William S. Orser
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Nuclear Operations

March 18, 1992
NRC-92-0021

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Reference: Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

Subject: ~~Submittal of Revision 5 to the Fermi 2 Updated Final Safety Analysis Report and the~~ Annual 10CFR50.59 Safety Evaluation Summary Report

~~Pursuant to 10CFR50.71(e) and 10CFR50.59(b)(2), Detroit Edison hereby submits Revision 5 to the Updated Final Safety Analysis Report (UFSAR) for Fermi 2 and the annual Safety Evaluation Summary Report.~~

~~The signed original and ten additional copies of the UFSAR, Revision 5, are enclosed; one copy will be submitted to Region III and one copy to the NRC Resident Inspector. The UFSAR contains changes made since submittal of Revision 4 in March 1991 and as a minimum describes the plant configuration through September 20, 1991, six months prior to this submittal. Changes associated with Revision 5 are annotated by revision bars in the appropriate margin marked with a "5". All revised pages are marked Rev 5 3/92.~~

Also enclosed is the annual Safety Evaluation Summary Report containing a brief description of changes to plant design, procedures, tests, experiments, and the UFSAR.

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If you hav- any questions, please contact Evelyn F. Madsen at (313) 586-4205.

Sincerely,

Enclosure

cc: T. G. Colburn
A. B. Davis
R. W. DeFayette (one copy of UFSAR)
S. Stasek (one copy of UFSAR)

I, W. S. Orser, do hereby affirm that the foregoing statements are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

W. S. Orser
Senior Vice President

On this 18th day of March, 1992, before me personally appe ed W. S. Orser, being first duly sworn and says that he executed .he foregoing as his free act and deed.

Rosalie A. Armetta
Notary Public

ROSALIE A ARMETTA
NOTARY PUBLIC STATE OF MICHIGAN
MONROE COUNTY
MY COMMISSION EXP. NOV. 20,1995

U.S. NUCLEAR REGULATORY COMMISSION
DOCKET NO. 50-341
LICENSE NO. NPF-43
ENCLOSURE TO NRC-92-0021

Fermi 2

SAFETY EVALUATION SUMMARY REPORT 1991

Detroit EDISON

Enclosure to
NRC-92-0021

FERMI 2

SAFETY EVALUATION SUMMARY REPORT

1991

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

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SAFETY EVALUATION SUMMARY REPORT

1991

AS-BUILT NOTICES

SAFETY EVALUATION SUMMARY

Safety Evaluation No:	88-0052	UFSAR Revision No.	5
Reference Document:	ABN 8660-1	Section(s)	6.4; 9.4
		Table(s)	N/A
		Figure Change	[X] Yes [ ] No

Title of Change: Control Center Heating, Ventilation, and Air Conditioning (CCHVAC) System Setpoints

SUMMARY:

This as-built notice revised the master instrument list and CCHVAC mechanical system diagrams to reflect the as-built control setpoint for the normal and recirculation modes of the CCHVAC system (+ 1/4" W.C.). This control setpoint for both modes of CCHVAC is consistent with the control center pressure requirement (+ 1/4" ± 1/8" W.C.) provided in the UFSAR.

This as-built notice revised the master instrument list and CCHVAC drawings to reflect the as-built control setpoints and control pressures identified in the UFSAR. Consequently, this is a documentation change only. The context of this notice does not impact the operation of CCHVAC in the recirculation mode.

SAFETY EVALUATION SUMMARY

Safety Evaluation No:	88-0232	UFSAR Revision No.	5
Reference Document:	ABN 9877-1	Section(s)	N/A
		Table(s)	N/A

Figure Change [X] Yes [ ] No

Title of Change: Breathing Air Radiation Monitor System Changes

SUMMARY:

Various sections of the UFSAR describe the breathing air radiation monitor system. This system would have been used if station air was utilized for breathing air purposes. Since there is no intention to use station air for breathing air, the breathing air radiation monitors will not be installed and made operational.

These radiation monitors are not subject to any equipment malfunctions considered in the UFSAR. Further, there is no process or electrical link between these monitors and other plant equipment. Regulatory Guide 8.15 for monitoring station air radioactivity has not been violated as station air will not be used as breathing air.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0162	UFSAR Revision No. 5
Reference Document: ABN 12059-1	Section(s) 9A.4; 9A.5
	Table(s) N/A
	Figure Change [X] Yes [ ] No

Title of Change: Reactor, Auxiliary, and Turbine Building Wall Fire Barrier Upgrade

SUMMARY:

This evaluation justifies upgrading the following exterior plant walls to rated fire barriers:

1. The north, south, and west exterior walls of the reactor building below the metal siding at elevation 684'-6".

2. The north and south exterior walls of the auxiliary building including the cable vault roof located on the south wall.

3. The west exterior wall of the turbine building below the metal siding at elevation 679'-6".

These walls are being fire rated to ensure that trailers and other combustible materials that are located in close proximity to the plant during refueling outages and for other possible reasons do not present a fire hazard to safety related equipment and circuits within the plant.

The above walls are at least 18" thick. Underwriters Laboratories, the National Fire Protection Association, and the American Concrete Institute confirm that a reinforced concrete wall with a minimum thickness of 8" is a 3-hour rated fire barrier. All penetrations and openings in the above walls are either 3-hour fire rated penetration seals or are technically justified as adequate fire stops in accordance with the guidance in NRC Generic Letter 86-10. The original unlabeled cable vault security door has been replaced with a 3-hour rated door. This door continues to meet security plan requirements.

The previously approved Appendix R analysis assumed that combustibles are maintained at an acceptable distance from safety related buildings and that no fire in the yard area would cause damage or enter any of the fire zones facing the unrated exterior walls. Providing a rated fire barrier in an exterior wall is an acceptable alternative suggested by NRC Generic Letter 86-10 (Reference Questions and Answers 3.1.3.) and, therefore, assures that this assumption is still valid.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0054 UFSAR Revision No. 5

Reference Document: ABN 11273-1 Section(s) N/A

Table(s) 6.2-2; 7.5-5; 9A.6.1-1

Figure Change [X] Yes [ ] No

Title of Change: Emergency Equipment Cooling Water (EECW) Drywell Supply and Return Valve Renumbering

SUMMARY:

This as-built notice documents the interchange of valve number designations for Division I EECW drywell supply isolation valve P4400F606A and Division I EECW drywell return isolation valve P4400F607A. This change ensures that the valve numbering scheme for both the Division I and Division II valves is consistent. License Amendment 70 incorporating the valve number change was approved by the NRC.

This change does not make any hardware changes. The evaluation of the hardware changes associated with the valve renumbering is documented in safety evaluation 90-0134 as part of engineering design package EDP 11273. The as-built documentation of this change will improve the man/machine interface by standardizing the numbering designations of the valves between Divisions I and II.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0055 UFSAR Revision No. 5

Reference Document: ABN 12531-1 Section(s) N/A

Table(s) 3.2-1

Figure Change [X] Yes [ ] No

Title of Change: Emergency Diesel Generator (EDG) Standby Fuel Oil Pump Discharge Line Material Code Change

SUMMARY:

This safety evaluation justifies changing the specification code for each EDG standby fuel oil pump discharge line from ASME III to the Diesel Engine Manufacturers Association (DEMA) code. The SB-75 copper tubing and SB-164 fittings specified under the ASME III code are no longer available. The diesel manufacturer recommends S-75 copper tubing and ASTM A276 type 316 fittings specified by the DEMA code as equivalent replacement parts. Reclassification of the parts from ASME III to DEMA allows the use of market available copper tubing and fittings.

This specification change does not change the operation or function of the EDGs. The new parts will transfer fuel and maintain the pressure boundary in the same manner as the original parts. The discharge line is still a QA I, seismic category I installation meeting all the regulatory requirements for the EDGs.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0056 UFSAR Revision No. 5

Reference Document: ABN 8264-1 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Documentation of the Permanent Installation of the Vibration Velocity Transducers for the Reactor Recirculation Pumps

SUMMARY:

ABN 8264-1 documents the permanent as-built configuration of the Bentley-Nevada velocity transducers originally installed by temporary modification 85-057. Three velocity transducers are installed on the frames of each pump. The transducers are used as a backup to the Robertshaw vibra-switch alarm switches. If a spurious alarm develops and the drywell is inaccessible to allow raising the setpoint, periodic readings of the velocity transducers are made to ensure the velocities are less than 0.4 in./sec.

This modification does not change the function of the recirculation pump vibration and monitoring system. The control circuits for this system are independent of the recirculation pump motor controls. As a result, this modification does not affect the operation of the recirculation pumps. A malfunction of the vibration monitoring and surveillance system does not contribute to the malfunction of other safety related equipment.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0060 UFSAR Revision No. 5

Reference Document: ABN 12319-1 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Reactor Core Isolation Cooling (RCIC) System Diagram Revision

SUMMARY:

This evaluation justifies revising the RCIC system UFSAR figure to include RCIC vibration monitoring equipment installed during pre-operational and startup testing of the RCIC turbine and pump. The equipment is not currently identified on any plant base configuration design documents. This equipment consists of:

1. Ten vibration transducers mounted on the RCIC turbine and pump bearing housings.

2 One terminal box.

3. Ten interconnecting cables routed from the vibration transducers to the terminal box.

This equipment has been left in place due to poor accessibility. It will remain permanently installed and is utilized for intermittent vibration monitoring in conjunction with portable vibration testing equipment provided by test personnel.

