has been selected to withstand the pressure transients in the HPCI pump suction line.

This modification does not affect the function of the HPCI system or any other safety related system. The function of the replacement pressure switch is identical to the original pressure switch. A failure of either the original pressure switch or the replacement pressure switch results in the same effect: a loss of HPCI pump suction high pressure annunciator function in the control room or a loss of the HPCI pump suction line pressure boundary. The pressure switch has been installed and is maintained as a QA1/Seismic 1 component to assure its pressure boundary integrity safety function. The pressure setpoint of the HPCI pump suction pressure switch remains within the alarm setpoint specified in Fermi 2 Technical Specification Table 3.4.3.2-2 , "Reactor Coolant System Interface Valves Leakage Pressure Monitors".

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 89-0075 UFSAR Revision No. 6

Reference Document: EDP 9752 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Reactor Building Closed Cooling Water (RBCCW) Recorder Replacement

SUMMARY:

This modification replaced the RBCCW recorder P42R800 with a new recorder. Servicing the original Bailey Model SR recorder was no longer feasible since it was no longer manufactured and spare parts were either unavailable or had a 24 month lead time. The new Tracor Westronics Series 1100 (Model 1110) recorder is a similar component replacement. The new recorder has the same scale range as the previous recorder and accepts the same input signals.

This modification does not change the function or operation of any system and the recorder does not perform any safety function.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0118 REV1 UFSAR Revision No. 6

Reference Document: EDP 11264 Section(s) 7.5; 7.6

Table(s) 7.5-2; 7.6-2

Figure Change [X] Yes [ ] No

Title of Change: Installation of Narrow Range Torus Pressure Instrumentation Loops

SUMMARY:

This modification added two redundant divisional narrow range torus pressure instrumentation loops; relocated the existing wide range torus pressure transmitters T50N414A and B and their associated instrument lines; replaced the original outdated torus and drywell pressure recorders, T50R802A and B; and relocated recorder T50R802A. The narrow range transmitters were installed to address the concerns of Human Engineering Discrepancy HED 967. These transmitters allow the operators to accurately read the torus pressures to the precision required to initiate or terminate torus sprays when called for in plant emergency procedures. The wide range torus pressure transmitters and their sensing lines were relocated to ensure that water does not collect in the sensing lines in the event that the torus becomes flooded. This also enhances ALARA during maintenance and surveillance testing because the transmitters have been relocated in a lower radiation zone. The new torus and drywell pressure recorders monitor narrow range torus pressure. Recorder T50R802A, which is used during drywell/torus spray operation, has been relocated to place it closer to the Division 1 drywell/torus spray controls.

The narrow range torus pressure instrumentation will aid in the mitigation of an accident involving a steam release to primary containment. The types of equipment used in this design are presently used in similar designs in the plant. Redundant instrumentation channels are provided such that a single failure of any one power system will not affect the availability of instrument readings. Instrument fuses have been coordinated so that a failure in the instrumentation loop will not cause a failure of the DC system. The instrument tubing runs are in a different part of the building than the existing torus sensing lines. However, the design is equivalent to existing lines tapping off the torus and any radioactive release resulting from an instrument line break

Safety Evaluation 90-0118 REV1 (continued):

can be isolated from the control room by utilizing the sensing line containment isolation valves.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0080 UFSAR Revision No. 6

Reference Document: EDP 12511 Section(s) 10.4; 11A.III.A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Circulating Water (CW) System Decant Pump Power Source

SUMMARY:

This modification provided 480 VAC Equipment Bus 72J, Position 1C as the power source for the third CW system decant pump, W2500C003. This switchgear originally provided power for the circulating water chlorination system injector water booster pump, N2300C001. However, this pump is not used because the circulating water chlorination system has been retired. Additionally, the third decant pump utilizes the injector water booster pump control logic and the control room decant flow indication range has been increased from 0-20,000 gpm to 0-30,000 gpm. The new flow range provides acceptable flow indication when operating three decant pumps simultaneously.

Operation of the decant pumps does not support or interact with any equipment important to safety. Analysis of the switchgear, transformer R1400S032A loads, associated power cabling, overcurrent device settings, and control logic has concluded that the subject power source is acceptable for providing power to the third decant pump. Simultaneous operation of three decant pumps does not impact the Fermi 2 Environmental Protection Plan because the 30,000 gpm maximum decant flow rate is below the 31,319 gpm Fermi 2 National Pollutant Discharge Elimination System (NPDES) permit limit.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0088 REV1 UFSAR Revision No. 6

Reference Document: EDP 12541 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Installation of Reactor Building Closed Cooling Water (RBCCW) Pipe Taps

SUMMARY:

This modification adds 6" pipe taps to the suction and discharge lines of RBCCW pump P4200C001 to provide a means of connecting auxiliary cooling equipment to the RBCCW system. Each tap includes a 6" gate valve, capped 3/4" vent and drain valves, and blank flanged terminals. The tap terminals are located in the turbine building to allow the auxiliary cooling equipment to be connected in a non-seismic area. This modification also adds an opening in the turbine building / auxiliary building security screen for running a temporary hose between a turbine building chiller and the permanent piping in the auxiliary building. The opening can be locked closed to block access when the temporary hose is removed. The temporary auxiliary cooling equipment hook-up provides backup cooling when the in-place RBCCW heat exchangers are taken out of service for preventive maintenance. Prior to this modification, during high heat load conditions, the RBCCW heat exchangers could only be taken out of service when the emergency equipment cooling water (EECW) system surveillance runs were scheduled. This was required to provide adequate cooling for the equipment served by the RBCCW system. This modification allows greater flexibility in scheduling RBCCW heat exchanger outages.

This modification does not change the function or operation of the RBCCW system. The pipe taps are not considered to be a part of the EECW system and, therefore, are not required to perform any safety-related function. This modification meets seismic II/I criteria and is installed according to ANSI B31.1.0 requirements. A tap line break within the auxiliary building is bounded by the existing feedwater line break analysis. The area of the turbine building / auxiliary building opening meets the requirements of the Physical Security Plan.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0095   UFSAR Revision No. 6

Reference Document: EDP 11380   Section(s) 9.3

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Turbine Building Oil Room Floor Drain Sump Pump Replacement

SUMMARY:

This change replaced the inoperable obsolete turbine building oil room sump pump G1101C004Q with a comparable self-priming centrifugal pump. The original pump had a capacity of 71 gpm at 50' of head and the new pump has a capacity of 64 gpm at 45' of head. The new installation required a change in the mounting baseplate and minor re-piping of the suction and discharge lines. There was no change in the fuse, starter, or cable sizes. The thermal overload heater was changed to accommodate the lower full load current of the new pump motor.

This change does not affect safety-related equipment as the sump pump does not interface with or support safety-related equipment. This change has no effect on the seismically induced radwaste tank rupture scenario evaluated in the UFSAR. The design flowrate of the new sump pump is well within the downstream oil/water separator 200 gpm maximum design flowrate. The floor drain system will function in the same manner as before. The difference in nominal flowrates between the original and new pumps will not affect system performance.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0097 REV2 UFSAR Revision No. 6

Reference Document: EDP 12108 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Instrument Air System (IAS) Air Dryer and Receiver Tank Replacement

SUMMARY:

This modification installed two new air dryers; a receiver tank; and the required piping, valves, and instrumentation on the first floor of the turbine building. A flow element was installed in the piping between the receiver tank and the IAS distribution headers and a flow indicator was mounted near the receiver tank. The modification also removed the original IAS air dryers and receiver tank and terminated the original system air supply and component cooling lines. Each dryer is a full capacity heatless regenerative dessicant unit with its own prefilter and afterfilter. The new air dryer system was installed because installation of the eighth condensate demineralizer required the removal of the original air dryer system. Instrument air could not be interrupted during the removal of the original air dryer system. Therefore, the new system was installed and placed into service prior to removal of the original air dryer system. The flow indicator was installed to monitor instrument air usage. It provides the capability of detecting abnormally high flowrates that are indicative of system air leaks and equipment failures.

No new modes of operation are introduced because the design, function, and installation of the new air dryer system is the same as the original system described in the UFSAR. The new system provides the same quality of instrument air as the original system. Therefore, the new air dryer system does not impact the function of any component or system required for the safe shutdown of the reactor. Loss of IAS does not prevent components or systems from performing their safety related functions. The location of the new air dryer system is in the same type of fire, radiation, environmental, and seismic zones as the original equipment.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0101 REV1 UFSAR Revision No. 6

Reference Document: EDP 11251, EDP 11283, EDP 12419, EDP 12663, PDC 12974

Section(s) 10.4

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Miscellaneous Eighth Condensate Filter Demineralizer (CFD) Installation Modifications and Correction of CFD Human Engineering Deficiencies (HED)

SUMMARY:

The following changes were made to the CFD system in preparation for the installation of the eighth CFD and to correct CFD HEDs:

1. Seven original CFD differential pressure indicators have been replaced with dual scale, digital readout differential pressure indicators on local control panel H21P250.

2. New CFD flow indicators have been added on local control panel H21P250. This change allows the local operator to operate the CFDs without being in constant communication with the control room operator.

3. All CFD controls on the main control room combination operating panel (COP) H11P816 insert A500 have been removed. CFDs are not placed into service from the control room. Therefore, these controls are not required.

4. Wiring, relays, and contacts have been installed in local control panel H21P250. This work was preparatory to installing the eighth CFD.

5. The original control room CFD conductivity recorder on COP H11P816 has been replaced with a microprocessor based CFD common recorder. The new recorder monitors CFD effluent conductivity, differential pressure, and flow.

6. The CFD effluent conductivity transmitter outputs have been redirected to the new control room CFD common recorder.

7. The individual CFD differential pressure and flow indicators on COP H11P816 have been deleted. The outputs of the differential pressure

Safety Evaluation No. 91-0101 REV1 (continued):

transmitters and square root extractors have been redirected to the new control room CFD common recorder.

8. The CFD vessel flow inputs to the process computer have been deleted. The CDF vessel flow is being recorded on the new CFD common recorder.

9. A CFD system differential pressure indicator has been installed in the control room. This change allows the operators to monitor CFD fouling and gives them advanced warning when the CFD bypass valve is about to open.

10. The control room CFD bypass valve position indication has been relocated from COP H11P816 to COP H11P805. This change places the position indicator on the condensate/feedwater panel near the existing mimic for the CFD and bypass valve.

11. The hotwell coil steam supply system supply pressure indicator, supply controller, and valve N11F606 backlighted indicating pushbuttons have been removed from the control room. This change made room for the new CFD system differential pressure and the CFD bypass valve position indicators.

12. Miscellaneous CFD system diagram drafting errors have been corrected.

This modification does not change the function of the CFD system as defined in the UFSAR. It does not degrade the performance of or increase the challenges to any safety system. This modification does not result in increased radiation doses or radioactive material releases. There is no impact on the loss of feedwater flow transient described in the UFSAR and there is no adverse impact on any radiation barriers. The replacement of the control room conductivity recorder with a microprocessor based recorder was evaluated against the concerns stated in NRC memorandum "Analog to Digital Instrumentation Replacements Under the 50.59 Rule", dated July 1,1991. The new recorder does not create the potential for a new type of malfunction of equipment important to safety because the recorder is not safety related, it is not in a divisional system, and the loss of indication from the recorder is not critcal to system operation. The reactor coolant system chemistry Technical Specification limits are not affected by these changes.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0102 UFSAR Revision No. 6

Reference Document: EDP 11263 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Modification of the Intermediate Range/Source Range Monitor (IRM/SRM) Drive Switch

SUMMARY:

This modification changed the IRM/SRM "drive-in" switch from a momentary drive switch to a continuous drive switch. This change will relieve the operator from having to continuously press the drive switch to fully insert the SRM or IRM detectors. The "power on", "SRM/IRM select", and the "drive out" switches are not changed by this modification.

The function, mounting location, and configuration of the new switch have not changed. This change does not affect the ability of the detectors or detector drive units to perform their functions and has no radiological effects on any accident previously evaluated in UFSAR accident analyses.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0103 REV1 UFSAR Revision No. 6

Reference Document: EDP 3105 Section(s) N/A

Table(s) 6.2-2

Figure Change [X] Yes [ ] No

Title of Change: Installation of New Local Leak Rate Test (LLRT) Connections

SUMMARY:

This modification added new LLRT test connections to main steam line drain line inboard containment isolation valve B2103F016 and reactor core injection cooling steam line inboard containment isolation valve E5150F007. The test connections consist of double isolation valves and flexible metal hose welded to a new test connection nipple installed on each valve. The new test connection assemblies eliminate cutting and recapping the existing test connection nipples during each LLRT test.

This modification does not alter the LLRT criteria or the method by which LLRTs are performed. No new radiological release or bypass pathways affecting the site environmental or accident analyses are created. The failure of the test connections will result in a small break loss of coolant accident that is within the bounding analyses evaluated in the UFSAR. The design is seismically qualified and meets the ASME code requirements for each system's specifications.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0104 UFSAR Revision No. 6

Reference Document: EDP 10745 Section(s) 2.2; 6.4; 9.2; 9.4; 9.5 10.4; 12.2; A1.78; A1.95

Table(s) 9.3-1; 9.4-15

Figure Change [X] Yes [ ] No

Title of Change: Installation of Circulating Water System (CW) Biocide Injection System

SUMMARY:

This modification installed a new CW biocide injection system. This system uses bromo-chloro dimethylhydantoin (BCDMH) as the biocide. The three skid-mounted brominators have a capacity of 100 gpm each and are installed in a parallel configuration. The new system uses general service water to dissolve the chemical and inject the resultant solution into the CW system. The system is located in the circulating water pump house (CWPH) chlorine room. This modification also removed a majority of the original chlorination equipment and modified the CWPH HVAC system logic associated with the operation of the chlorine room by deleting the supply and exhaust fans chlorine trip. The ventilation system now has the ability to operate both fans individually and the existing temperature control logic is unaffected. The installation of this system is a result of an extensive review of the original chlorination system which determined that the CW chlorination system was not the most effective biocide system to use.

No previously evaluated accidents result from the application of a biocide in the CW system. Use of BCDMH has no significant impact on the capacity margin for the condensate polishing demineralizer system to deal with a condenser tube leak. The CW system is not responsible for any initiating events described in UFSAR chapters 6 or 15 and is not required to respond to any of these events. The modifications made to the CWPH HVAC system are limited to the chlorine room subsystem and result from the removal of chlorine. There is no impact on any accident scenario or main control room habitability because the control center HVAC chlorine detectors remain in place. There is no impact on equipment important to safety.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0115

UFSAR Revision No. 6

Reference Document: EDP 11275

Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Addition of High Pressure Coolant Injection (HPCI) and Reactor Core Injection Cooling (RCIC) Actuation Alarms

SUMMARY:

This modification added RCIC and HPCI system actuation alarms to control room panels H11P601 and H11P602, respectively. The need for the alarms was identified in Human Engineering Deficiency Report HED 1179 because the Fermi 2 UFSAR contains an accident analysis for inadvertent HPCI initiation and it is possible that operations personnel may not readily identify a HPCI/RCIC actuation without an alarm.

The installation of the HPCI and RCIC actuation alarms does not affect the operation of the HPCI and RCIC systems or other plant systems used to mitigate accidents. The equipment installed does not interface with any safety related equipment. This modification provides additional plant status information and may help prevent a level 8 trip of the HPCI and RCIC systems by alerting the operators that the systems have actuated.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0117 UFSAR Revision No. 6

Reference Document: EDP 12100 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Installation of General Service Water (GSW) System Biocide Injection System

SUMMARY:

This modification installed a new biocide system to treat the GSW and fire protection systems pump discharge headers and discharge piping. This system uses bromo-chloro dimethylhydantoin (BCDMH) as the biocide. The two skid mounted brominators have a capacity of 50 gpm each and are installed in parallel at the GSW pump house. The biocide injection point has been moved from the GSW pump pit to the 48" mains supplying plant loads. The existing GSW chlorination equipment located in the Fermi 1 plant services building has been abandoned in place. Control panel P41-P406, chlorine supply valve P4100F607, chlorine detector P41N402, the chlorine injector, and associated piping have been removed from the GSW pump house. The new injection system uses GSW to dissolve the chemical and inject the resultant solution into the GSW and fire protection systems. This modification also corrects a deficiency between the as-built GSW system and plant drawings by adding an existing GSW to radwaste evaporators drain valve P4100F520 to GSW system piping and instrumentation (P&ID) drawing M-2010 and functional operating sketch (FOS) M-5726. This deficiency was detected during an operations walkdown and previously documented in Potential Design Concern PDC 12968.

No previously evaluated accidents result from the application of a biocide to the GSW system. A failure of the biocide injection system will not affect safety related equipment. The GSW system is not required for the safe shutdown of the plant and this modification does not affect the GSW system service requirements for the emergency equipment cooling water (EECW) and emergency equipment service water (EESW) systems. BCDMH is compatible with the equipment served by the GSW system. Adding P4100F520 to the GSW P&ID and FOS drawings is a drawing correction and it does not change the physical plant configuration. The existence of this valve does not affect any accident scenarios and its failure is not an initiating event for any analyzed accidents.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0123 | UFSAR Revision No. 6

Reference Document: EDP 12420 | Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Installation of Concrete Walls and Monorail for the Eighth Condensate Filter Demineralizer (CFD)

SUMMARY:

This modification installed concrete shield walls and a monorail in preparation for the installation of the eighth CFD. The concrete shield walls and the monorail are located on the turbine building first and second floors, respectively. The walls have penetrations to allow for the routing of pipes, conduits, instrument lines, and HVAC ducts.

The shield walls and monorail for the eighth CFD do not have any safety related function. They are designed for structural adequacy. The walls provide adequate radiation shielding similar to the walls for the original seven CFDs. The walls and foundation are not subject to any UFSAR accident evaluations.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0125 UFSAR Revision No. 6

Reference Document: EDP 12487 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: General Service Water (GSW) Header Pressure Control Valves Isolation Valve Replacement

SUMMARY:

This modification replaced the original GSW header pressure control valve isolation butterfly valve P4100F134 with a gate valve. This replacement was necessary because the original valve did not provide the isolation required for GSW header pressure control valves P4100F401A and B maintenance. The new gate valve provides assurance that the pressure control valves can be isolated and GSW system outages are avoided. This change also modifies hanger X21-2005-G81 to support the additional weight of the new gate valve.

This modification has no impact on the safe shutdown capability of the plant. There is no impact on existing accident scenarios because, although the valves are different types, their function and operation is the same.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0127 REV4 UFSAR Revision No. 6

Reference Document: EDP 11473 Section(s) 3.6; 5.5; 7.4

Table(s) 8.3-15

Figure Change [X] Yes [ ] No

Title of Change: Restoration of Reactor Core Injection Cooling (RCIC) Steam Inlet Warmup Bypass Valve Function

SUMMARY:

This modification restored the function of the RCIC turbine steam inlet warmup bypass valve E51F095 by:

1. Replacing the original solenoid operated valve with a motor operated valve (MOV) and redesignating it as E5150F095.

2. Replacing the blank in the restricting orifice E51-D011 with the originally designed 5/16" diameter flow orifice.

3. Adding lanyard potentiometers to the RCIC turbine governor valve E5150F044 and steam inlet isolation valve E5150F045 for increased diagnostic monitoring of RCIC system through the General Electric Transient Analysis Recording System (GETARS).

The Fermi 2 power uprate feasibility study indicates that the 25 psig increase in main steam pressure experienced while operating under power uprate conditions reduces the RCIC turbine overspeed response design margin. The Fermi 2 power uprate technical specification submittal addressed this concern by making a commitment to reinstate the warmup bypass line. The original solenoid operated pilot ported valve was replaced with a motor operated globe valve to improve valve reliability. The original valve was susceptible to particulate blockage which resulted in its inability to fully close. The installation of lanyard potentiometers on E5150F044 and E5150F045 carries out a recommendation in General Electric Service Installation Letter SIL 336 for improved monitoring of RCIC quick-start capability and proper valve sequencing.

This modification does not change the function of the bypass valve or RCIC system. It provides a greater RCIC turbine overspeed design margin. The new MOV meets the same ASME Class II code integrity requirements as the original valve. The new MOV is powered from the existing 260/130 VDC Division 1 RCIC MOV power source. Electrical design considerations such as battery capacity, short circuit current, voltage, and fuse sizing have been adequately addressed. The reinstallation of the flow orifice was previously analyzed as part of the original design of the plant and, therefore, does not change the

Safety Evaluation 91-0127 REV1 (continued):

function of the RCIC system. The installation of the lanyard potentiometers on E5150F044 and E5150F045 has no effect on the design or function of the valves or the RCIC system.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0003 UFSAR Revision No. 6

Reference Document: EDP 10820 Section(s) 6.2; 9.3; 11.4

Table(s) 3.2-1

Figure Change [X] Yes [ ] No

Title of Change: Hardened Torus Vent Installation

SUMMARY:

The hardened torus vent allows a controlled torus airspace release to the environment to prevent exceeding the primary containment pressure limit during severe accidents that are outside of the design bases. This modification:

1. Extended the original torus vent through the reactor building wall into a dedicated stack.

2. Installed two torus vent secondary containment isolation valves (TVSCIV) to maintain secondary containment integrity when venting is not required.

3. Installed a radiation monitor to alert the operators that a release is in progress.

4. Supplied non-interruptable air system (NIAS) instrument air to the existing torus primary containment air operated valves .

5. Modified the torus vent pipe support system as required by the new design conditions.

Prior to this modification, Fermi 2 emergency operating procedures directed the operators to vent the torus to the refueling floor inside the secondary containment. It was expected that the discharge would subsequently blow out siding and escape to the environment. This event would breach secondary containment integrity and contaminate the reactor building. The hardened torus vent does not challenge secondary containment integrity and allows reactor building accident recovery activities to continue unhampered due to the absence of radioactive particulates and gases in the building. This change is a recommendation of the NRC Mark I Containment Performance Improvement Program and fulfills the requirements of NRC Generic Letter 89-16.

Torus venting is used to prevent overpressurization damage to primary containment during severe accidents that are beyond any design basis accident. It is only used after all other means of containment cooling are ineffective and primary containment is jeopardized. The design of this system ensures that the primary and secondary containment design bases are not compromised while

Safety Evaluation No. 92-0003 (continued):

the system is in standby. The TVSCIVs are designed to assure the availability of the standby gas treatment system (SGTS) when torus venting is not performed. The TVSCIVs are redundant, supplied by divisional power supplies, and fail closed. They are normally positioned closed. The TVSCIVs are operated by a keylock switch and are administratively controlled. Therefore, inadvertent operation of the TVSCIVs is prevented. Opening the TVSCIVs renders the SGTS inoperable. However, temporary loss of the SGTS is acceptable because the system is no longer needed to remove post-accident fission products from secondary containment when primary containment is vented to the environment. Plant emergency operating procedures govern the use of these valves. Changing the source of instrument air for the torus primary containment isolation valves does not change their control logic. These valves fail closed on a loss of instrument air and, therefore, do not rely on pneumatics to achieve their safety objectives. Sufficient NIAS capacity exists for the new air usage loads introduced by these valves. The torus primary containment isolation valves still isolate during a postulated LOCA. However, the isolation signal is bypassed when torus venting is required. Bypassing the isolation signal is governed by Fermi 2 emergency operating procedures. The piping system is designed as a Class D, Seismic I system and has been reanalyzed at the new operating conditions.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0005 REV1 UFSAR Revision No. 6

Reference Document: EDP 12107 Section(s) 10.4; 12.1

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Installation of Condensate Demineralizer CFD-H

SUMMARY:

This modification installed the eighth condensate demineralizer, CFD-H. This change will improve the condensate system's reliability and ensure quality condensate. The increased demineralizer capacity decreases the resin loading resulting in fewer resin backwashes and less radioactive waste. The additional capacity also covers the increase in condensate flow expected by the Power Uprate Program implemented during the third refueling outage.