The vibration equipment is non-Q, its installation has been field verified, and it conforms to seismic II/I criteria. This equipment does not affect the operation, function, or performance of any plant equipment. This modification does not introduce any new failure modes or scenarios that affect the reactor coolant or containment boundaries.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0081 UFSAR Revision No. 5

Reference Document: ABN 12430-1 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

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

SUMMARY:

This modification changed the #2 (tripped feedwater pump) and #3 (loss of heater drains) RRP speed limiter setpoints from 42% and 48%, respectively, to 37%. This was done to ensure that plant operation in the maximum extended operating domain (MEOD) does not place the plant in the thermal hydraulic instability region on the power/flow map or result in a scram following the loss of a single feedwater pump or heater drains pump. The 37% setpoint corresponds to 48% core flow. This provides a 3% core flow margin to the instability region. This setpoint also allows the condensate/feedwater flow to match the capacity of the polishing demineralizers.

A transient analysis was performed to determine the feedwater capacity available following a single feedwater pump trip concurrent with the loss of heater drains and the subsequent RRP runback. The results of this assessment showed that there is a possibility of a reactor water low level scram following the runback from the upper end of the maximum extended load line limit (MELLL) region (75% core flow at 100% power). To avoid a low water level scram, the plant operating procedures require the operators to start the standby feedwater system (SBFW) when the final reactor power is greater than 70%. The use of the SBFW system will help to maintain reactor water level allowing a controlled rod insertion to reduce power to within the licensing domain.

These changes enhance the plant's capability to operate outside of the thermal hydraulic instability region and to avoid a reactor scram. The recirculation system runback with the subsequent SBFW actuation is not an event analyzed in the UFSAR. However, it is bounded by the analyses for loss of feedwater flow, high pressure coolant injection, and the recirculation flow control failure transients. The function of the recirculation speed control limits is not altered. The use of the SBFW system to mitigate the possibility of a reactor low water level scram is in accordance with the design objectives in the UFSAR and approved plant procedures.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0096 UFSAR Revision No. 5

Reference Document: ABN 12648-1 Section(s) 9.2; A.1.27

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Revising the Residual (RHR) Heat Removal System Reservoir Volume in the UFSAR

SUMMARY:

This evaluation justified revising the UFSAR to correct the stated RHR reservoir volumes. A revised design calculation was performed to determine the RHR reservoir volume between elevations 569 ft and 590 ft and reflects the as-built volume of the RHR reservoir. This revised calculation used more accurate dimensions from architectural and civil drawings. The applicable results have been incorporated into the UFSAR to accurately describe the RHR reservoir volume for a given elevation. In addition, the UFSAR has been revised to reflect the correct remaining RHR reservoir volume after the 30 day LOCA supply is used. The figures in the design calculation were used to calculate this margin.

This revision does not change the design criteria, function, or operation of the RHR system. The revision to the design calculation is based on previously approved documents. The calculated RHR reservoir volumes still meet the minimum RHR reservoir 30-day water supply required by the Technical Specification 3.7.1.5.

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.: 88-0121 Figure Change
Implementation Document: ABN 6815-1

Safety Evaluation No.: 91-0002 Figure Change
Implementation Document: ABN 11662-1

Safety Evaluation No.: 91-0008 Figure Change
Implementation Document: ABN 11982-1

Safety Evaluation No.: 91-0022 Figure Change
Implementation Document: ABN 11560-1

Safety Evaluation No.: 91-0028 Figure Change
Implementation Document: ABN 11854-1

Safety Evaluation No.: 91-0036 Figure Change
Implementation Document: ABN 12163-1

Safety Evaluation No.: 91-0049 Figure Change
Implementation Document: ABN 11645-1

Safety Evaluation No.: 91-0068 Figure Change
Implementation Document: ABN 12345-1

Safety Evaluation No.: 91-0071 Figure Change
Implementation Document: ABN 12186-1

Safety Evaluation No.: 91-0126 Figure Change
Implementation Document: ABN 12784-1

END OF ABN SECTION

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POTENTIAL DESIGN CHANGES

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 88-0119 UFSAR Revision No. 5

Reference Document: PDC 9090 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: North and South Reactor Feed Pump Drip Drain Isolation Valves Removal and Piping Reroute

SUMMARY:

This modification removed the north and south reactor feed pump drip isolation valves and rerouted the south reactor feed pump drip drain piping. The isolation valves were removed because the possibility exists that if the valves were closed, the drip cavity would fill with seal water. It is then possible for the water to enter the bearing oil area and contaminate the lubricating oil. The associated drip drain piping of the south reactor feed pump was rerouted from an equipment drain to the floor drain system. This was done because the equipment drain system is not equipped to handle potentially oily water. The floor drain system utilizes an oil/water separator before the water is processed in the radwaste system. The design conditions for the drip drain piping were revised from 950 psig @ 430°F to 14.7 psia @ 212°F since the piping and the pump bearing hub are open to the atmosphere.

This modification insures the reliable operation of the reactor feed pumps by decreasing the probability of water contaminating the lubricating oil. Rerouting of the drip drain piping decreases the potential of oil entering the equipment drain system, thus improving the operation of the radwaste system. The subject piping and valves are not covered by the technical specifications. Therefore, there is no impact on the technical specifications.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0151 UFSAR Revision No. 5

Reference Document: PDC 11455 Section(s) 4.1; 4.2; 4.5

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Control Rod Replacement

SUMMARY:

165 of the original 185 control rods were replaced during the second refueling outage. 96 control rods installed in the non-control cell positions are General Electric Duralife 140-C matched-worth rods and the remaining 69 replacement control rods installed in the control cell positions are General Electric Duralife 215-C matched-worth rods.

The Duralife 140-C control rods have the following changes:

1. The top handle, wing sheaths, tie rod, and bottom coupling segment are fusion welded.

2. Thicker wing sheaths are used and the internal stiffening strips have been eliminated.

3. The number of B_4C absorber tubes has been increased from 76 to 84 and the tubes have been shortened from 143" to 137".

4. A 6" halfnium plate is inserted in the top of each wing.

5. An extra row of cooling holes has been added at the top and bottom of each wing.

6. Low cobalt materials are used for the pins and rollers.

The Duralife 140-C control rods are approximately 10% heavier than the original control rods due to the higher density of the halfnium.

Safety Evaluation No. 90-0151 (continued):

The Duralife 215-C control rods have the same changes as the Duralife 140-C control rods with the following exceptions:

1. The total number of B_4C tubes is reduced from 76 to 72. The inner diameter of the tubes has been increased to accommodate more B_4C per tube.

2. A halfnium metal edge strip runs the full 143" absorber length and halfnium plates are installed in the upper 6" of each control rod wing.

3. The bottom control rod drive (CRD) coupling segment and integral velocity limiter have been redesigned to reduce overall weight such that this model control rod is essentially the same weight as the original model control rod.

The improved construction materials enhance control rod integrity and the halfnium extends service life.

The replacement rods are dimensionally compatible with the control rod drives and the reactor core configuration such that there is no significant impact on control rod scram times. The channel bowing tolerance, seismic loading limits, mechanical limits, and material compatibility of the new control rods are equal to or better than the original control rods. The replacement control rods are matched-worth and bounded by the existing lattice physics analysis and the accident and transient analyses. The minimum critical power ratio (MCPR) remains greater than the safety limit MCPR in the applicable analyses. The use of fusion welds as opposed to the overlay and spot welded assembly of the original control rods, improves the control rod mechanical integrity and forms a crevice free structure which eliminates the potential for crevice corrosion damage. NRC approved License Amendment 66 allows the use of halfnium as an absorber. The use of low cobalt materials in the control rod pins and rollers eliminates additional cobalt from entering the reactor assembly. The additional weight of the Duralife 140-C control rods causes a small increase in control rod scram times. However, the weight is within the design margin for the control rod drives and the overall negative reactivity insertion rate with the replacement control rods is within the accident analyses and technical specification limits. Halfnium hydriding incidents reported in NRC Information Notice No. 89-31 are not a concern at Fermi 2 because the replacement control rods use bare halfnium exposed to reactor coolant flow. This prevents hydrogen diffusion from occurring because hydrogen cannot concentrate and the protective halfnium surface oxide layer is maintained.

END OF PDC SECTION

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ENGINEERING DESIGN PACKAGES

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 87-0009 UFSAR Revision No. N/A

Reference Document: EDP 4271 Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Installation of a Maintenance Power Distribution System

SUMMARY:

This modification installed a dedicated balance of plant (BOP) 480VAC maintenance power distribution system in the reactor building. The system consists of two main distribution panels feeding locally mounted disconnect receptacles located throughout the reactor building.

The maintenance power distribution system is dedicated to supplying power to maintenance loads and is powered from a balance of plant power source that originates at Fermi 1. This power system is not associated with any plant process system (safety related or BOP). The additional first floor reactor building fire loading contributed by the cable trays installed by this modification has already been accounted for in the fire hazards analysis.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 87-0063 REV 1 UFSAR Revision No. 5

Reference Document: EDP 6641 Section(s) 1.2; 3.1; 7.1; 7.6; 7.7

Table(s) 1.6-1; 7.7-1

Figure Change [X] Yes [ ] No

Title of Change: Rod Worth Minimizer System Replacement

SUMMARY:

This modification replaced the original rod worth minimizer (RWM) system with a stand alone microcomputer-based NUMAC RWM system. This changeout constitutes corrective actions to satisfy an NRC commitment to prevent recurrence of control rod manipulation errors and is part of the Reactor Operations Improvement Program.