This modification does not change the function of the CFDs. The new CFD is similar in design to the seven original CFDs to reduce the impact on operations and maintenance. However, since many of the original components are not available and design enhancements such as microprocessor-based programmable logic, flow straighteners and increased diameter septa have been instituted, an identical like-for-like replacement was not possible. The condensate, demineralized water, and air systems will not be overloaded by the new CFD. The additional HVAC loading is insignificant compared to the overall HVAC system capacity. The new CFD has no impact on radiological conditions because it is contained in its own shielded cell and valve gallery like the original CFDs. The microprocessor-based programmable logic controller (PLC) and digital flow controller have been reviewed against the concerns raised in the NRC memorandum, "Analog to Digital Instrumentation Replacements Under the 50.59 Rule", dated July 1, 1991. The new PLC and flow controller do not create the potential for a new type of malfunction of equipment important to safety because the new CFD is non-safety related, is not a divisional system, and is not required for safe shutdown. In addition, the new CFD protective interlocks and permissives are hard-wired like the original CFDs. The new CFD does not increase the probability or consequences of a loss of feedwater flow transient resulting from a loss of all CFDs and CFD bypass flow. The new CFD meets the regulatory positions in Regulatory Guide 1.56, Revision 1 (July 1978); water purity acceptance criterion 1 of Standard Review Plan Section 10.4.6; and the water chemistry control requirements of General Design Criterion 14 of Appendix A to 10CFR Part 50. The Technical Specification limits for reactor coolant chemistry in TS 3/4.4.4, Table 3.4.4-1 are not challenged by this change.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0006 UFSAR Revision No. 6

Reference Document: EDP 12915 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: North Reactor Feedwater Pump (RFP) Suction Strainer Drain Line Cap Installation

SUMMARY:

This modification cut the North RFP suction strainer drain line and added a threaded pipe cap. This line was originally routed to an equipment drain. This modification stops leakage past the existing drain isolation valve N2100F034A resulting in decreased steam and water leakage into the feedwater pump room and equipment drain system.

This modification does not affect any equipment important to safety. Any steam leak caused by the failure of N2100F034A and the pipe cap is bounded by the UFSAR feedwater line break analysis.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0008 REV1 UFSAR Revision No. 6

Reference Document: EDP 12542 Section(s) 15.7

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Refueling Bridge Mast, Grapple, and Gimbal Assembly Replacement

SUMMARY:

This modification replaced the original refueling bridge mast, grapple, and gimbal assembly. The original General Electric NF-400 mast triangular lattice construction was too flexible resulting in mast bracing failure and difficulty in precisely locating the mast over a target. The new General Electric NF-500 mast is of tubular construction that is structurally stronger and more rigid. The new mast is also 400 lbs. heavier than the original mast. The new grapple is similar to the original grapple except that it is connected to the mast with 4 bolts whereas the original grapple connects with three bolts. An adapter plate has been provided so that the original grapple can be installed as a replacement in the future. The new grapple has provisions for the installation of a television camera and has improved lighting. The new gimbal assemble is stronger to accommodate the heavier weight of the new mast.

The grapple design is not changed by this modification. The hoist cable stresses associated with the new mast were verified and found acceptable. The fuel damage resulting from a postulated fuel assembly drop over the core that includes fuel bundles, the new grapple, and the new mast is bounded by the UFSAR fuel handling accident analysis reviewed and accepted by the NRC in Fermi 2 Safety Evaluation Report NUREG -0798 Section 15.2.3.4. Seismic analysis of the refueling platform indicates that the structural integrity of the platform is not degraded by the additional weight and that stresses are below allowable stress values. The refueling platform interlocks for unsafe operation over the reactor vessel during control rod movements; grapple travel limits; and grapple hook engagement with hoist power are not changed by this modification. However, the hoist load interlock and the slack cable cutoff setpoints have been changed to take into account the additional weight of the new mast. These setpoint changes were incorporated into Technical Specification 4.9.6 by Technical Specification Amendment 86.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0015 UFSAR Revision No. 6

Reference Document: EDP 12538 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Installation of Refueling Floor Condensate Supply Tap

SUMMARY:

This modification added a 1" pipe, gate valve, and pipe cap to the reactor building condensate header located on the fifth floor of the reactor building. The piping was hot-tapped into the 3" elbow adjacent to valve G4100F015 located in the level control pit. This allows operations to use condensate for deconning and maintaining the steam dryer, reactor cavity walls, and dryer / separator pit walls in a wet condition during a refueling outage. In previous refueling outages, demineralized water was used. However, the use of demineralized water resulted in added plant water inventory. This made it difficult to operate as a zero discharge plant and placed additional demands on the makeup demineralizer and the potable water plant. The use of condensate eliminates these problems.

The new line is located away from equipment important to safety. The condensate storage and transfer system and the fuel pool cooling and cleanup (FPCCU) system are not required for the safe shutdown of the plant and this change does not affect their design functions. Internal flood protection in the level control pit is maintained because the new smaller 1" line taps into a larger 3" condensate line. Provisions were made to minimize the amount of metal shavings generated during installation. However, these shavings would not have any impact on the FPCCU system because they would be removed by the screen at the bottom of the surge tank, the Y-strainers G4102D002A and B located upstream of the FPCCU recirculation pumps, or the FPCCU demineralizers.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0019 UFSAR Revision No. 6

Reference Document: EDP 12738 Section(s) 6.3

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: High Pressure Coolant Injection (HPCI) Auxiliary Oil Pump Logic Modification

SUMMARY:

This modification allows the HPCI system to be available and operable while operating the auxiliary oil pump with HPCI system in standby. The auxiliary oil pump is run with the HPCI system in standby to purify the HPCI oil. This modification adds a HPCI turbine steam supply isolation valve E4150F001 position interlock to the ramp generator signal converter logic. Prior to this modification, running the auxiliary oil pump would open HPCI turbine stop valve E4100F067 and cause the ramp generator signal converter to initiate and saturate to maximum demand. The ramp generator would then cause the HPCI turbine control valve E4100F068 to open. If a HPCI initiation signal was received, E4150F001 would open and, because E4100F067 and E4100F068 were already open, a high steam flow trip could occur. As a result, the HPCI system could not be considered operable when the auxiliary oil pump was running with the HPCI system in standby. This change modifies the logic such that the ramp generator will initiate when both E4150F001 and E4100F067 are not fully closed. Therefore, E4100F068 remains closed with the auxiliary oil pump running and HPCI in standby and, if HPCI is initiated, the possibility of a high steam flow trip is avoided.

This modification does not affect the operation or function of the HPCI system other than to make initiation of the ramp generator signal converter logic dependent on E4150F001 position. It does not impact other systems that are important to safety. The change creates a design that is similar to the existing E4100F067 contact arrangement. The new components introduced by this change affect the operability of the HPCI system. However, when the reliability of these components is compared to overall HPCI reliability, their impact is negligible.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0026 UFSAR Revision No. 6

Reference Document: EDP 12774 Section(s) N/A

Table(s) 3.2-1

Figure Change [X] Yes [ ] No

Title of Change: Emergency Diesel Generator (EDG) Air Receiver Tank Safety Relief Valve (SRV) Replacement

SUMMARY:

The ASME Section III EDG Air Receiver SRVs R3000F035A through D and R3000F036A through D have been replaced with ASME Section VIII SRVs. The original SRVs have had multiple bench test failures, are difficult to repair, and are expensive to replace. The new SRVs will improve system reliability and reduce maintenance costs.

The replacement SRV ASME code standard is equivalent to the ASME code standards specified for the original SRVs. The replacement SRVs were purchased as CQ quality valves. However, they were tested and dedicated by Detroit Edison in accordance with EPRI Standard NP-7218 in order to upgrade them to QA level 1. The function of the new SRVs and the lift/reseat setpoints are identical to the original SRVs. Therefore, this change does not affect the operation of the EDG air start system. The new valves will maintain pressure boundary integrity for EDG starting during a design basis earthquake.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0028 UFSAR Revision No. 6

Reference Document: EDP 4536 REV A Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Addition of Condensate Storage and Transfer (CSAT) System Cold Fill Line Check Valves

SUMMARY:

This modification added check valves on the CSAT system cold fill lines upstream of north and south cold fill line isolation valves P11F193 and P11F194. A pressure gage was also installed in the north cold fill line. The cold fill line isolation valve P1100F341 valve position has been changed from normally open to normally closed. The new check valves and P1100F341 valve position change provide triple isolation between the high pressure (1250 psig) reactor water cleanup (RWCU) system and the low pressure (150 psig) CSAT system. This provides additional assurance that the RWCU system does not overpressure the CSAT system. Overpressurization of the CSAT system results in instrumentation damage and water spills due to CSAT system pressure relief valve lifts. The new pressure gage is used for testing the new check valves and monitoring their performance.

This modification has no impact on the CSAT tanks or the function of the RWCU system. It does not present any potential for damage to safety related equipment.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0032 UFSAR Revision No. 6

Reference Document: EDP 12870 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Replacement of Emergency Diesel Generator (EDG) Local KVAR Meters

SUMMARY:

This modification replaced the KVAR meters and added transducers at local panels R3000S005 through 8 for EDGs 11 through 14, respectively. The original electro-mechanical local meters were replaced because their readings differed from the main control room meter readings by an unacceptable amount. The new meters are an electronic design similar to the control room meters and are installed at the existing local KVAR meter location. Transducers are installed to make the circuitry similar to that of the control room meters. The new design ensures that the local meters and control room meters receive similar electronically generated signals which result in meter reading deviations within acceptable limits.

This modification improves the reliability of the local KVAR instrumentation. It does not change the function or operation of the EDG system. With the exception of input KVAR values, the standard calibration and testing procedures have not changed. There is no change to analyses contained in the UFSAR or SER.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0034 UFSAR Revision No. 6

Reference Document: EDP 13229 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: High Pressure Coolant Injection (HPCI) Barometric Condenser Condensate Pump Discharge Piping Modifications

SUMMARY:

The HPCI barometric condenser condensate pump discharge piping has been modified by replacing the 1/2" diameter pipe with 1" pipe and adding a 1" manual globe valve E4100F215. This eliminates nuisance alarms on control room annunciator alarm 2D51, "HPCI Pump Suction High". The nuisance alarms were caused by the inablity of the original discharge piping to pass adequate condensate flow to the clean radwaste system. As a result, the HPCI booster pump suction pressurized to the 70 psig setpoint of annunciator alarm 2D51. The 1" replacement line increases flow capacity and E4100F215 allows condensate flow to be adjusted to avoid condensate pump runout. E4100F215 has been added to the locked valve program as a locked throttled valve to prevent mispositioning.

This change does not impact the operation or function of the HPCI system or any other system used to mitigate accidents. The barometric condenser is not required to maintain HPCI operable. This change was made to the Non-Q, Seismic II/I, Group D portion of the discharge piping. This modification does not change control room annunciator alarm 2D51 setpoint and the alarm continues to provide assurance of the integrity of the reactor coolant system pressure isolation valves.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0036 UFSAR Revision No. 6

Reference Document: EDP 13300 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Installation of Circulating Water (CW) Pump Alternate Cooling Water Supply Taps

SUMMARY:

This modification installed an alternate cooling water supply tap on each CW pump. Each tap consists of a pipe tee, gate valve, and pipe cap. The tap assemblies are located downstream of the general service water (GSW) cooling water isolation valves P4100F801 through 805. In addition, valves have been added to the existing GSW supply header drain valves P4100F808 and 809 to allow the header to be flushed after CW pump cooling loads have been transferred to the alternate cooling supply. The need for alternate cooling water supply connections has arisen from a concern that the existing GSW 6" supply header and associated branch piping is being blocked by zebra mussel debris. The amount of debris is expected to increase after Clamtrol chemical treatment which is used to kill the adult zebra mussels.

This modification has no adverse impact on any existing accident scenarios. The alternate cooling water source remains the same, i.e. Lake Erie and the GSW pumps. The loss of cooling water will result in a loss of one or more CW pumps and a loss of condenser vacuum. Loss of condenser vacuum due to the loss of CW pumps has been previously evaluated in the UFSAR. The new connections do not affect any safety related, important to safety, or safety related support equipment. Their presence or failure has no impact on the ability of the GSW system to service other plant loads or the ability of EECW and EESW to perform their function upon loss of GSW.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0038 UFSAR Revision No. 6

Reference Document: EDP 12797 Section(s) 8.1; 8.2

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Main Transformer 2B Replacement

SUMMARY:

In order to avoid a sudden main transformer failure, the original Ferranti main unit transformer 2B was replaced with a Cooper transformer during the third refueling outage. The original transformer experienced degraded conditions due to corona discharge and possible insulation breakdown. The original main unit transformers were judged to have a relatively high probability of failure and inadequate short circuit capability. Main unit transformer 2A was replaced in December, 1991 and its replacement is evaluated in Safety Evaluation 91-0114 (See Fermi 2 Safety Evaluation Summary Report, 1991).