The new RWM design does not alter the purpose or function of the original RWM system. The reliability of the NUMAC RWM is enhanced by its self-test capabilities and availability of spare parts. The electroluminescent display provides the operator with a visual control interface that enhances human factors considerations. Like the previous RWM system, failure of the new RWM system results in an automatic loss of rod movement permissives. The consequences of the RWM failure are bounded by the control rod drop accident analyzed in the UFSAR.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 87-0127 REV 1   UFSAR Revision No. 5

Reference Document: EDP 2189   Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Reactor Building Exhaust Plenum Radiation Monitor D11-P280 Sample Line Modifications

SUMMARY:

This modification sloped the sample lines down from the higher elevations; heat traced the sample lines and sampler SA-13; added a condensate collection bottle; and added bypass valving for additional sample points. This modification was implemented to prevent condensation from forming within the sample lines and possibly damaging radiation monitor D11-P280. This modification also allows maintenance and troubleshooting to be performed on the monitor without disconnecting the sample lines and allows temporary grab samples to be taken when the radiation monitor is inoperable.

This change is a hardware and system enhancement. All hardware installed by this modification conforms to seismic II/I criteria. This change has no effect on any analyzed accidents in UFSAR chapter 15. There is no change to offsite or onsite radiation doses and there is no change to radioactive material releases.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 89-0065 UFSAR Revision No. 5

Reference Document: EDP 9734 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Fuel Channel Hoist Modification

SUMMARY:

The channel hoisting device located over the new fuel inspection stand was modified to facilitate channel handling activities. The installed channel hoisting device was a temporary installation which was used during the initial fuel channeling activities. This scheme was based on a GE proposal. The winch and motor assembly were located on the parapet wall and the rigging cable routed up to the roof level and across just above the new fuel inspection stand. This modification retains the same configuration. Therefore, the motor/winch assembly and their support are the same. This modification improved the structural components at the roof level with a new beam and the selection of proper rigging hardware.

This hoist cannot lift heavy loads and cannot move loads over the spent fuel pool or open reactor vessel. The hoist's movement is in the vertical or near vertical direction. Its usage is limited to new channel installation and maintenance activities. This modification was designed to perform safely for the rated load capacity of 500-lbs. The magnitude of the load is small compared to the loads evaluated in "Load Drop Analysis of Heavy Loads". Therefore, a load drop from this hoist will not damage the refueling floor slab structure. If the channel is dropped on the new fuel bundle, it may damage the fuel but it will not have any radiological consequences. This is not a new scenario since the new fuel channeling activity has not changed. A variety of events that qualify as fuel-handling accidents have been investigated. The most severe accident is dropping a spent fuel bundle into the reactor core. This accident produces the largest number of failed fuel rods. Therefore, the use of the hoisting device will not create the possibility of an accident of a different type than any previously evaluated in the UFSAR.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 89-0068 UFSAR Revision No. 5

Reference Document: EDP 9134 Section(s) 12.2

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: "Machine Shop Pressure High" Alarm Window, Recorder Point, and Local Horn Alarm Removal

SUMMARY:

This modification removed the machine shop high pressure control center alarm window 8D20, the associated sequential recorder point 001X41, and the associated local horn alarm. Originally, the machine shop was to function as a hot shop. This required that the machine shop be maintained at a negative pressure with respect to the Office Service Building (OSB) to prevent the spread of contamination and the exfiltration of contaminated air. Alarm window 8D20 was provided to inform the operators when the pressure differential between the OSB and the machine shop exceeded 1/8" W.C. Since the machine shop is not used as a hot shop, the spread of contamination and the exfiltration of contaminated air is no longer a concern. Therefore, the alarm window, sequential recorder point, and local alarm horn are not required.

The equipment removed by this modification is not required because the machine shop, by procedure, cannot be used as a hot shop unless special radiological controls have been established and approved by Radiation Protection. The machine shop has an exposure level of less than 0.5mr/hr and it is monitored for radiation release through an area radiation monitor. The function of the HVAC remains the same.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 89-0076 REV 1 UFSAR Revision No. 5

Reference Document: EDP 6740 Section(s) 3.10

Table(s) 3.10-3

Figure Change [X] Yes [ ] No

Title of Change: Reactor Pressure and Level Instrumentation Rack Replacement

SUMMARY:

This modification replaced two existing reactor pressure and level instrument racks with a completely redesigned version of each rack. This method of replacing an existing rack with a completely new rack was selected primarily to meet schedule requirements and Technical Specification operability restraints. The time period that the individual or collective set of instruments installed on the racks can be out-of-service is limited to the period of time when irradiated fuel is not handled in the secondary containment; there are no core alterations in progress; and there are no operations in progress with a potential for draining the reactor vessel.

Installation of the instruments and racks will not result in the increase in offsite or onsite radiological releases since the racks and instruments are designed to meet or exceed all applicable Edison and NRC Regulatory requirements. The enhanced human factors characteristics of the rack design result in a decrease in the probability of a malfunction due to surveillance testing. In addition, the single failure capability of the protective system is retained by this change.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 89-0093 UFSAR Revision No. 5

Reference Document: EDP 4274 Section(s) 3.12; 8.3

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Recirculation Motor Generator (MG) Set Control Panel Modifications

SUMMARY:

Both Division 1 and Division 2 control cables enter the MG set control panels. These cables are part of the recirculation pump trip (RPT) portion of the anticipated transient without scram (ATWS) system. This change provided additional separation between opposite division cables and wiring inside the control panels. This was achieved by the addition of physical barriers surrounding the cable of one of the divisions in each panel and providing a fire retardant barrier between the redundant trip coils (TC) on the generator field breaker to which the divisional cables are routed.

The addition of fire retardant barriers internal to the MG set control panels enhances the ability of the systems to perform their intended safety function. These barriers provide adequate protection for at least one of the QA-1 divisional trip signal circuits for any of the following hazards:

1. Short circuit of QA-1 circuit
2. Short circuit of a balance of plant circuit
3. Gross failure of one trip coil
4. Crimping of wire
5. Fuse failure
6. Small fire internal to panel

The only adverse affect would be an insignificant addition to the combustible loading in the event of a major fire. However, these barriers are not intended to protect against a major fire nor is the equipment required for the Appendix R Fire Scenario.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 89-0100 UFSAR Revision No. 5

Reference Document: EDP 9828 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Iso-Mimic Panel Modifications

SUMMARY:

This modification removed the dynamic portion of the iso-mimic panel and replaced it with a status indication display showing the status of the main steam isolation valve (MSIV) trip logic circuits. The new display shows the status of the MSIV isolation relays plus seven variables in each trip system that can cause a MSIV isolation for each of the four channels. The variables being displayed for each channel are: main steam line low pressure; reactor water low level 1; reactor building and steam tunnel high temperature; main steam line high flow; main steam line high radiation; condenser low vacuum; and turbine building high temperature. This information will help prevent operator errors in assessing MSIV trip logic status.

This modification affects control room annunciation. It does not affect automatic or manual operation of the MSIV trip logic circuits. There is no change to the onsite or offsite radiation doses or radioactive material releases. This modification installed an operator aid which functions in a manner similar to the existing annunciation system. This modification does not affect the operation of any safety-related equipment. All new conduits installed by this modification are seismically mounted in accordance with Fermi 2 standard specifications. All new balance-of-plant (BOP) cables are routed in BOP cable trays and conduits until they reach safety-related cabinets. In safety-related cabinets, BOP cable and wiring is separated from cabinet internal wiring carrying safety-related signals as much as possible. All new inputs are taken from dry contacts on existing relays. The status of each contact is sensed with a low energy v[illegible] source which cannot degrade the function of the safety-related circuits on the relay coils.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 89-0116 REV 1 UFSAR Revision No. 5

Reference Document: EDP 3303 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Post Accident Sampling System (PASS) Ventilation System

SUMMARY:

This modification installed a fan and filter assembly to provide ventilation at the PASS sampling station. The primary purpose of the ventilation system is to remove heat from the PASS sampling station to extend the life of electrical components. A ventilation duct is attached to the top of the PASS panel. Air is drawn through the sampling station by a ventilation fan located outside the post accident sampling room. The fan discharges through a prefilter, HEPA filter, and charcoal adsorber into the labyrinth area located south of the post accident sampling room. Filtered air is subsequently drawn from the plant by the turbine building exhaust system. The ventilation fan will be operated whenever the PASS sampling station is operated. Manual fan control and indication is provided at a local instrument rack. Filter differential pressure indication is available to the operator inside the post accident sampling room. Additionally, this modification replaces a temporary PASS ventilation penetration seal with a permanent seal.