The replacement transformer has the same voltage ratio as the original transformer. The replacement transformer MVA rating is higher than the original transformer. However, the increased transformer capacity has no effect on associated equipment or systems. The components and circuits associated with the main unit transformers were reviewed and either accepted as is or modified to accept the change. Engineering considerations such as load sharing and circulating current between transformers 2A and 2B; voltage regulation; short circuit current; protective relaying; auxiliary power feeder rating; power uprate impact; and fire protection deluge system adequacy were reviewed and found acceptable. Therefore, there is no change to the failure mode of the main unit transformer and associated systems, components, and circuits.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0042 UFSAR Revision No. 6

Reference Document: EDP 10145 Section(s) 7.7

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Installation of Reactor Recirculation Pump (RRP) Speed Limiter 4

SUMMARY:

This modification added RRP speed limiter 4, a manual defeat switch, and associated relay logic to the recirculating flow control system (RFCS). This change provides for a defeat of the runback 2 and runback 3 logic on a loss of heater drains and/or one feedwater pump trip in single loop operation to prevent entry into the thermal hydraulic instability region. The change is as follows:

- An automatic defeat switch has been provided for the runback 2 and runback 3 logic of each RRP to prevent a runback to 37% of rated speed during single loop operation RRP trip events. RRP speed limiter 2 and limiter 3 are unnecessary for all single loop operations.

- An automatic runback to 75% of rated speed (runback 4) has been added to each RRP speed control circuit to ensure compliance with Technical Specification 3.4.1.1 operating RRP speed limits during single loop operation. This ensures that instability region B (> 77% rod line and < 45%core flow) is avoided in the event of a RRP trip.

- A manual defeat switch has been installed at COP H11P603 to allow a defeat of runback 2 and runback 3 logic during reactor startup, shutdown, and power maneuvers. For power levels below approximately 65% RTP, runbacks initiated by RRP speed limiter 2 and limiter 3 are unnecessary because steam and feedwater flowrates are in equilibrium and the runbacks can introduce undesirable plant transients. When heater drains pumps are out of service (at reactor power less than 50 to 55% RTP), limiters 2 and 3 limit core flow to approximately 48% if both recirculation pumps are in service. This results in all low power maneuvers being performed in regions close to the thermal hydraulic instability region.

- An automatic interlock circuit has been installed as part of the runback 4 logic to prevent automatic actuation of runback 4 if runback 2 or runback 3 are already in progress.

Safety Evaluation No. 92-0042 (continued):

- o Reactor recirculation loop A and B indicating lights at COP H11P603 have been modified to provide indication that runback 2 and runback 3 logic has been armed.
- o An alarm window has been installed at COP H11P603 to inform the operators that the manual switch is in the defeat position or runbacks 2 and 3 have been automatically defeated.

This change will improve plant availability and capacity factor by avoiding unnecessary protective trips and by eliminating unnecessary restraints on reactor maneuverability.

All runback 4 and manual defeat circuitry is the same quality level as the existing RFCS components. This change does not impact the performance of the RFCS or any other plant system used to mitigate accidents. No safety related equipment will be directly actuated by the runback 4 logic. Runback 4 will only be in effect during single loop operation. The runback 2 and runback 3 manual defeat switch will only be used during startup and shutdown for short periods of time to facilitate reactor maneuverability. The limiter 4 setpoint is within single loop operation Technical Specification setpoint limits. RFCS operation and malfunctions are bounded by the existing UFSAR Chapter 15 analyses. Therefore, there is no change to offsite or onsite radiation doses or radioactive material releases.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0043 UFSAR Revision No. 6

Reference Document: EDP 7579 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Removal of Post Accident Sampling System (PASS) Iodine and Particulate Sampling Capabilities

SUMMARY:

This change removed the particulate and iodine sampling components from the gas portion of the PASS by eliminating the PASS gas sample three way valve P34F006, associated tubing, and the particulate/iodine cartridge drawer P3400Y005. The gas sample is tubed directly to the gas cooler P3400B003 inlet. The PASS continues to have the capability to obtain containment atmosphere noble gas and hydrogen samples. The iodine and particulate sample results needed to assess core damage are obtained from reactor coolant system (RCS) samples. This change enhances PASS reliability and performance by reducing the chance for equipment or logic/interlock failures, removing the chance for leakage past cartridge drawer seals, and decreasing inleakage problems due to the reduced number of fittings.

This modification does not change the PASS liquid sampling capabilities. The removal of the PASS gaseous particulate and iodine sampling components does not degrade the capability to respond to or mitigate core damage during an accident because RCS sample particulate and iodine radiochemistry analysis results are used to estimate core damage. This change does not affect the Emergency Plan because PASS gaseous particulate and iodine samples are not used to estimate offsite dose or to estimate containment radiation when the containment high range radiation monitors are inoperable or offscale.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0044

UFSAR Revision No. 6

Reference Document: EDP 12990

Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: General Service Water (GSW) System Isolation Valve Replacement

SUMMARY:

This modification replaced GSW reactor building closed cooling water (RBCCW) temperature control valve P42F400 isolation valves P4100F052 and P4100F054 and GSW turbine building closed cooling water (TBCCW) temperature control valve P43F402 isolation valves P4100F084 and P4100F086. the original butterfly valves leaked and prevented maintenance on the RBCCW and TBCCW temperature control valves. The new butterfly valves are an improved design incorporating a double seal for improved isolation capability.

This modification affects the non seismic, non-Q, piping group D portion of the GSW system. This change is a functional equivalent replacement and it does not affect the design bases, functions, or operation of any structures, systems or components. All previous accident analysis results remain unchanged. Minor material differences between the original valves and the new valves have been evaluated and found acceptable.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0048 REV1 UFSAR Revision No. 6

Reference Document: EDP 13231 Section(s) 8.3

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Residual Heat Removal (RHR) Pump D Motor Replacement

SUMMARY:

This modification replaced the original RHR pump D motor. The original motor has a 2000 hp rating and the new motor has a 2250 hp rating. The original motor was replaced in order to perform the disassembly, inspection, and refurbishment activities required as corrective action by Deviation Event Report DER 88-1197 without impacting the third refueling outage schedule. DER 88-1197 addressed the large size GE induction motor surge ring support bracket failure concerns in NRC Information Notice 87-30 and General Electric Rapid Information Communication Information Letter RICSIL-16.

The replacement motor is seismically and environmentally qualified. It has been installed in accordance with Class 1E seismic requirements. Evaluation of the new motor assures that:

- The motor is capable of continued operation in LOCA pressure, temperature, and radiation conditions.
- The motor is designed for the 40 year operating life of the plant and for a single postulated 100 days of continuous operation.
- The motor has been tested to criteria similar to the original motor preoperational test criteria.
- The motor can operate during and after a seismic event.
- The higher locked rotor current of the new motor (1850 amps versus 1530 amps) is bounded by the results of emergency diesel generator preoperational testing.

Although the new motor is 7.8% heavier than the replacement motor, there is no impact on structural long-term load control and the effect on soil and foundation bearing capacity is negligible.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0049 UFSAR Revision No. 6

Reference Document: EDP 12934 Section(s) N/A

Table(s) 6.2-2

Figure Change [X] Yes [ ] No

Title of Change: Installation of Local Leak Rate Test (LLRT) Boundary Valve for E4150F002

SUMMARY:

This modification installed high pressure coolant injection (HPCI) steam line drain valve E4150F080 to improve local leak rate testing of HPCI steam line inboard isolation valve E4150F002. E4150F080 and the installation of a test blank flange in the upstream flow orifice E4150N400 allow E4150F002 to be leaked tested in the accident direction. Previous LLRTs were performed by the conservative bonnet testing method. This method includes both gate valve wedge seats and bonnet leakage in the overall test results. This change reduces E4150F002 LLRT failures and subsequent valve refurbishment because accident direction testing measures the more realistic downstream wedge seat leakage. Successful testing is expected because this valve has performed satisfactorily as a boundary valve for HPCI steam line outboard isolation valve E4150F003 LLRTs.

This modification does not change the function of the HPCI steam line drain or steam supply line. The valve design rating envelopes the steam line drain operating and design accident conditions. Piping loads and stresses have been analyzed and are acceptable. Failure of the valve body or bonnet is the equivalent to breaking a 1" steam line within containment. This accident is bounded by the steam line break within containment analysis. E4150F080 has been added to the Locked Valve Program to ensure that a drainage pathway remains open except for testing and maintenance. Removal of the E4150N400 test blank flange is verified by performance of surveillance procedure NPP 24.202.02, "HPCI Flow Rate Test at 165 psig Reactor Steam Pressure".

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0050 UFSAR Revision No. 6

Reference Document: EDP 12120 Section(s) 3.7; 3.9

Table(s) 3.9-18; 3.9-20; 3.9-21
3.9-23; 3.9-24; 5.2-3

Figure Change [X] Yes [ ] No

Title of Change: Drywell Primary Pressure Boundary Piping Snubber Reduction

SUMMARY:

This modification removed 53 snubbers on the following primary pressure boundary piping:

- Main steam lines A and D.
- High pressure coolant injection (HPCI) branch connection to main steam line A.
- Safety relief valve (SRV) discharge lines to main steam lines A and D.
- Recirculation loops A and B.
- Residual heat removal (RHR) supply and return piping to recirculation loops A and B.
- Reactor water cleanup (RWCU) system piping.

Seven of the removed snubbers were converted to struts and minor modifications were made to piping mirror insulation.

This reduction was achieved through application of current piping technology and analysis methods, acceptable to the NRC, that optimize nuclear plant piping suspension configurations. Snubber reduction results in: 1) a decrease in costs associated with snubber periodic maintenance, inspection, and testing; 2) a decrease in personnel radiation exposure levels; and 3) a reduction in the possibility of snubber failures that results in a decreased potential for extended outages and in increased piping system reliability.

The change to the piping support configuration has no impact on the function or operation of the above systems or any interfacing systems. There is no impact on the safety related function of any component and there is no impact on any accident scenarios evaluated in the UFSAR. The insulation modification has no significant effect on containment temperatures. The snubber reduction design calculations utilize the design criteria and design loadings provided in the UFSAR or as specified in NRC regulatory documents. The revised pipe stresses

Safety Evaluation No. 92-0050 (continued):

and component loadings do not exceed the UFSAR allowable values. Piping in-line components and terminal connections have been evaluated for any revised loadings. Rattlespace with adjacent equipment has been reviewed and is acceptable.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0057

UFSAR Revision No. 6

Reference Document: EDP 13251

Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Feedwater Flow Calibration Sampling Probe Installation

SUMMARY:

This modification installed four sampling taps in the non-safety related portion of the condensate and feedwater system to monitor feedwater flow performance using a tracer calibration verification technique. The new taps are needed because original sampling tap locations do not meet the 420' mixing length required by this technique. Two inlet taps are installed on the 24" feedwater lines downstream of the No. 6 north and south feedwater heaters common outlet header and two outlet taps are installed on the 20" feedwater piping upstream of the B21F076A and B check valves. The new tap locations provide a mixing length of 520'. The inlet taps inject a metered (<1 gpm) water/chemical tracer mixture into the the piping for a short period of time (<1 hour). The outlet taps serve as the removal point for the mixed tracer and feedwater allowing the tracer technique to compare or calculate a feedwater flow rate to an accuracy of 0.5%.

The condensate and feedwater system is not required for safe shutdown or emergency safety features support functions. This modification complies with the condensate and feedwater system fabrication design requirements by utilizing welded joints and high quality materials to protect against pipe breaks. Any possible break is bounded by the UFSAR Chapter 15 feedwater pipe break analysis. Safety related equipment is not impacted by the inlet taps because they are located in the No. 6 north feedwater heater room. The outlet taps are located in the reactor building steam tunnel. This location has been previously analyzed for adverse environments and the failure or malfunction of safety related equipment has been analyzed for the bounding pipe breaks.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0063

UFSAR Revision No. 6

Reference Document: EDP 13500

Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Removal of Reactor Water Cleanup (RWCU) Pump A Suction Strainer

SUMMARY:

This modification removed the RWCU pump A suction strainer G3303D003A along with its associated valves G3300F137A, F138A, and F139A. The strainer body was replaced with a spool piece. The strainer was originally installed for post-construction cleanup and the strainer baskets were removed after plant preoperational testing was completed. The empty strainer body collected radioactive debris and its removal eliminated a major radiation source within the RWCU pump A room.