The only credible accident associated with this modification is the failure of the charcoal adsorber, which would lead to a small release of radioactivity. UFSAR Section 15.11 evaluates a similar accident, a failure of the gaseous radwaste system, including gross failure of the charcoal adsorber with an associated release to the environment. The off gas charcoal adsorber contains approximately 20,000 lbs of charcoal (Reference UFSAR Section 11.3.3.3.9) while the PASS adsorber contains approximately 13 lbs of charcoal. Based on the relative sizes of the two adsorbers and the accident evaluation from UFSAR Section 15.11, any release from the PASS charcoal adsorber will result in an insignificant amount of activity released compared to the off gas adsorber. The PASS charcoal adsorber was constructed and tested to the requirements of ANSI N509/510. Therefore, the gross failure of the housing is considered an

Safety Evaluation No. 89-0116 REV 1 (continued):

extremely unlikely event. The conduit penetration is sealed in accordance with Detroit Edison Specification 3071-198, meeting the requirements of the UFSAR.

In a PASS accident scenario it is highly probable that operators would require respiratory protection due to general area airborne radioactivity. Thus even without the PASS ventilation system, the operability of the PASS panel is not diminished. This change extends component life by removing internal heat via forced air flow. It also reduces operational exposures by drawing any airborne contamination created by sampling leaks or drips away from the personnel drawing the sample.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 89-0121 UFSAR Revision No. 5

Reference Document: EDP 2187 REV A Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Sprinkler System Pressure Indicator and Drain Installation in the Residual Heat Removal (RHR) Complex, Cable Spreading Room, and Diesel Driven Fire Pump Room

SUMMARY:

This modification installed pressure indicators and drains on wet pipe sprinkler systems in the RHR complex, cable spreading room, and diesel driven fire pump room in order to facilitate surveillance testing and bring the subject sprinkler systems into conformance with National Fire Protection Association (NFPA) Code Standard 13. The gages are for testing purposes and the drains are used for performing maintenance on the systems.

This design change does not change the function of these sprinkler systems. The addition of the gages and drains meets the design requirements of the existing systems and is bounded by previous pipe break analysis.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 89-0140 REV 1 UFSAR Revision No. 5

Reference Document: EDP 10531 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Non-Interruptible Air Supply (NIAS) Aftercooler Drain Modification; NIAS Aftercooler Safety Relief Valve Drawing and Calculation Corrections

SUMMARY:

This change modified the NIAS aftercooler drain system for each division by (1) replacing the original carbon steel system with upgraded Seismic Category I and QA Level I piping, valves, Y-strainer, and condensate trap made of stainless steel and (2) replacing the condensate trap bypass piping with a manual blowdown piping arrangement. In addition, drawings have been revised to reflect the actual drain system configuration, QA level classification, and piping group designation changes. As part of this modification, various drawing and stress calculation revisions were made to correct discrepancies between the design and field installations of NIAS aftercooler safety relief valves F207A and B.

The new drain system will eliminate the previous corrosion induced trap clogging problems and allow the use of a permanently installed alternate manual condensate blowdown flowpath when the condensate trap is out of service.

This modification does not change the design basis or operation of the NIAS aftercooler drain system. Failure of this drain system does not create an accident which goes beyond the previously evaluated loss of an entire division. The drawing and calculation changes do not impact the NIAS system design basis or operation.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 89-0154 UFSAR Revision No. 5

Reference Document: EDP 10714 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Installation of Primary Containment Water Level Instrumentation

SUMMARY:

A Detailed Control Room Design Review Team was established to fulfill the requirements of NUREG 0737 to identify and provide improvements in the control room that offer a high probability of improving plant safety by strengthening the man/machine interface. EDP 10714, in conjunction with previously installed EDP 8483, corrected one of these discrepancies by adding instrumentation which monitor drywell and torus pressure. This new instrumentation allows primary containment water level monitoring up to elevation 650 ft. Primary containment water level monitoring capability up to the maximum floodable level of the containment is needed for Fermi 2 Emergency Operating Procedures.

The design of the sensing line is consistent with the requirements of General Design Criteria 54 and 56 and Regulatory Guide 1.11 for instrument sensing lines penetrating primary reactor containment to ensure primary containment integrity and limit the potential offsite dose below 10CFR100 requirements. Seismic installation of conduit, cables, recorder, transmitters, and instrument tubing ensures that the integrity of surrounding components or systems will not be impacted. The design of the drywell pressure sensing line is consistent with the design of other existing instrument lines penetrating primary containment and falls within the envelope of an instrument line pipe break accident scenario as described in UFSAR Section 15.6.2. Additionally, the installation of the drywell pressure sensing line does not connect to the reactor coolant pressure boundary and does not impact any component or system related to the safe shutdown of the reactor. Utilization of the containment penetration for drywell pressure sensing is allowed by NRC approved License Amendment 57.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 89-0160 UFSAR Revision No. 5

Reference Document: EDP 9417 Section(s) 11.7

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Onsite Storage Facility Modifications

SUMMARY:

The on-site storage facility (OSSF) was originally designed for handling, shipping, and storage of dry active waste (DAW) and asphalted waste contained in 55-gallon drums. An overhead crane system was installed to cover essentially all areas of the building for remote operations with these drums. Two things occurred which slightly modified the initial design input assumptions. First, parts of the OSSF are used for other (but similar) purposes such as DAW sorting and storage of temporary shield blankets. Secondly, inasmuch as the asphalt system is not used, onsite vendors have been processing and storing (for short-term) radwaste in large (170 ft^3) liners. This modification will modify some of the OSSF walls enabling the aforementioned activities to be accomplished in a more efficient, safe manner with less radiation exposure. This is necessary because the walls are too high for the movement of the 170 ft^3 liners by means of the overhead crane.

The OSSF is not a Seismic Category I designed building, and wall alterations will not affect its structural integrity. The changes in the OSSF have no direct interfaces with other plant systems or with nearby equipment which could cause malfunction of equipment important to safety. The OSSF contains no equipment important to safety. The only equipment involved or interfaced with these changes are the OSSF crane, the forklift truck, and any portable vendor processing equipment. These changes will enable all of this equipment to work more efficiently, thus reducing the risk of an accident. Offsite doses to the general public from OSSF operations will not increase as a result of these changes, and any internal radiation-environmental changes have been fully discussed and approved by plant radiation protection personnel. This modification does not change or add any pathways for the release of radioactivity to the environment or for radiation exposure to the public.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0028 UFSAR Revision No. 5

Reference Document: EDP 8321 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: FW Heaters 2N, 2C, and 2S Level Transmitter Replacement

SUMMARY:

This modification replaced the original differential pressure type level transmitters on feedwater heaters 2N, 2C, and 2S with displacement type transmitters. This change will enhance plant reliability by reducing spurious heater alarms and control actions.

This modification does not change the operation or function of the affected feedwater heaters. The new transmitters have the same quality level, output signal, and loop scaling factors as the original transmitters. All downstream instrument loop components, setpoints, and functions remain the same. The small bore piping, insulation, and transmitters added to the second floor of the turbine building do not adversely impact the plant fire protection program.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 9\-0029 UFSAR Revision No. 5

Reference Document: EDP 8322 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: FW Heaters 3N, 3C, and 3S Level Transmitter Replacement

SUMMARY:

This modification replaced the original differential pressure type level transmitters on feedwater heaters 3N, 3C, and 3S with displacement type transmitters. This change will enhance plant reliability by reducing spurious heater alarms and control actions.

This modification does not change the operation or the function of the affected feedwater heaters. The new transmitters have the same quality level, output signal, and loop scaling factors as the or[illegible]l transmitters. All downstream instrument loop components, setpoints, a[illegible] [illegible]nctions remain the same. The small bore piping, insulation, and transmitters added to the second floor of the turbine building do not adversely impact the plant fire protection program.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0030 UFSAR Revision No. 5

Reference Document: EDP 8323 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: FW Heaters 4N, 4C, and 4S Level Transmitter Replacement

SUMMARY:

This modification replaced the original differential pressure type level transmitters on feedwater heaters 4N, 4C, and 4S with displacement type transmitters. This change will enhance plant reliability by reducing spurious heater alarms and control actions.

This modification does not change the operation or the function of the affected feedwater heaters. The new transmitters have the same quality level, output signal, and loop scaling factors as the original transmitters. All downstream instrument loop components, setpoints, and functions remain the same. The small bore piping, insulation, and transmitters added to the second floor of the turbine building do not adversely impact the plant fire protection program.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0032 UFSAR Revision No. 5

Reference Document: EDP 11222 Section(s) 5.5; A.1.96

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Outboard Main Steam Isolation Valve (MSIV) Internals and Valve Cover Replacement

SUMMARY:

The valve stem components and valve covers for outboard MSIVs B2103F028B, C, and D have been replaced to improve plant availability and limit personnel radiation exposure through reduction of local leak rate testing (LLRT) failures and by reducing the chance of a MSIV stem failure. The changes to each valve include: (1) replacing the valve stem and poppet and pilot poppet assemblies; (2) implementing a live-load packing assembly and antirotational devices; (3) machining the valve cover/bonnet to allow poppet backseating on the cover/bonnet instead of the valve stem and to provide adequate clearance for the use of stud tensioning devices; and (4) removing the valve, piping, and piping supports associated with the valve stem leakoff piping assembly as the new live load packing assembly eliminates MSIV stem leakoff.