This modification does not change the function or operation of the RWCU system and does not impact the function of any other plant safety system. Reactor water chemistry is unaffected by this change. The spool piece is similar in design to the existing RWCU piping. A postulated failure of the spool piece is bounded by the existing high energy line break analyses.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0065 UFSAR Revision No. 6

Reference Document: EDP 12932 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: 130 VDC Distribution Panel Circuit Fuse Size Change

SUMMARY:

This modification changed the fuse sizes for 130 VDC distribution Panel 2PC3-16, positions 3 and 4. It also adds the fuses for these circuits and 130 VDC distribution panel 2PC3-17, position 11 to Detroit Edison fuse sizing specification design documents. Detroit Edison drawing 6SD721-2530-12 (UFSAR Figure 8.3-11) has been revised to remove the fuse size for these circuits and a note has been added to state that the fuse sizes for these circuits are specified in the fuse sizing design documents. The original fuse sizes for 130 VDC distribution panel 2PC3-16 positions 3 and 4 were 15 amperes. A recalculation of the fuse sizes using the Detroit Edison fuse sizing design documents indicated that 10 ampere fuses will provide the necessary cable short circuit protection, overcurrent protection, and fuse coordination for the subject circuits.

The fuse size changes are based on the design criteria in Detroit Edison fuse sizing specifications. The fuse size changes do not impact the overall design basis, operation, or function of any safety related or balance of plant systems. Review of the Fermi 2 Safety Evaluation Report Subsection 8.3.2 indicates that the fuse replacement has no impact on the 130 VDC distribution panels.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0066 REV1

Reference Document: EDP 13505

UFSAR Revision No. 6

Section(s) N/A

Table(s) 6.2-2

Figure Change [ ] Yes [X] No

Title of Change: Reactor Water Cleanup (RWCU) Supply Inboard Containment Isolation Valve Body Tap Installation

SUMMARY:

This modification installed a 1/2" carbon steel nipple on the RWCU supply inboard containment isolation valve G3352F001 valve body to allow body tap local leak rate testing (LLRT) of the valve. This LLRT method permits testing by introducing the test pressure between the valve seats. This change allows the option of either performing body tap LLRTs or the original accident direction testing.

This modification does not interfere with the function of G3352F001. There is no change to the valve actuator, power supply, or instrumentation logic. There is no adverse affect on the integrity of the RWCU piping. This modification does not prevent the performance of the normal G3352F001 accident direction LLRTs. A nipple failure is enveloped by the break size spectrum evaluated in the small break LOCA analyses discussed in UFSAR Sections 6.2 and 6.3. Body tap LLRTs comply with 10CFR Appendix J requirements and are accepted by the NRC in the Fermi 2 Safety Evaluation Report.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0069 UFSAR Revision No. 6

Reference Document: EDP 13525 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Installation of Condenser Waterbox Draindown System Recirculation Line

SUMMARY:

This modification installed an 8" butterfly valve, a 6" butterfly valve, and cross tie piping in the condenser waterbox draindown system to provide the capability for recirculating water between the north and south circulating water inlet waterboxes. In the recirculation mode, a suction is taken on the south waterboxes by the existing waterbox draindown pump and water is discharged to the north waterboxes. This recirculation process allows for chemical treatment or any other evolution that requires recirculation of water on the circulating water side of the condenser.

The condenser waterbox draindown system is not required for plant operation or shutdown and is not located in the vicinity of equipment required for plant shutdown. Failure of the new valves and piping is bounded by the existing turbine building flood analysis in UFSAR Subsection 10.4.5.3.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0070

UFSAR Revision No. 6

Reference Document: EDP

Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: General Service Water (GSW) Fan Cooling Unit Outlet Valve Replacement

SUMMARY:

This modification replaced the GSW fan cooling unit outlet valves P4100F348, F350, F352, and F353 globe valves with gate valves. The original valves were eroded beyond repair. The replacement valves were installed on flanged spool pieces to facilitate valve removal.

This change does not affect the design bases, functions, or operation of any structures, systems, or components. All previous accident analysis results remain unchanged. The new gate valves are an acceptable replacement because their valve position is always full open and, therefore, they do not function as throttle valves.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0075 UFSAR Revision No. 6

Reference Document: EDP 13609 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Reactor Recirculation Pump (RRP) Suction and Discharge Valve Bonnet Vent Assembly Removal

SUMMARY:

This modification removed the bonnet vent assemblies from RRP suction valves B3105F023A and B and RRP discharge valves B3105F031A and B. Removal of these vent assemblies eliminates their susceptibility to high-cycle fatigue failures and the resultant reactor coolant leakage. The original purpose of the bonnet vent assemblies was to vent the suction and discharge valve bonnets during recirculation loop fills and to relieve bonnet pressure if the valves became pressure locked. However, these vent paths have not been used during the three previous operating cycles.

This change does not affect the design basis, functions, or operation of any structures, systems, or components. The issues in General Electric Service Information Letter SIL No. 368, "Recirculation System Isolation Valve Locking" are still addressed in that the below-seat drain line or the loosening of valve packing can be utilized to relieve bonnet pressure.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0078

UFSAR Revision No. 6

Reference Document: EDP 13289

Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Main Condenser Waterbox Sample Line Replacement

SUMMARY:

This modification replaced and rerouted the main condenser northeast inlet, southeast inlet, and east outlet water box sample lines. The original 1/4 inch sample lines were excessively long which resulted in little or no sample flow. The new 3/8 inch sample lines are routed to new sample locations and provide accurate monitoring of the circulating water (CW) system to prevent fouling and lost plant capacity.

The new sample lines do not interact with and are not located near systems required for safe shutdown of the reactor. Failure of these sample lines does not affect the function of the CW system because it has adequate makeup capacity to overcome any sample line leakage. Sample line failure is bounded by the turbine building flooding accident analysis described in UFSAR subsection 10.4.5.3.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0085

UFSAR Revision No. 6

Reference Document: EDP 11333

Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Residual Heat Removal (RHR) Cooling Tower Low Inlet Temperature Alarm Removal

SUMMARY:

This modification removed the control room annunciators 7D7 and 7D8 RHR cooling tower low inlet temperature alarm input signals to eliminate nuisance alarms. The purpose of these input signals was to prevent RHR cooling tower icing by alerting the operator that RHR cooling tower fan operation is prohibited when the cooling tower inlet water temperature falls below 60°. The RHR cooling tower inlet temperature is measured by thermocouples that are designed to measure flowing water. When the cooling towers are not in operation, the inlet pipes are empty and the thermocouples measure ambient temperature. This condition caused nuisance alarms when the ambient temperature dropped below the RHR cooling tower low inlet temperature setpoint.

This modification does not affect the function or performance of the RHR complex service water system; the UFSAR and Fermi 2 SER assumptions; or the Fermi 2 Technical Specifications. Plant procedure SOP 23.208, "RHR Complex Service Water System", continues to provide the administrative controls to prevent RHR cooling tower icing by prohibiting cooling tower fan operation when the RHR cooling tower inlet water temperature is less than 60°F. In addition, the RHR reservoir low temperature (<43°F) alarm inputs to control room annunciators 7D7 and 7D8 alert the operators of potential icing conditions.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0087 UFSAR Revision No. 6

Reference Document: EDP 13688 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Reactor High Pressure Alarm Annunciator Input Change

SUMMARY:

This modification changed the input source for control room annunciator 3D168, "Reactor Pressure High", from wide range reactor pressure transmitters C32N005A and B to narrow range reactor pressure transmitter C32N008. This change provides increased pressure setpoint accuracy.

The indication from both the wide range and narrow range reactor pressure transmitters is still available to and monitored by the operator. This modification does not change the 1045 psig alarm setpoint. The instrumentation involved in this modification is used for indication only. It does not perform a safety related function and is not used to mitigate accidents analyzed in the UFSAR.

END OF EDP SECTION

FERMI 2

SAFETY EVALUATION SUMMARY REPORT

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PROCEDURES, TESTS, AND EXPERIMENTS

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0045 UFSAR Revision No. N/A

Reference Document: UFSAR 11.7 Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Storage of Dewatered Radwaste at the On-Site Storage Facility (OSSF)

SUMMARY:

The evaluation justifies the long term storage of dewatered radwaste at the OSSF until it can be shipped to a licensed offsite burial facility. It is necessary to store radioactive waste at Fermi 2 because of the closure of the three existing burial sites to Michigan radioactive waste generators. This waste is fully dewatered or solidified at the OSSF portable station in polystyrene high integrity containers (HICs) and solidification liners. The containers and liners are capped and stored in the OSSF. The radioactive materials in these containers are the same as those analyzed in the original design and licensing of the OSSF. However, the physical form (dewatered material and cement solidification) and the containers (larger HICs) are different. The processing of radioactive waste in a container was evaluated in Safety Evaluation 88-0186.

The temporary storage of dewatered radioactive waste in the OSSF does not affect the liquid radioactive waste accident analyzed in the UFSAR because the waste is considered solid and does not increase the radioactive waste system radioactivity inventory. The consequences of a dewatering container spill or rupture has been evaluated in Safety Evaluation 88-0186 and they do not impact the liquid radioactive waste accident analysis. The OSSF is designed to contain any accidental spills. The dewatered radioactive waste is stable within a HIC. The OSSF structure and OSSF crane capacity are adequate to handle the load of a filled HIC. Temporary storage of dewatered radioactive waste does not affect fire, flood, or tornado protection described in the UFSAR.

SAFETY EVALUATION SUMMARY

Safety Evaluation No:	92-0004	UFSAR Revision No.	6
Reference Document:	LCR 92-009-UFS	Section(s)	3.1; 9.1; 13.5; A.5.29
		Table(s)	N/A
		Figure Change	[ ] Yes [X] No

Title of Change: ANSI N15.8 Commitment Revision in UFSAR

SUMMARY:

This change revised the UFSAR as follows:

1. UFSAR Section A.5.29

   The commitment to ANSI N15.8, "Nuclear Material Control Systems for Nuclear Powerplants", has been changed to state that only the applicable portions of the Code of Federal Regulations will be complied with and that the special nuclear material (SNM) control and accountability system will be covered by Fermi Management Directive FMD SE3, "Special Nuclear Material Accountability".

2. UFSAR Subsection 13.5.10

   Reference to the shipping of spent fuel has been deleted from this subsection because there are currently no procedures that adequately specify all actions required for shipping spent fuel.

3. UFSAR Subsections 13.5.1 and 13.5.10

   The "radioactive-materials-handling procedures" terminology has been replaced with "SNM control and accountability procedures" because the former terminology is no longer used. The position responsible for implementing the SNM control and accountability procedures has been changed from the Principal Engineer - Reactor to Supervisor - Nuclear Fuel to describe the current practice. The Principal Engineer - Reactor is responsible for fuel accountability procedures.

4. UFSAR Subsections 3.1.2.6.2, 3.1.2.6.3, and 9.1.1.2.1

   These subsections have been revised to state that new fuel "may" be stored in the new fuel vault. This change provides consistency with UFSAR Subsections 9.1.2.1 and 9.1.2.5 that currently acknowledge new fuel storage in the spent fuel pool and with UFSAR Subsection 1.2.2.15.1 that currently states that new fuel may be stored in a dry vault in the reactor building.

Safety Evaluation No. 92-0004 (continued):

There are no accidents associated with SNM control and accountability. The misplaced bundle accident is not affected because no changes are being made in the way fuel is handled or in the core verification process. The SNM control and accounting program will continue to be in compliance with applicable regulations. The program continues to cover:

- 10CFR70.51(a)(8) — Definition of a physical inventory.
- 10CFR70.51(b) — Maintaining records.
- 10CFR70.51(c) — Establishment and maintenance of written material control and accounting procedures.
- 10CFR70.51(d) — Physical inventory requirements.
- 10CFR70.53 — Material status reports.
- 10CFR70.54 — Nuclear material transfer reports.
- 10CFR73.67(g)(2)(ii) — Nuclear material transfer reports.
- 10CFR74.13 — Material status reports.
- 10CFR74.15 — Nuclear material transfer reports.

Applicable portions of ANSI N15.8 necessary for an effective and efficient SNM control and accountability system have been implemented. FMD SE3 establishes the requirements for SNM control and accountability and assigns responsibility for implementing those requirements.