The replacement of the outboard MSIV valve cover and internals, as well as the removal of the stem leakoff piping was carried out to improve valve operability, reliability, and maintainability. This modification does not affect the design basis, function, or sequence of timing of the MSIVs. The new parts are fabricated to the original design requirements, but are modified to improve packing performance and reduce the potential for valve failure. Removal of the stem leakoff piping does not create a new equipment failure mode as a packing leak would release steam/water in the same general area as before.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0037 UFSAR Revision No. 5

Reference Document: EDP 10610 Section(s) 6.2

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Containment Nitrogen Inerting, Venting, and Purging System Isolation Valve Upgrade

SUMMARY:

This modification upgraded the containment inerting, venting, and purging system isolation valves by upgrading the limit switches and position indication balance of plant power supply; removing the non-Q pneumatic control solenoid valves; and replacing the QA1 pneumatic control solenoid valves with a type that has a higher maximum operating differential pressure rating. This modification also deleted the existing non-qualified automatic containment pressure control system. This change brings the containment nitrogen inerting, venting, and purging system isolation valves in agreement with the description in the UFSAR and resolves the concerns of NRC Information Notice 88-24 and NRC Notice of Violations 89011-01A and 89011-02A.

With the exception of the automatic containment pressure control system that was deleted, the new control configuration functions identically to the previously analyzed system. The automatic mode of the containment pressure control system was not used and the UFSAR, Technical Specifications, and NRC Safety Evaluation Report do not take credit for it in any accident analysis. Removal of the non-safety controls eliminates a condition wherein a common mode failure within the non-safety circuitry could have caused any of the isolation valves to open inadvertently. The new components and circuitry meet the requirements for class 1E safety grade components. This modification does not impact the stroke times of the isolation valves.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0039 UFSAR Revision No. 5

Reference Document: EDP 11331 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Reactor Head Vent Solenoid Valve Supply Power and Associated Equipment Removal

SUMMARY:

This modification disconnected the power supply and removed all associated control center controls for reactor head vent [illegible]id valves B21F403 and B21F404. Both valves are not functional because [illegible]ent piping upstream was plugged per EDP 10792. (See SE 89-0196 in Fermi 2 Safety Evaluation Summary Report, 1989.) This modification satisfies human factor criteria by removing equipment which is no longer required.

This modification does not affect the design, function, or operation of the reactor vessel because it is functionally equivalent to the previous configuration created by EDP 10792.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0040 UFSAR Revision No. 5

Reference Document: EDP 9385 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Addition of Remote Manual Close Control Switch in the Control Room for Drywell Equipment Drain Pump Discharge Valve

SUMMARY:

This modification provided a remote manual close control switch in the control room for drywell equipment drain pump discharge valve G1154F018 to comply with Reg. Guide 1.62 and the UFSAR. This modification replaces an interim design change (PDC 9384) that located the remote manual close control switch in the relay room.

This modification does not change other existing functions of the valve. It enhances the operators' ability to quickly close the valve if automatic isolation fails. The new components installed by this modification are of the same quality as the existing components. Single failure criteria for any postulated failure of G1154F018 is still satisfied in that, for any postulated failure of the new switch, a redundant outboard containment isolation valve exists to carry out isolation for this drain path.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0041 UFSAR Revision No. 5

Reference Document: EDP 7571 Section(s) N/A

Table(s) 6.2-2

Figure Change [X] Yes [ ] No

Title of Change: Scram Discharge Volume (SDV) Drain Valve Replacement and Addition of Manual Block Valves and Test Taps

SUMMARY:

This modification replaces the SDV inboard drain valve C1100F011 and SDV outboard drain valve C1100F181 and installs manual block valves, test taps, and test tap block valves to facilitate local leak rate testing (LLRT) of the SDV vent and drain isolation valves. The original SDV drain valves have experienced excessive leakage in three consecutive as-found LLRTs. The resultant refurbishment has resulted in outage delays. The new drain valves provide a tighter shutoff capability. The SDV vent and drain manual block valves, test taps, and test tap block valves have been added to facilitate individual valve LLRTs. This will allow testing with the valves isolated from the control rod drive system (CRD) and will result in improved trouble shooting capability and more efficient testing and valve rework. The manual block valves and test tap block valves are designated as locked valves and are controlled under locked valve administrative controls.

The SDV power supply, controls, and indicators are not altered or impacted by this modification. C1100F011 and C1100F181 use the same pneumatic supply for opening and spring closure as the original valves. This modification maintains and satisfies the design requirements of General Electric Design Specification 22A6249 Revision 3, Data Sheets 22A6249AB Revision 10, and the Detroit Edison Specifications. The effect on drain flow capacity due to the replacement of the drain valves and installation of the manual block valves was evaluated in a design calculation. It was concluded that although this modification reduces the drain flow capacity by several gallons per minute, it does not adversely impact SDV draindown capability. The effects of leaving a test tap block valve or leaving a drain or vent manual block valve closed were evaluated. Leaving a test tap block valve open would result in the leakage of reactor condensate. This leakage is no different than the leakage that could be experienced if other manual valves in the CRD system are left open and, therefore, the addition of the two test tap block valves does not significantly change the probability of any one valve being left open. Leaving a drain or vent manual block valve closed will impede condensate drainage. However, redundant SDV instrument volume level transmitters will detect the increasing level and alert the operators or initiate a reactor scram if corrective action

Safety Evaluation N. 90-0041 (continued):

is not taken to restore drainage. The LLRT test procedures, locked valve program, and routine operator rounds provide multiple administrative coverage to ensure that the SDV vent and drain block valves are open and the test tap shutoff valves are closed and capped.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0044 REV 3 UFSAR Revision No. 5

Reference Document: EDP 11044 Section(s) N/A

Table(s) 8.3-1

Figure Change [ ] Yes [X] No

Title of Change: Installation of a Dedicated 208Y/120V Maintenance Power Distribution System in Containment

SUMMARY:

This modification provided a permanent 208Y/120V maintenance power distribution system in containment. The system provides two separate distribution networks; one within the drywell and the other in the secondary containment. The new system consists of (2) 45 KVA transformers with their associated raceway/cables and distribution networks. The primary power sources for the drywell and the secondary containment are Southside/MCC 72E-3A and Northside/MCC 72B-4A, respectively. An alternate source of power is provided utilizing a transfer switch and prefabricated cables capable of tying into the existing 480V maintenance distribution system. The system is de-energized when not in use and derated from 100 amperes to 70 amperes when used during startup and power operations.

This system has no functional purpose related to or interface with any safety systems. All components are sized and protected so that their electrical ratings are not exceeded. A fault within this system will only affect maintenance activities and will not de-energize other electrical equipment. The installation is non-Q with the exception of the containment penetration which is QA level 1. This change is in compliance with Reg. Guide 1.63. The use of double fuses and de-energization is in accordance with the UFSAR. Plant procedures administratively control deenergization and derating of the system.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0046 UFSAR Revision No. 5

Reference Document: EDP 7703 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Addition of a Body Feed System to the Condensate Polisher Demineralizer System

SUMMARY:

This modification adds a body feed system to the condensate polishing demineralizers. This consists of adding a mixing tank fill line, mixing tank drain piping, and individual vessel influent feed tubing. Body feeding consists of metering small amounts of demineralizer resin to the filter columns (septa) during demineralizer operation. Body feeding compensates for precoat imperfections, promotes depth filtration, and reduces the differential pressure buildup rate. As a result, filter performance is improved and demineralizer precoats are reduced.

The addition of body feeding is within the design resin loading of the demineralizers because the initial precoat loading is lower (0.15 psf vs. 0.20 psf). The loss of the body feed system will not impact any previously evaluated accident analyses or affect safety related equipment. If interlocks fail and resin is continuously fed to the demineralizers, the vessels will be removed from service due to high differential pressure. Any pipe breaks in the body feed system will not impact the reactor building leakage analysis. The addition or loss of the body feed system does not change the technical specification chemistry limits.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0054 REV 1 UFSAR Revision No. 5

Reference Document: EDP 11429 Section(s) 6.2

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Modification of the High Point Vent on Division I RHR Return Piping

SUMMARY:

This EDP modifies the high point vent by removing one of the two vent isolation valves, shortening the remaining pipe spools, and modifying the sock-o-let fillet weld to reduce the susceptibility to vibration loading.

The proposed design meets the dual containment barrier design code requirement through the use of one isolation valve and a threaded pipe cap. The 10 CFR 50 General Design Criteria 55 and 56 provisions for primary containment penetrations and reactor coolant pressure boundary design are met through the use of: (1) a manual isolation valve and a threaded pipe cap on the LLRT vent connection, and (2) the RHR system inboard containment isolation valve and the vent isolation valve.

This modification has no effect on the venting operation. The alternating stress levels will be reduced by a factor of approximately three. The classification of the piping beyond the remaining vent isolation valve is currently ANSI B31.1, 1500# rating and will remain so in the revised design. An effect of removing the vent isolation valve is a reduction in the administrative controls required to ensure vent closure following use since there is one less barrier between the header piping and containment. The existing administrative controls on valve lineup and cap verification assure leak tightness.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0061 REV 1 UFSAR Revision No. 5

Reference Document: EDP 7808 Section(s) 9.5

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Replacement of the Sound Powered Headset Communication System with a Telephone System

SUMMARY:

The purpose of this modification is to convert the existing sound powered headset system to a telephone sytem; provide additional telephone jacks, telephones, and sound proof telephone booths; provide additional Hi-Com public address handsets, amplifiers, and loud speakers; and improve the radio and Hi-Com communications in the control room. This modification provides expanded communications coverage and enables the plant operators and I&C technicians to conduct their activities more efficiently.