The change to the description of the new fuel vault is a clarification and is, therefore, administrative in nature. This change makes the UFSAR more consistent with the SER (NUREG-0798). No change has been made to the method of new fuel storage and new fuel stored in the spent fuel pool storage racks is acceptable because the racks have been analyzed for new fuel.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0007 UFSAR Revision No. 6

Reference Document: LCR 92-005-UFS Section(s) 9A.2; 9A.5; 9A.6

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Revised Electrical Conduit Sealing Criteria

SUMMARY:

This evaluation justifies revising the following UFSAR subsections:

1. UFSAR Subsection 9A.2.3.1.1 has been revised to include a discussion of the revised conduit sealing criteria developed for the Wisconsin Electric Company and approved by the NRC. The criteria is as follows:

   o Conduits that terminate in junction boxes or other non-combustible enclosures need no additional sealing.

   o Conduits that run through an area but do not terminate in that area need not be sealed in that area.

   o Open conduits smaller than 2 inches diameter that terminate 1 foot or greater from the barrier need not be sealed.

   o Open conduits of 2 inches diameter that terminate 3 feet or greater from the barrier need not be sealed.

   o Conduits greater than 2 inches diameter should be sealed.

2. UFSAR Subsections 9A.2.2.3 and 9A.5.d.1.j have been revised to cross reference the portions of the fire hazard analysis which discuss internal conduit seals with UFSAR Subsection 9A.2.3.1.1.

3. UFSAR Section 9A.2, "References" has been revised to include the conduit seal fire test report and supporting test data; the NRC safety evaluation for the use of the conduit seal fire test report data; and the NRC acceptance letter for the conduit fire seal test data.

4. A note has been added to UFSAR Subsection 9A.6.8.2.1.c to state that electrical conduit penetration seals need not be inspected if they meet the requirements in UFSAR Subsection 9A.2.3.1.1 described above.

These changes only apply to the fire protection program. Electrical conduit seals will still be provided where required to maintain control center pressure

Safety Evaluation No. 92-0007 (continued):

boundary, secondary containment, and radiation barriers. The use of the new criteria does not affect the operability or function of the fire barriers. The criteria do not adversely affect any safety related systems or equipment. The criteria have been reviewed and approved by the NRC.

SAFETY EVALUATION SUMMARY

Safety Evaluation No:	92-0017	UFSAR Revision No.	6
Reference Document:	UFSAR	Section(s)	10.4
		Table(s)	N/A
		Figure Change	[ ] Yes [X] No

Title of Change: Changes to UFSAR Condensate Deaeration and Feedwater Quality Discussions

SUMMARY:

This evaluation discusses changes to UFSAR Subsections 10.4.1.1.3 and 10.4.7.1.2 concerning condensate deaeration and feedwater quality, respectively.

1. UFSAR Subsection 10.4.1.1.3

   UFSAR Subsection 10.4.1.1.3 has been revised to change the action level for injecting oxygen into the condensate from 20 ppb to 10 ppb oxygen. Review of plant specific operating data and the requirements established in the BWR Owner's Group indicate that there is no need to inject oxygen when oxygen levels fall below 20 ppb. If the main condenser deaerates the condensate to values consistently less than 10 ppb, actions will be taken to restore oxygen levels and/or evaluate the consequences consistent with the BWR Owner's Group Guidelines and site programs. Fermi 2 operating experience has shown that dissolved oxygen concentrations range from 10 to 50 ppb.

   This change is consistent with present plant operation. There is no appreciable difference in the corrosion rate of carbon steel components exposed to dissolved oxygen levels of 10 to 20 ppb. Corrosion product samples are routinely collected on site. These samples are analyzed and results show that the level of corrosion products is less than anticipated in UFSAR subsection 10.4.1.1.3. Therefore, this change does not affect the UFSAR feedwater line break analysis because a severed line break due to generalized corrosion is improbable. Reactor water dissolved oxygen is not affected because the oxygen contributed by feedwater is insignificant compared to oxygen contributed by radiolytic decomposition.

2. UFSAR Subsection 10.4.7.1.2 Feedwater Quality

   The UFSAR Subsection 10.4.7.1.2 description of methods for obtaining and analyzing dissolved and suspended solids has been revised to state

Safety Evaluation No. 92-0017 (continued):

that integrated discreet on-line sampling methods are used as opposed to grab and composite samples. By using the analytical methods of an integrated sampling program, feedwater and condensate system corrosion products can be detected to the parts per trillion level. This provides lower levels of detection and a higher degree of accuracy.

This change reflects improvements made in the sampling techniques used to ensure that the corrosion profile of the plant is being monitored effectively. Accident scenarios for small line breaks have been analyzed in UFSAR Subsection 15.6.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0018 REV1 UFSAR Revision No. N/A

Reference Document: 23.718.10 Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Evaporation of Contaminated Water

SUMMARY:

This evaluation justifies the evaporation of contaminated water through humidifiers enclosed in one or more cabinets of the plant laundry facility clothes dryers. The contaminated water was produced from previous wet-washing of clothing and ongoing plant cleaning activities. The contaminated liquid is transported to the dryer location in 55 gallon drums. It is then transferred to a 55 gallon drum which gravity feeds the contaminated water through float valves to the humidifiers. Approximately 2 inches of water is maintained in each humidifier tray where a self-wicking filter draws in the moisture. The moisture is evaporated when fans draw air through the filter. The moisture leaves the dryer cabinets, passes through a local high efficiency particulate absorbing (HEPA) filter, and is vented to the radwaste building HVAC system. The clothes dryer heater and drum are not used for this operation. The water is evaporated through a room temperature convection process rather than by heating. This enables the separation of solids from their liquid carrier and the evaporation of the liquid. The solids that remain on the humidifier filter are processed as solid radwaste and the vapors are processed as gaseous radwaste. This process eliminates the need to release the material as liquid radwaste and is in line with the Fermi 2 goal of eliminating liquid radwaste discharges to the environment.

Each humidifier is administratively controlled by:

1. Limiting the total activity of the liquid to be evaporated to 0.01 uCi/ml. This is the concentration in the source term for the liquid and solid radwaste systems accident analysis described in UFSAR Subsection 15.7.3.3.1.

2. Limiting the total activity for all humidifiers to 100 mCi.

This process does not interface with any equipment important to safety. There is no effect on the operability of the ventilation exhaust treatment system because the process does not impact the radwaste building HVAC HEPA filters. Maintenance activities such as humidifier filter changeout are controlled by the Fermi 2 Radiation Protection Program and are within the day-to-day operational capabilities of the plant staff. 10 CFR 20, Appendix B limits are

Safety Evaluation No. 92-0018 REV1 (continued):

not exceeded by this process. An airborne release (due to humidifier filter and tray dryout) would result in an increase in gaseous effluents that are within the scope of the expected radwaste building source terms described in UFSAR Subsection 11.3.2.6. The worst case release is within the expected annual plant laundry system releases. A liquid radwaste spill from the humidifiers or during transport to the evaporators would be collected by the floor drain system and processed with no unplanned release. The radwaste spill accident described in UFSAR Subsection 15.7.3 is not affected because the additional 220 gallons of liquid radwaste contributed by the humidifiers is insignificant when compared to the 272,000 gallons assumed in the accident analysis.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0020 UFSAR Revision No. 6

Reference Document: DER 92-0052 Section(s) 3.7 References

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Residual Heat Removal (RHR) Complex Floor and Wall Response Spectra Curves Revision

SUMMARY:

This evaluation justifies incorporating the current RHR complex floor and wall response spectra curves in the UFSAR. These curves are in Sargent & Lundy Report No. SL-3147, Revision 3 dated 4/15/83. The original curves, from Sargent & Lundy Report SLS No. 675 dated 8/23/73, are outdated and are based on a preliminary analysis. This change makes the UFSAR analysis consistent with the data used in the RHR complex structural, system, and component analyses. It is in accordance with the proposed remedial corrective action in Deviation Event Report DER 92-0052.

This change is for documentation purposes only. The design of the RHR complex structures, systems, and components is not affected by the UFSAR figure replacement because Sargent & Lundy and Detroit Edison document reviews show that the latest floor response spectra were used in the RHR complex analyses. Therefore, accidents previously evaluated in the UFSAR are not affect by this change.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0022 UFSAR Revision No. 6

Reference Document: LCR 92-054-UFS Section(s) 9A.4

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Fire Protection Program Editorial Changes and Clarifications

SUMMARY:

This evaluation justifies the following changes made to UFSAR Appendix 9A, "Fire Protection Analysis and Review of Appendix A to BTP APCSB 9.9-1".

1. UFSAR Section 9A.4.1.3.3 has been revised to state that the objective of the Fire Protection Program, as it applies to reactor building fire zone 2, is to prevent a fire from affecting redundant shutdown equipment. This change brings the text into closer agreement with Section III.G.2 of Appendix R to 10CFR50. This is an editorial change.

2. UFSAR Figures 9A-2 through 9A-12 have been revised to identify the non-rated fire zone boundaries identified in the text of UFSAR Sections 9A.4.1 and 9A.4.2. These changes bring the figures into better agreement with the UFSAR text. These changes are drawing clarifications.

The above changes have no impact on the Fermi 2 Fire Protection Program. No physical changes have been made to the plant and no maintenance, operations, or testing procedures are affected. These changes do not affect the function or operability of plant systems or components including the fire protection requirements for fire zone barriers.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0029 UFSAR Revision No. N/A

Reference Document: SOE 92-03 Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Addition of Granular Activated Carbon (GAC) to Radioactive Waste Deep Bed Demineralizers

SUMMARY:

Sequence of Events SOE 92-03 added 7 ft^3 of GAC on top of a reduced volume of mixed bed resins in the radwaste deep bed demineralizers. The new demineralizer configuration reduces the resin column volume from 49 ft^3 to 42 ft^3 to accommodate the GAC. The addition of GAC provides a more effective means of treating total organic carbons (TOC) while maintaining primary ion exchange capabilities. This SOE is being performed in such a fashion that accumulated data and test results can be reviewed to determine if this demineralizer configuration will be made permanent.

This modification does not change the requirements of the Effluents Program, chemistry specifications, or the Process Control Program (PCP). This change does not alter the radwaste deep bed demineralizer volume. Ion exchange capabilities are not substantially altered because, in the original bed configuration, resin replacement was driven by TOC removal and not by ion exchange exhaustion. Resin replacement will still be driven by TOC removal. However, resin life will be slightly extended and TOC removal will be more efficient. This change does not affect the radwaste system tank rupture analysis or the supporting source terms. GAC is not expected to be a large contributor to activity sources when compared to the resin activity contribution because GAC does not have the ion exchange capabilities of the resin. GAC is less dense and softer than the demineralizer resins. Therefore, its impact on radwaste system equipment is less severe and there is no change to radwaste handling practices. GAC radwaste processing is acceptable because the PCP allows up to 70% GAC in any liner. The radwaste deep bed demineralizer GAC concentration is less than 20%.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0030 UFSAR Revision No. 6

Reference Document: LCR 92-061 Section(s) 9.4

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Guidance for Taking Essential Room Coolers Out of Service

SUMMARY:

This evaluation justifies the use of a single design calculation document to provide guidance on which essential room coolers may be taken out of service without impacting the reliability of the safety related equipment that they serve. This documentation provides the operability determinations for various cooler configurations to ensure that applicable Technical Specifications are not violated. The configurations were evaluated using either steady state or transient heatup calculation methodologies. The steady state methodology assumed worst case peak environmental profile temperatures in the areas or rooms adjacent to the room being evaluated and that, once a cooler was taken out of service, it was not returned to service. The safety related equipment in the room was evaluated for operability based on the resultant calculated steady state room temperature. The transient heatup methodology assumed that, when room cooling becomes unavailable, the equipment in the room will continue to operate and function for a period of time without cooling. This method calculated how long the equipment can perform its safety related function. In the past, essential room cooler operability determinations were conducted using Engineering Functional Analyses, Deviation Event Reports, or other design calculations. This subject design calculation consolidates these operability determinations into one document.

The results of the subject design calculation document do not impact the bases for the emergency equipment cooling water (EECW) system Technical Specifications which specify actions for one inoperable EECW division. The essential cooler configurations that were evaluated in the subject design document are within the bounds of existing safety analyses. All safety related equipment and systems will perform as described in the UFSAR and Fermi 2 Safety Evaluation Report.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0033 UFSAR Revision No. 6

Reference Document: LCR 92-093-UFS Section(s) 9.5; 9A.6

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Removal of Standby Gas Treatment System (SGTS) Exhaust and Cooling Fans CO_2 Lockout

SUMMARY:

This change removed the SGTS Exhaust and Cooling Fans CO_2 initiation lockout by lifting leads to SGTS panels H21P295A and H21P295B. This prevents the SGTS CO_2 system from automatically disabling the SGTS and ensures that the SGTS will automatically start within the 33 second time frame assumed in the containment pressure response analysis. This change was proposed in an engineering evaluation that assessed the impact of the SGTS CO_2 sustem on the operability of the SGTS.

This modification does not affect any other SGTS interlocks or functions. The SGTS will still automatically initiate as described in the UFSAR and there is no change to the LOCA analysis. The CO_2 system will continue to automatically actuate at 310 °F. Alarm response procedures require the operator to place the SGTS exhaust and cooling fans control switches in the off-reset position if the CO_2 discharge control room annunciator alarms 8D48 or 17D22 are received. This ensures that CO_2 remains in the charcoal bed where fans were running and prevents any fan in standby from starting and sweeping the CO_2 out of its respective charcoal bed. SGTS inoperability due to CO_2 operation is addressed in UFSAR Table 9.5-2, "Failure Mode and Effects Analysis: Inadvertent Operation of Safety-Related Fire Protection System". It states that the SGTS is lost until it is manually restarted. Any delay in shutting down the exhaust and cooling fans has a negligible affect on CO_2 system fire suppression capabilities. The CO_2 system is sized for ten charcoal bed injections. Each injection has a ten minute duration. Therefore, there is an adequate reserve if a delay in manual action causes the initial CO_2 injection to be swept out of the system.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0037   UFSAR Revision No. 6

Reference Document: LCR 92-065-UFS   Section(s) 6.4; 9A.5; A.1.78; A.1.95

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Plant Breathing Air System Changes

SUMMARY:

This evaluation justifies the following changes to the UFSAR:

1. UFSAR Subsection 9A.5.d.4.(h) - The description of the location and the configuration of the plant breathing air system control room outlet connections has been changed from having outlet connections on the control room panels to having a five connection manifold on the south wall of the control room. This change accurately reflects the location and configuration of the plant breathing air system control room outlet connections.

2. UFSAR Subsection 6.4.2.5 - This subsection has been revised to state that breathing air is supplied to a five connection manifold. The description of the breathing air mask/harness set location has been changed from having a mask/harness at each hose to having individual mask/harness sets located at the manifold. These changes accurately reflect the outlet configuration and the mask/harness set location.

3. UFSAR Section A.1.78 - The distance to the nearest transportation line has been changed from 4 miles to 3.5 miles. The new distance is more accurate. In addition, a new sentence has been added that states that, in accordance with Regulatory guides 1.78 and 1.95, control room habitability was analyzed for the rupture of a 90 ton chlorine railroad tank car and the resulting control room chlorine concentrations are well within NRC guideline limits. The new analysis reflects the fact that there are no major onsite sources of chlorine.

4. UFSAR Section A.1.95 - The statement that Fermi 2 complies with Regulatory Guide 1.95, Revision 1, Position C.4.c has been removed. This section now states that Fermi 2 complies with Regulatory Guide 1.95 Positions C.1 and C.2 because there are no onsite sources of chlorine. It states that the nearest source of an accidental chlorine release is 3.5 miles away and the control room chlorine concentration was calculated for such a release using the methodology in Regulatory Guide 1.95 and NUREG 0800. It also states that the operators will be allowed 10 minutes to don their breathing gear due to the low chlorine

Safety Evaluation No. 92-0037 (continued):

concentrations in accordance with NUREG 0800.

5. UFSAR Subsection 6.4.3.4 - The time allotted for the operators to don their breathing gear has been changed from 2 minutes to 10 minutes. This change reflects the low calculated chlorine concentration in the control room.

There are no accidents analyzed in the UFSAR that involve a failure of the breathing air system. The single 5 point manifold does not impact the function of the plant breathing air system. The margin of safety has been significantly increased with the removal of onsite chlorine sources. The chlorine tank car accident is now the nearby potential chlorine source. The NRC requirements state that the control room chlorine concentration shall be no greater than 15 ppm over the first two minutes after detection. The new calculations assume an offsite release and result in a control room chlorine concentration below 4 ppm after 10 minutes. Therefore, the revised time limit for donning breathing devices is acceptable.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0041 UFSAR Revision No. N/A

Reference Document: RERP Plan Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Miscellaneous Changes to RERP Plan Table D-1

SUMMARY:

This evaluation justifies changes made to RERP Plan Table D-1, "Classification of Emergencies". The following changes included:

1. Reactor Control System Failures

   An indication for a site area emergency has been changed from "failure of the standby liquid control systems (SLCS) to bring the reactor subcritical" to "boron injection is required". This more accurately reflects the condition transient requiring operation of reactor shutdown systems.

   A general emergency classification has been added for "a transient requiring operation of shutdown systems with failure to scram with core damage evident". The RERP did not previously address this condition. The general emergency classification is appropriate for this condition because it is an example of a NUREG-0654, Appendix 1 core melt sequence.

2. Degraded Safety Systems and Safety Limits

   The condition for an alert has been changed from "a complete loss of any function essential to achieve plant cold shutdown" to "a complete loss of any function essential to achieve and maintain plant cold shutdown". The condition now recognizes that functions may be lost while in cold shutdown that would prevent being able to maintain cold shutdown.

   The indication for an alert was revised to include "the actions of 20.205.01, Loss of Shutdown Cooling, are not sufficient to achieve or maintain reactor coolant temperature at or below 200 ^{o}F". Prior to this revision, the indication only stated that the limiting condition for operation and action statements of Technical Specification 3.4.9.2 were not met. Now, either indication would cause the condition to be met.

Safety Evaluation No. 92-0041 (continued):

3. Loss of One Fission Product Barrier

   The indication for a site area emergency concerning reactor water level has been changed from "reactor water level low at less than TAF (top of active fuel) as shown by reactor water level indicators" to "unable to increase reactor water level as shown by downward trend on reactor water level indicators". The revised wording is more conservative in that the water level does not have to decrease to the top of active fuel before a site area emergency is declared.

4. High Radiation or Contamination Levels Within Facility

   The phase "in normally occupied areas of the plant" has been removed from the condition and indications for a site area emergency and replaced by the phrase "in unlocked areas of the plant". The original phrase is considered too subjective and could not be adequately defined.

   The phrase "in occupied plant areas" has been removed from the indications in the "significant unexpected increase in radiation or airborne activity levels" condition. This phrase was too subjective and could not be adequately defined.

These changes have no effect on the operation of plant systems or equipment. The implementation of these changes will result in a reduction in the time required to augment the plant staff in an emergency and reduce the consequences of many accidents evaluated in the UFSAR. Emergency classification indications are not a factor in the probability of any accident evaluated in the UFSAR or in the Technical Specification bases.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0045 UFSAR Revision No. 6

Reference Document: LCR 92-069-UFS Section(s) 9.5

Table(s) 9.5-3

Figure Change [ ] Yes [X] No

Title of Change: Removal of UFSAR Normal Lighting Facilities Table

SUMMARY:

The evaluation justifies removing UFSAR Table 9.5-3, "Normal Lighting Facilities", and modifying the description of the normal lighting system in UFSAR Subsection 9.5.3. The UFSAR description includes the statement: "The Normal Lighting System consists of fixtures and facilities placed in areas of the plant to meet the target light intensities identified by the Illuminating Engineering Society (IES) for nuclear powerplants/industrial facilities." This change will allow more flexibility in plant lighting design to utilize industrial lighting advancements that have occurred since the original UFSAR lighting system description was written.

This change does not impact the function of the normal lighting system. It does not impact the design basis nor remove mercury intrusion protection stated in the normal lighting description in UFSAR Subsection 9.5.3.2. The impact of lighting failures on licensing basis accidents is limited to: 1) fixture failure with its potential to initiate an accident or reduce a response method and 2) illumination loss which could reduce operator response capability. Fixture failure is not affected by this change and there is no change to the description of emergency lighting used by operators to perform safety functions following an accident. This change conforms to the content requirements of NUREG 0800, "Standard Review Plan"; Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Powerplants"; and NUREG 0798, "Safety Evaluation Report Related to the Operation of Enrico Fermi Atomic Power Plant, Unit No. 2".

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0047 UFSAR Revision No. 6

Reference Document: 44.000.001 Section(s) 7.2

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Utilizing the Westinghouse Sensor Response Time Testing (WSRTT) Noise Method for Performing Sensor Response Time Testing

SUMMARY:

This evaluation justifies using the WSRTT noise method for performing reactor protection system (RPS), emergency core cooling system (ECCS), and containment and reactor vessel isolation control system (CRVICS) sensor response time testing during normal plant operation at full power. The UFSAR has been revised to allow the use of the WSRTT noise method as alternative technique for testing sensor response time. The WSRTT noise method calculates sensor response time by employing noise analysis techniques. The traditional response time testing method is the hydraulic ramp method. This method injects a simulated signal into the sensor and can only be performed during plant shutdown. This method does not provide complete response time testability because a portion of the instrument impulse line is bypassed. The impulse line time delay is accounted for in Fermi 2 response time testing procedures which utilize the hydraulic ramp method. The advantage of the WSRTT noise method is that it can measure the complete sensor response time; can determine the sensor response time characteristic while the plant is operating; and detect a degraded sensor without removing it, valving it out to inject test signals, or internally perturbing the plant to test sensor response.

The WSRTT noise method has been accepted by the NRC for use at Byron, Unit 1 and Millstone, Unit 3. A comparison of response time results for several testing methods reported in NUREG/CR-5383, "Effect of Aging On Response Time of Nuclear Plant Pressure Sensors", indicates that the WSRTT noise method is equivalent to the hydraulic ramp method. On-line testing of the RPS, ECCS, and CRVICS is acceptable because UFSAR Subsections 7.2.2.2.1, 7.3.1.3.4, and 7.3.2.3.3 state that these systems are testable during reactor operation. A test equipment failure will not cause the master trip unit to malfunction because the sensor is isolated from the test equipment by an isolation amplifier. Procedural administrative controls ensure that only one redundant transmitter will be tested at a time and that divisional separation is maintained during testing to further reduce any possible impact of testing on the plant. These controls also ensure that the plant is in normal steady state conditions close to full power while measurements are taken. The measurements

Safety Evaluation No. 92-0047 (continued):

are voided if a transient occurs and are repeated when steady state operation resumes.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0051 UFSAR Revision No. 6

Reference Document: COLR 4.0 Section(s) 4.1; 4.2; 4.3; 4.4; 5.2 Chapter 15

Table(s) 7.6-10

Figure Change [ ] Yes [X] No

Title of Change: Fuel Cycle 4 Core Operating Limits Report (COLR)

SUMMARY:

This evaluation justifies changes to the COLR for fuel cycle 4. These changes include:

1. Adding the thermal limits for the new fuel types used in fuel cycle 4. 316 GE6 fuel bundles were replaced with 224 GE11 fuel bundles and 92 reinserted GE6 fuel bundles.

2. Changing rod block monitor (RBM) setpoints and removing the RBM time constants.

3. Reflecting Fermi 2 Technical Specification Amendment 87 that allows power operation at 3430 MWt and changes the LOCA methodology to the NRC approved SAFER/GESTR-LOCA application methodology.

The use of GE11 fuel does not change the method of plant operation or result in any modifications to the permanent facility. Fuel bundle damage analysis indicates that less activity is released to the environment by the GE11 bundle design than by the 8x8 fuel bundle designs. Average linear heat generation rate (APHGR), minimum critical power ratio (MCPR), and linear heat generation rate (LGHR) operating limits applicable to the GE11 fuel have been established to ensure that the Technical Specification safety margins are maintained. The standby liquid control system (SLCS) is capable of inserting the maximum required negative reactivity and the minimum cycle 4 shutdown margin (SDM) is greater than the minimum required SDM in the Technical Specifications.

Maximum average linear heat generation (MAPLHGR), LGHR, and operating MCPR limits do not change plant operation. MAPLHGR and LGHR limits do not result in any modifications to the facility. However, the addition of the operating MCPR limits result in RBM setpoint changes. These limits have been calculated using NRC approved methods.