This change does not affect the operation of safety related equipment or systems and does not impact existing accident analyses. This modification pertains to non-Q systems and does not impact the fire protection or appendix R criteria. The direct communication system between the control room and the refueling platform required by the technical specifications is not affected by this modification.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0088 UFSAR Revision No. N/A

Reference Document: EDP 10328 Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Refueling Platform Power/Control and Communications Cable Replacement

SUMMARY:

This modification replaced the power/control cable and the communications cable on the refueling platform with a single composite cable that handles both the platform power/control and communications functions.

This modification does not create an unreviewed safety question. There is no potential for fuel damage or radioactive release due to the extent of potential combustibility of cables. Fire hazards are not created because the location is already considered a light combustible loading area and the cable is separated from other cables. No fuel damage or accidental criticality will occur. During refueling, the fuel platform will fail "as is" on a loss of power. During normal operation, the refueling platform is de-energized and reactor vessel head is installed.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0097 UFSAR Revision No. 5

Reference Document: EDP 10452 Section(s) 6.2

Table(s) 6.2-2; 6.2-13; 6.2-15

Figure Change [X] Yes [ ] No

Title of Change: Main Steam Isolation Valve Leakage Control System (MSIVLCS) Isolation Valve Replacement

SUMMARY:

This modification replaced the original solenoid operated MSIVLCS isolation valves B21F433, F434, F437, and F438 with solenoid piloted bellows sealed air operated valves. To accomplish this valve changeout four solenoid operated air pilot valves; two non-interruptible instrument air system (NIAS) isolation valves, P5000F1007 and P5000F1008; and the associated piping, tubing, conduit, cable, and associated supports were installed and four snubbers were deleted. The MSIVLCS isolation valves were replaced because all four had previously failed the required functional testing and F434 had failed a local leak rate test. This resulted in increased maintenance costs, extended plant down time, and increased personnel exposure.

This modification enhances system operability and reliability. The modification has been designed in accordance with Detroit Edison approved procedures and specifications. There is no change to the system design basis, function, or sequence of operation. The effect of a single failure which results in the failure of one isolation valve to open or close does not reduce MSIVLCS redundancy or reliability. A malfunction of the components installed by this design does not impact the NIAS system, the associated power supplies, control circuits, or any adjacent equipment important to safety.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0099 UFSAR Revision No. 5

Reference Document: EDP 11633 Section(s) 9.4

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Recirculation Pump Motor Generator (MG) Set Cooling Unit High Temperature Trip Setpoint Change

SUMMARY:

This modification changes the trip setpoint of the recirculation pump MG set cooling units from 105°F to 125°F in order to prevent unnecessary trips of the cooling units. Normal cooler outlet temperatures are approximately 100°F.

This modification has no effect on other equipment or MG set instrumentation and logic. MG set motor temperature will still be maintained well below its high temperature alarm and trip setpoints. By allowing the cooling units to operate longer, their ability to remove waste heat is improved and MG set performance is enhanced.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0105 UFSAR Revision No. 5

Reference Document: EDP 11260 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Reactor Water Cleanup (RWCU) Filter Demineralizer Effluent Flow Controller Replacement

SUMMARY:

This modification replaces the pneumatic recording control stations G33R174A & B with new pneumatic controllers. Replacement of the recording pen ink in the old controllers caused inadvertent changes in filter demineralizer flowrate which resulted in unnecessary dumping of demineralizer resins and nuisance alarms in the control room. The new controllers have independent auto and manual control units with fully balanceless and bumpless transfer, a highly visible scale display, and no recording function. Operations and Chemistry personnel requested elimination of the recording function because it does not provide vital information.

This modification does not change the function of the controller or system and does not adversely affect any component or system related to the safe shutdown of the reactor. Removal of the recorder function enhances system performance because the recorder maintenance activities which caused inadvertent changes in demineralizer flowrate have been eliminated.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0109 REV 1 UFSAR Revision No. 5

Reference Document: EDP 11300 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Installation of a Second Reheater Seal Tank (RST)

SUMMARY:

This modification installed an additional RST and the required piping, components, and associated controls to provide a separate RST for each moisture separator reheater (MSR). The west MSR now drains to the original (north) RST and the East MSR drains to the new (south) RST. This modification also modifies the north RST to accommodate the new piping configuration. The original centerline drain nozzles are capped and a new nozzle has been installed at the top of the tank. A nozzle and vent line to the 5S feedwater heater extraction line was added. Baffles were installed inside the tank. The north RST differential pressure type level transmitters which control the normal and emergency drains were replaced with displacement type transmitters. The RST level alarms and indication test pushbutton were also eliminated.

This modification eliminates RST level instability and/or tank blow through at high power levels and during turbine intercept valve testing. The addition of a second RST doubles the capacity of the drain system. Relocation of the drain nozzle ensures that a higher drain level can be maintained without covering the drain nozzle. The replacement of the level transmitters makes the level control system less sensitive to wave action and turbulence in the RSTs resulting in less severe level transmitter output fluctuations and fewer erroneous signals. The installation of the baffle plate reduces wave action.

This modification does not modify the MSRs or MSR drain flow rate. The piping is designed in accordance with applicable codes and Detroit Edison specifications. There is no impact on the previous pipe break analysis in the UFSAR. The addition of the second RST does not affect the passive steam bypass system. The MSR drain lines are sized to carry the same drain flow as the previous configuration. An analysis of reheater bypass flow shows that, for the 0%, 13%, and 26% flow bypass conditions considered, the reheater flows lead to lower peak fuel pin heat fluxes than those obtained by using the reheater flow pattern assumed in the bypass flow analysis in the UFSAR. Analysis of the effects of routing the RST vents to the 5N and 5S feedwater heater extraction steam lines indicate that the extraction flow will increase by 3% and the effect is, therefore, considered negligible.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0114 REV 1 UFSAR Revision No. 5

Reference Document: EDP 11190 Section(s) 9.1

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: New Fuel Uprighting Stand Relocation

SUMMARY:

This modification relocates the fuel uprighting stand 1.25 feet north and 4 feet west in order to bring the stand within reach of the new fuel transfer crane. This will allow the new fuel crane to be used to transfer fuel bundles from the new fuel uprighting stand to the new fuel inspection stand. Six new floor anchors are installed, a UFSAR Figure change has been made to delete the abandoned or non-existent anchor locations, and two UFSAR sections have been revised to rename the "unloading stand" as the "new fuel uprighting stand".

This modification does not alter the safety function of any other plant system or equipment. The modification does not impact the heavy load analysis for the 5th floor RB storage areas, structures or overhead crane as the new fuel uprighting stand weighs less than 2000 lbs and is not considered a heavy load.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0115 REV 1 UFSAR Revision No. N/A

Reference Document: EDP 11282 Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change Reheater Seal Tank Control Center COP H11P805 Changes and Additions

SUMMARY:

This modification made the following changes to the control center operating panel COPH11P805:

1. Provided the controls and indication for a second Reheater Seal Tank being added by EDP 11300.

2. Addressed a portion of HED-455 by removing a number of unused controls (push-button/indicators) for valves removed or abandoned in place by PDC 9357, PDC 9359, EDP 8938, EDP 10778, and EDP 11300.

This modification does not change the function or operation of the feedwater heater drain system or its associated components. This system does not affect the operation of any systems required for the safe shutdown of the plant nor does it change any accident scenarios credible to this control room panel.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0116 UFSAR Revision No. 5

Reference Document: EDP 11502 Section(s) 6.2

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Division II Residual Heat Removal Return Piping High Point Vent Modification

SUMMARY:

This modification removes the second isolation valve and replaces it with a threaded pipe cap. The purpose of the change is to reduce the vibration induced stress levels of the vent line and reduce the probability of failure at the weldment of the vent line and 24" RHR pipe "sock-o-let". This design does not satisfy the requirements of UFSAR section 6.2.4.4.3 which states that test, vent, and drain connections on the Class 1 system (which are part of the containment boundary) are provided with at least two isolation valves and are sealed with a threaded pipe cap. However, 10 CFR 50 General Design Criteria 55 and 56 are met because penetration to primary containment isolation is accomplished through the use of the remaining vent valve and pipe cap; reactor coolant pressure boundary isolation is accomplished through the use of the remaining vent valve and the RHR system inboard containment isolation valve.

This modification does not change the safety function or operation of the RHR system. Any failures introduced by this modification are bounded by the small break analysis. Administrative controls exist to ensure validation of valve closure after use.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-011s UFSAR Revision No. N/A

Reference Document: EDP 11819 Sectio(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: High Pressure Coolant Injection (HPCI) High Steam Flow Transmitter Time Characteristics Modification

SUMMARY:

The purpose of this modification is to increase the reliability of the HPCI high steam flow isolation instrumentation loop by filtering out undesirable process noise and eliminating noise as a cause of spurious HPCI isolations. A capacitor has been added between the output of the transmitter and the input of the associated trip unit. The new capacitor adds 2 seconds to the loop response time but overall loop response time is well below the Technical Specification response time requirement of 13 seconds.

This modification does not change the function of the HPCI system. The analyzed basis for a high energy line break in the HPCI steam supply line is not compromised by this change because no credit is taken for the operation of the high steam flow logic to mitigate a steam supply line break. Failure of the added components does not affect any accident or transient analysis and can be detected during shift surveillances.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0121 REV 3 UFSAR Revision No. 5

Reference Document: EDP 11274 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Installation of Station Air Header Pressure Instrumentation for the Control Room

SUMMARY:

The purpose of this modification is to provide the control room operators with a more distinct indication of station air pressure by installing station air header pressure indication in the control room. The modification consists of the installation of a pressure transmitter, a pressure indicator, and the associated tubing and cabling. The original air pressure indication, consisting of Division I and Division II NIAS compressor discharge pressure, was reviewed by the Detailed Control Room Design Review Team and found to be an inadequate indication of station air pressure. This concern is documented in Human Engineering Deficiency HED 1177.

This modification does not adversely affect any component or system related to the safe shutdown of the reactor. The indication provided by this modification is not required for any safety functions nor is it required to operate after a design basis accident. The components used for this modification are similar to other components in the plant that have proven reliability.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0123 REV 1 UFSAR Revision No. 5

Reference Document: EDP 11266 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Control Room Multipoint Recorder Replacement

SUMMARY:

This modification resolves part of Lead Human Engineering Discrepancy HED 987 by replacing the 18 original L&N Speedomax multipoint recorders with 13 Westronics Series 3000 and DDR10 recorders. The new recorders provide a legible trace and have an easily readable display, point selectability, and low maintenance. Due to the physical size of the replacement recorders, some recorders that monitor similar variables have been combined into one recorder to provide room for installation and free up control room operating panel space for future instrumentation.

This modification will not adversely affect the function of existing equipment or their performance as originally designed. Performance is enhanced by the improved readability, added point selectability, and reduced maintenance.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0125 UFSAR Revision No. 5

Reference Document: EDP 11803 Section(s) 10.2

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Low Point Drain Installation for Feedwater Heater 4 Extraction Steam Line

SUMMARY:

This modification installed a drain on the extraction line of the 4N feedwater heater between check valve N3000F402A and the heater at a previously undrainable low point. A walkdown of the feedwater heater room during plant outage 90-04 revealed damage to the 4N heater support pedestal and extraction steam line pipe support N30-3199-G17. It was also noted that the 4N feedwater heater shifted north approximately 2.5 inches. The cause was determined to be a fluid transient in the extraction steam line.

The new drain line is connected to an existing drain between N3000F402A and N3016F603. Both drain lines are isolated by a common isolation valve. To avoid having an open bypass around check valve N3000F402A, a check valve has been installed in the new drain valve.

This modification has no effect on any of the UFSAR chapter 15 accident analyses. However, since there are no provisions for testing the new drain line check valve as required in UFSAR section 10.2.2.6, the impact of water induction due to a failure of the subject check valve was analyzed. It was determined that the open area of the subject check valve disc would not allow a significant amount of drains from the 4N feedwater heater to back flow into the low pressure turbine and that no turbine overspeed hazard exists.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0136 UFSAR Revision No. 5

Reference Document: EDP 11527 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Main Steam Line Pressure Tap Removal

SUMMARY:

The purpose of this modification is to reduce the potential for main steam line leakage by removing the pressure tap source valves and capping 4 nuclear boiler system pressure taps and isolation valves on each of the main steam lines. These taps were originally used to obtain pressure data during startup testing and are no longer used. This modification affects the 3/4 inch instrumentation lines located between the outboard main steam isolation valves (MSIV) and the third MSIVs downstream of the main steam drain lines.

No new equipment is added by this modification. This modification will not affect the operation of the MSIV leakage control system or the main steam lines drains. In the event of a LOCA, all potential MSIV leakage originating from primary containment will be contained. The potential for steam line leakage is reduced because removal of the pressure taps reduces vibratory stress loads at the tap connections. This design satisfies UFSAR stress limits for pressure, weight, seismic, and transient loadings.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0137 UFSAR Revision No. 5

Reference Document: EDP 8972 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Post Accident Sampling System (PASS) Valve Cable Rerouting and Reterminating

SUMMARY:

This modification rerouted and reterminated four cables that are part of the control scheme for the PASS residual heat removal liquid sample valves P34F402A and P34F402B to ensure that each valve is controlled and powered from its respective divisional control panel. Originally, the valves were wired such that the Division I valve was controlled and powered from Division II panel H11P618. The Division II valve was controlled and powered from Division I panel H11P617. This modification ensures that the sample valves are controlled from the same division as the division being sampled. This modification also retags the valves and associated limit switches to agree with other PASS system components and updates UFSAR figure 11.4-8 to show the corrected PIS numbers.

This modification does not change the function of the PASS system as described in the UFSAR. Plant operation is enhanced in that the sample valve controls for each division RHR sample train are on the proper control panel.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0138   UFSAR Revision No. 5

Reference Document: EDP 9922   Section(s) 7.2

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Reactor Protection System (RPS) Electric Protection Assembly (EPA) Logic Card Replacement

SUMMARY:

The purpose of this modification is to enhance the performance of the EPAs by replacing the original design EPA logic card with an improved logic card and modification kit. This modification addresses the concerns of G.E. SIL #496, revision 1 and RICSIL #026. EPA performance and reliability will be enhanced by the following:

- Elimination of spurious trips from causes internal to the logic card

- Resolution of the IC chip lockup concern described in RICSIL #026

- Provision for a connector interface to the logic card

- Improved access to test points needed during routine calibration of the logic card

- Reduction of the number of mechanical cycles to which the circuit breaker is subjected by adding a switch to disconnect the circuit breaker undervoltage release (UVR) coil during logic card calibration

- Rated voltage is provided to the UVR coil.

As a result of this modification, the time delay continuous adjustment range has changed from 0.2 - 3.6 seconds to 0.3 - 3.6 seconds.

This modification does not change the function of the EPAs or RPS. It enhances RPS availability and improves EPA performance. The change in time delay continuous adjustment range is still within the analyzed limits and is, therefore, acceptable.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0139 UFSAR Revision No. 5

Reference Document: EDP 11068 Section(s) 7.1; 7.2; 7.6

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Process Computer Core Monitoring Software Replacement

SUMMARY:

This modification replaced the process computer system core monitoring GEXL+NSSS software with General Electric 3D-MONICORE software. DEC work station computers were also added to support the 3D-MONICORE system.

This modification does not change the process monitoring system. The DEC work station computers do not provide information to other plant systems. The absolute accuracy of the 3D-MONICORE core physics model has been established by comparing its calculated results with gamma scan measurements carried out at Edwin I. Hatch, Unit 1 following cycles 1 and 3. The 3D-MONICORE uncertainties are covered by the safety limit margins employed in the process computer. In the event that the process computer is not available, the software operates from an onsite DEC Microvax C3800 with full backup from a DEC Vax Station 3100. This software backup capability is a significant improvement over the former system.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0142 UFSAR Revision No. 5

Reference Document: EDP 11889 Section(s) 6.4; 7.1; 7.3

Table(s) 9.4-2

Figure Change [ ] Yes [X] No

Title of Change: Control Center HVAC (CCHVAC) Control Logic

SUMMARY:

The purpose of this modification is to change the CCHVAC control logic to: (1) prioritize the recirculation mode of operation over the chlorine mode and (2) require the operation of the mode select reset pushbutton for mode actuation. The recirculation mode was prioritized to ensure that, in the event of a chlorine detector failure followed by a subsequent LOCA/radiation release, the CCHVAC will automatically transfer to the recirculation mode. The former logic required the operators to manually initiate the recirculation mode. The mode selector logic was changed to require mode select reset button operation for all modes of actuation. This change provides a uniform method for mode initiation. The former logic did not always require the operator to press the reset button to initiate a mode. For some modes, the mode would be initiated when the mode was selected. For other modes, the mode selector reset button was pressed to initiate the selected mode.

This modification does not add any equipment or affect the equipment within the system. The original intent of the CCHVAC mode selection process has been maintained. The automatic initiation signals for both the chlorine and recirculation modes override any manually selected mode. The UFSAR accident analyses address the operation of the recirculation and chlorine modes individually. A LOCA is not addressed when operating in the chlorine mode and a chlorine accident is not addressed when operating in the recirculation mode. A single failure of a chlorine detector will not prohibit the CCHVAC recirculation mode from automatically initiating. Calculations show, that with CCHVAC in the recirculation mode, all chlorine accident scenarios result in control center chlorine concentrations less than the toxicity limit of 15 ppm. Control center personnel will be limited to 5 rems whole body or its equivalent consistent with the requirements of General Design Criterion 19 of 10 CFR 50, Appendix A.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0147 UFSAR Revision No. 5

Reference Document: EDP 9310 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Removal of Limit Switches on Reactor Water Cleanup (RWCU) System Check Valves

SUMMARY:

This modification removed the actuator and disc position limit switches on RWCU check valve G330F121 and the actuator position limit switches on RWCU system check valve G330F120. In addition, the controls and position indication for G3300F121 and the actuator position indication for G3300F120 that are located in the control room have been removed. The solenoid valve for G3300F121 has been removed and fittings and labels have been provided at the air manifold shutoff valve and check valve actuator tubing to identify the connection points for the installation of a pneumatic jumper for future inservice inspection testing (IST). The position limit switches removed by this modification have been high maintenance items and their repair has raised ALARA concerns.