Safety Evaluation No. 92-0051 (continued):

The RBM setpoint changes do not change plant operation or RBM function. The setpoint changes have been accounted for in the cycle 4 rod withdrawal error analysis and are within the bounding operating limit.

Removal of the RBM time constant specifications from COLR does not result in any modification to the facility. This change does not invalidate the rod withdrawal error analysis because the RBM hardware currently enforces this constraint. The RBM hardware values are bounded by those values assumed in the rod withdrawal error analysis.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0060 UFSAR Revision No. N/A

Reference Document: RERP Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Primary Containment Purging Emergency Classification

SUMMARY:

This evaluation justifies adding a general emergency declaration to the RERP Plan for the condition of emergency venting or purging primary containment through the torus hardened vent or to the refueling floor. The torus hardened vent was installed during the third refueling outage and is evaluated in Safety Evaluation 92-0003. Emergency venting or purging is required whenever torus pressure is at the primary containment pressure limit (PCPL) or when primary containment hydrogen concentration reaches 6% with oxygen concentration above 5%. Emergency venting or purging bypasses secondary containment and provides a direct path from the primary containment atmosphere to the environment. If the PCPL or hydrogen and oxygen limits are exceeded, it is likely that other fission product barriers have failed or failure is imminent. Therefore, a general emergency declaration is appropriate for this condition.

This change does not affect any plant or facility system. Implementation of this emergency classification would likely result in a reduction in the time required to provide protective action for the plant staff and the public and reduce the consequences of the accident.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0062 UFSAR Revision No. N/A

Reference Document: 35.000.233 Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Reactor Water Cleanup (RWCU) Bottom Head Drain Line Freeze Seal

SUMMARY:

This evaluation justifies applying a freeze seal to the RWCU bottom head drain line which facilitated local leak rate testing (LLRT) of the RWCU inboard isolation valve G3352F001 and repair of RWCU bottom head drain valve G3352F101 during the third refueling outage. G3352F101 is the normal LLRT boundary valve between G3352F001 and the reactor vessel. However, G3352F101 did not fully isolate and required repair. The freeze seal was installed on the 2 1/2" line between the reactor pressure vessel and G3352F101 located inside the reactor pedestal. The exact location was based on accessibility and on Nuclear Production Procedure NPP 35.000.233, "Installation of Freeze Plugs". The freeze seal was installed when all fuel assemblies were out of the reactor vessel. Non-destructive examinations were performed before freezing to ensure that the seal was not located at or near any discontinuities and after freezing to ensure that no defects resulted from freezing. A temperature of -300 ^{O}F was maintained to avoid freeze plug slippage. The area around the freeze plug was roped off to avoid mechanical impact to the piping where the freeze seal was installed and its possible brittle fracture failure.

A drain line break during power operation is evaluated in UFSAR Section 3.6. A failure of this line with the core off-loaded is bounded by this existing evaluation. With the core off-loaded, the only consequence of a pipe break is the flooding of the drywell and torus. The break flow out of the 2 1/2" line would be made up from the residual heat removal system or the core spray system. Therefore, the fuel pool level can be maintained. The Battelle report, "Developement of Guidelines for Use of Ice Plugs in Pipeline Maintenance and Hydrostatic Testing", dated November 15, 1982, shows that the installation and removal of the freeze seal causes no permanent changes in physical properties such as strength, toughness, or microstructure. NPP 35.000.233 ensures that the ice plug is completely melted prior to returning the pipe to service to avoid component damage that could result from a partially melted plug. Water chemistry is maintained by the fuel pool cleanup system when the RWCU system is out of service and the freeze seal is installed.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0068 UFSAR Revision No. 6

Reference Document: LCR 92-121-UFS Section(s) 12.3; 13.2

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: General Employee Training (GET) Requalification Interval Extension

SUMMARY:

This evaluation justifies adding a 25% extension provision to the twelve month GET requalification interval requirements in UFSAR Subsections 13.2.4.2.1 and 12.3.3. This provision also includes a requirement that the requalification dates do not exceed 3.25 times the interval over three requalification intervals as a safeguard to prevent the possibility of skipping a requalification interval. The extension provision was added to maintain the qualification and the site access of rotating shift personnel whose availability for GET training occasionally falls one month beyond the twelve month interval.

The training interval extension only affects the basic training requirements for plant access. The subject training does not involve training or qualifications required for operation or maintenance of plant equipment and has no effect on any other qualification requirements.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0072 UFSAR Revision No. N/A

Reference Document: NPP 23.623 Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Verification of Refueling Interlocks with Defective Position Indicator Probes (PIP)

SUMMARY:

This evaluation justifies allowing verification of Technical Specification 3.9.1 required refueling interlocks with defective PIPs. Nuclear Production Procedure NPP 23.623, "Reactor Manual Control System", has been revised to allow control rod position to be administratively verified and controlled and allows defective PIP inputs to be manually jumpered to the reactor manual control system (RCMS).

When the reactor was in the refueling mode during the third refueling outage, the control rod PIPs were not totally functional because they were sending erroneous position signals to the RCMS. This prevented proper evaluation by the RCMS of the state of the core reactivity controls and the "refuel one rod out" interlock. This interlock and the PIPs are the means by which the "all rods in" condition is determined for the refueling interlock system. This procedure change allows the erroneous signals to be removed and enforce administrative controls to ensure that the function of the "all rods in" input to the refueling interlocks cannot be defeated. The following administrative controls were incorporated:

- o The affected control rod is visually verified to be fully inserted and both hydraulically and electrically disarmed. An abnormal lineup sheet (ALS) controls the status of the affected control rod.

- o The control rods that do not have operable PIPs are declared inoperable until their PIPs are repaired.

- o All control rods are fully inserted. Insertion is verified by either visual verification for control rods with inoperable PIPs or by "full in" indication for control rods with operable PIPs.

- o Rod insertion verification is performed at the conclusion of the refueling interlocks surveillance and a rod block is inserted to preserve adequate shutdown margin during fuel reload.

Safety Evaluation No. 92-0072 (continued):

- Proper operation of the refueling interlocks is verified prior to core alterations by Surveillance Procedure 24.623, "Reactor Manual Control/ Reactor Mode Switch/ Refueling Platform - Refueling Interlocks".

- The installation of jumpers to remove the erroneous PIP inputs to the RMCS will be documented in a procedure controlled by the Nuclear Shift Supervisor.

- Compliance with Technical Specification Surveillance Requirement 4.9.3, "Control Rod Position", includes verification of operable PIPs and verification of the ALS for control rods without operable PIPs. Verifications will be performed at the surveillance frequency specified in Technical Specification Surveillance Requirement 4.9.3.

This procedure revision does not change the function of the refueling interlock system because the visual verification method is acceptable for initial pre-reload checks and compliance with Refueling Technical Specification Surveillance Requirement 4.9.3, "Control Rod Position". There is no change to reactivity controls provided by the "one rod out" interlock. This change does not degrade the measures used to prevent fuel drop accidents and cannot impact fuel handling or secondary containment functions.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0074 UFSAR Revision No. 6

Reference Document: LCR 92-103-UFS Section(s) 13.2; 17.2

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Training Program Description Revisions

SUMMARY:

This evaluation justifies revising UFSAR Section 13.2 training program descriptions to reflect current practice. The changes include:

1. Deleting the outdated description of the licensed operator training program and stating that this program was accredited by the Institute for Nuclear Power Operations (INPO) in accordance with 10CFR55 on December 18, 1985.

2. Adding the ability to use plant and maintenance procedures in the instrument and control and maintenance personnel training program descriptions.

3. Updating the descriptions of the non-licensed operator and the radiation protection technician training programs.

4. Adding a new subsection that describes the technical staff personnel training program.

5. Changing the references from "Assistant Vice President and Manager - Nuclear Production" to "Vice President - Nuclear Operations" to reflect the current organizational title.

These changes enhance the current training program descriptions and allow timely program improvements. They have no impact on structures, systems, or components that perform a safety function. The licensed operator training program continues to meet or exceed the requirements of 10CFR55 and is enhanced by its adherence to INPO ACAD 91-012, "Guidelines for Training Qualification of Licensed Operators".

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0082 UFSAR Revision No. 6

Reference Document: LCR 92-114-UFS Section(s) 8.3

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Changes to the UFSAR Description of the Emergency Diesel Generator (EDG) Voltage Regulation Controls

SUMMARY:

This evaluation justified removing the description of the EDG manual voltage regulator (MVR) from the UFSAR and clarifying that the automatic voltage regulator setpoint is adjustable from both the local panel and the main control room. An operability determination performed as part of Deviation Event Report DER 92-0173 determined that the EDGs are operable without the MRV since it is not used in any safety related EDG operations. The DER investigation also showed that the MVR would not be capable of serving as a backup to the automatic voltage regulator (AVR) in an emergency situation since constant manual operator action would be required to follow the varying automatic loads. As a result of the DER investigation, Potential Design Change PDC 13297 was initiated to abandon the MVR. The description of the local and control room AVR setpoint adjustment capability in the UFSAR reflects as-built plant design.

This revision does not affect the EDG automatic start and loading sequence; the EDG voltage response for ECCS motors; or the UFSAR Chapter 15 accident analyses involving the loss of offsite power and subsequent reactor shutdown using the EDGs. No credit has been taken for using the MVR as a backup to the AVR or for the use of the MRV in any accident scenario. This revision has no impact on the onsite power system as evaluated in the Fermi 2 NRC safety evaluation report (NUREG 0798) or on the EDG design criteria in NRC Regulatory Guide 1.9, "Selection, Design, and Qualification of Diesel Generator Units Used as Standby (Onsite) Electrical Power Systems at Nuclear Power Plants".

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0093

UFSAR Revision No. N/A

Reference Document: UFSAR

Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Quality Assurance Management Qualifications Evaluation

SUMMARY:

This evaluation justifies selecting an individual for Supervisor - Production Quality Assurance that meets all the requirements specified in the UFSAR with the exception of the one year experience required in quality assurance program implementation. This individual has 46 weeks experience in a quality assurance function gained through his previous assignment as Director - Plant Safety. As the Director - Plant Safety, this individual was responsible for the evaluation of plant corrective action; normally a quality assurance function. This individual has 24 years experience with 15 years related to nuclear power plant operation. The nuclear experience includes positions in equipment performance engineering, outage management, corrective maintenance scheduling and planning, maintenance engineering, and plant safety. This experience more than compensates for the 6 weeks experience that the individual lacks in quality assurance program implementation.

The UFSAR requirements for the Supervisor - Production Quality Assurance are based on Detroit Edison criteria only. The production quality assurance qualifications do not affect plant operations, maintenance, or design. The limits defined in the Fermi 2 Technical Specifications bases are not impacted by this evaluation.

The following Technical Specification Amendments were incorporated into Revision 6 of the UFSAR. The NRC safety evaluation (which is based on the Detroit Edison evaluation supporting the change) that accompanies each amendment provides the basis and justification for the UFSAR revision.

T. S. Amendment	Description	UFSAR Section/Table
82	Removal of Radiological Effluent Technical Specifications (RETS)	11.4 12.2
83	3.0.4 exception (Generic Letter 87-09)	9A.6
87	Authorization to increase power from 3293 Mwt to 3430 Mwt	1.1 1.2 Table 1.6-3 3.6 3.9 Table 3.9-4 4.4 4.5 Table 4.5-9 5.1 5.2 Table 5.2-5 6.2 6.3 7.4 7.6 Table 7.6-9 8.1 8.2 9.1 Table 9.1-1 Table 9.1-2 Table 9.1-3 Table 9.1-4 Table 9.3-1 Table 9.3-2 Table 9.3-3 10.1 Table 10.1-1 10.2 10.4 Table 10.4-5 Table 10.4-6 Table 10.4-7 Table 10.4-8 Table 10.4-9 11.1 11.2

Technical Specification Amendments (continued):

T. S. Amendment	Description	UFSAR Section/Table
87	Authorization to increase power from 3293 Mwt to 3430 Mwt	11.3 11.5 App. 11A 12.1 12.2 15

END OF SAFETY EVALUATION REPORT

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UPDATED FINAL SAFETY ANALYSIS REPORT

Detroit Edison

ARMS-INFORMATION SYSTEMS
DTC TDFSAR DSN UFSAR
PAGE REV 6 PIS
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PIS #935 RECIPIENT
APPROVAL REQ'D YES NO
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