This modification does not change the function of these check valves. The controls, actuator position indication, and disc position indication are not required for G3300F121. G3300F121 only needs an actuator when it is required to be open as part of a valve lineup for local leak rate testing. The actuator position indication is not required for G3300F120. G3300F120 only requires disc position indication for IST program testing. Removal of the limit switches does not impact the seismic calculations for these valves as the weight of each limit switch is less than 10 ounces. This is less than 1% of the total valve weight and, as such, is insignificant.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0155 REV 1 UFSAR Revision No. 5

Reference Document: EDP 11591 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Removal of Feedwater Check Valve Limit Switches

SUMMARY:

This modification made the following changes to feedwater check valves B2100F076A, B2100F076B, B2100F010A, and B2100F010B:

B2100F076A and B

1. The actuator limit switches were removed.

2. The actuator position indication was removed from control room operating panel COP insert H11P603A504.

3. A connection was installed in the actuator air supply piping to allow the use of a pneumatic jumper to facilitate actuator testing.

B2100F010A and B

1. The actuator and disc position limit switches were removed.

2. The actuator solenoid valves were replaced with shutoff valves.

3. Local wiring and flexible conduit were removed and the resultant conduit and electrical box openings were plugged.

4. The control indication was removed from control operating panel COP insert H11P603A504.

This modification removed unneeded indication for B2100F076A and B and unneeded indication and controls for B2100F010A and B. It also enhances the maintainability of the check valves in that their limit switches have been high maintenance items.

Safety Evaluation No. 90-0155 REV 1 (continued):

This modification does not affect the check valves' ability to close on reverse flow and does not degrade their performance as containment isolation valves. The valves will respond to the accident conditions as assumed in the UFSAR. The B2100F076A and B position indication required by the IST program remains in place. This design reflects consistency with NUREG 0700, "Guidelines for Control Room Design Reviews" for control placement on control room operating panels.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0156   UFSAR Revision No. N/A

Reference Document: EDP 11354   Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Instrument and Control (I&C) Instrument Line Snubber Removal

SUMMARY:

This evaluation justifies the removal of 100 snubbers from 25 I&C instrument lines. 28 of the snubbers have been replaced with struts and the remaining 72 snubber locations do not require support. The original piping stress analysis conservatively assumed excess flow check valve testing would occur at a reactor temperature of 546 degrees F. As a result, the original pipe stress analysis predicted large thermal movements. Snubbers were installed to accommodate thermal movement and seismic loads. However, excess flow check valve testing is performed when the piping temperature is less than 200 degrees F (operational condition 4). Reanalysis of the above instrument lines, using 200 degrees F as the operating temperature, allowed the above snubbers to be removed without overstressing the instrument lines or the remaining supports. The analysis also indicated that 28 of the above snubbers would be replaced by struts. Elimination of these snubbers will: (1) decrease the costs associated with the periodic maintenance, inspection, and testing of snubbers; (2) decrease the radiation exposure to personnel; and (3) decrease the potential for extended outages due to snubber failures.

This modification has no adverse impact on plant safety nor does it not impact the function or operation of any interfacing system. The ability of the subject instrument lines to function has not been degraded by the removal of these snubbers. The qualitative vibration assessment in UFSAR section 3.9.1.1.3 has not been affected.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0160 UFSAR Revision No. 5

Reference Document: EDP 11563 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Installation of Reactor Building HVAC (RBHVAC) Ductwork Catch Pan and Drain

SUMMARY:

This modification added a duct section in the RBHVAC ductwork at the scram discharge volume (SDV) vent to RBHVAC duct connection. The ductwork section is made of stainless steel and equipped with a catch pan and drain line. The drain line is routed from the ductwork to reactor building floor drain sump G1101D076 via a basement level instrument blowdown drain. An access panel is provided on the exposed side of the duct for decontamination purposes. Plant operators had observed water on the floor of the residual heat removal Division 1 heat exchanger room after reactor scrams. Analysis of the water indicated that the source was reactor condensate. Further investigation revealed that the water was entering the RBHVAC duct work from the SDV vent line. Inspection of the galvanized steel ductwork revealed significant corrosion damage. Installation of the stainless steel ductwork will prevent further corrosion damage and the catch pan and drain will prevent condensate spills.

This portion of the RBHVAC ductwork is not required for safe shutdown of the reactor or for maintaining secondary containment. This modification does not change the operation or function of the SDV system, the RBHVAC system, or floor drain sump G1101D076. The installed components are structurally mounted to prevent interference with safety related components or systems under design accident and seismic conditions. The design of the drain line minimizes crud trapping and hot spot formation and provisions are made for flushing the line if contamination is detected.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0163 UFSAR Revision No. 4

Reference Document: EDP 11974 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Installation of Primary Containment Atmospheric Grab Sample Taps

SUMMARY:

This modification installs grab sample taps at the primary containment radiation monitoring skid H21P284 for the purpose of taking primary containment atmosphere grab samples. The location of the taps ensures that the manual grab sample rig will be automatically isolated from containment in the event of a LOCA. Each sample tap is equipped with a manual isolation valve and a plug. The operation of the isolation valve is administratively controlled by independent verification by its users. Since the sample taps are located in Division I, the Division II sample taps may be used with strict administrative controls when the Division I taps are not available. The use of the Division II taps is justified in another safety evaluation.

The new design maintains primary containment integrity during any design basis accident as well as providing grab sample capability during normal plant operations. This modification does not impact the reactor coolant boundary nor does it change the operation or function of the primary containment monitoring system (PCMS). The administrative controls placed on the operation of the sample valves ensure that any potential for breaching containment or rendering the PCMS inoperable due to the operation of these valves is eliminated.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0004 UFSAR Revision No. 5

Reference Document: EDP 7832 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Turbine Building Closed Cooling Water (TBCCW) Corrosion Control

SUMMARY:

This evaluation justifies the use of sodium molibate/sodium nitrate as a corrosion inhibitor in the TBCCW system. An evaluation of the pure water application method (no corrosion inhibitor chemistry) used in the TBCCW system showed that only 5% of the system volume was flowing through the filter demineralizers and that turbidity levels were unacceptable. The evaluation concluded that chemical treatment of the TBCCW systam was required. To monitor the performance of the corrosion inhibitor, coupons were installed in the return flow path upstream of the heat exchangers. The coupons are made of metals found in the TBCCW system. Pipe taps were installed in the supply and return headers to allow the use of a temporary portable filter demineralizer when the corrosion inhibitor cannot be injected into the system.

This modification enhances the reliability of the TBCCW system by constantly monitoring for system boundary degradation. The TBCCW system is not safety related and is not required for the safe shutdown of the reactor. If a pipe tap breaks and causes a breach of the TBCCW system boundary, the proble would be identified by the observation of opening makeup tank fill valve, low makeup tank level, and low system pressure alarms. The initiation of any of these alarms will alert the operator to take appropriate action to mitigate the loss of inventory. If a coupon should break free, the pump suction strainers will prevent it from entering the pump.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0006 UFSAR Revision No. 5

Reference Document: EDP 11577 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Residual Heat Removal (RHR) Air Actuator Replacement and Limit Switch Removal

SUMMARY:

This modification replaced the air actuators and removed the actuator limit switches on RHR check valves E1100F050A and E1100F050B. The new actuators increase the check valve disc stroke from 22% to 68% open. This ensures that the check velve disc open limit switch will actuate when the valve is opened and provide open indication in the control center. The former actuator's stroke was too short to actuate the open limit switch. As a result, the position indicators in the control center showed no open or closed indication when the check valve was opened. The removal of the actuator limits switches eliminates unneeded indication from the control center.

This modification does not change the function of the check valves. The actuators are used for check valve testing only. The change in actuator nitrogen consumption does not impact the operation of the nitrogen system. The change in actuator weight does not impact the existing seismic analysis. Check valve position can still be observed in the control center by using the disc position indication.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0010 UFSAR Revision No. 5

Reference Document: EDP 11948 Section(s) 8.2

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: 4160 V. Bus Voltage Low Alarm Voltmeter and Controller Replacement

SUMMARY:

This modification replaced the contact making voltmeters and controllers for the Division I and II 4160 V. bus low voltage alarms with digital voltmeters and controllers. The internal components and the indicator for the original instrumentation is obsolete.

The new instrumentation performs the same function as the original instrumentation in that a degraded grid voltage condition alarms to allow the operator to take action before an undervoltage bus trip actuates. The alarm setpoints and time delay settings have not been changed.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0013 UFSAR Revision No. 5

Reference Document: EDP 10442 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Replacement of the 5N and 5S Feedwater Heater Condensate Overpressure Thermal Relief Valves

SUMMARY:

This modification replaced the 5N and 5S feedwater heater condensate overpressure thermal relief valves, N2000F302A and B. The original relief valves had a 40% blowdown range. This range was considered unacceptable because it caused the relief valve setpoint to drift resulting in valve lifts during plant transients. The new relief valves have a blowdown range of 20%. The new valves are equipped with a built-in travel stop to prevent valve over-travel and possible setpoint drift.

The setpoint, valve size, and mass flow rate of the new relief valves are the same as the former relief valves. There is no change to the design function or intended system response to plant operating transients. There are no failure modes or mechanisms possible that are not assumed for the former valves.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0024 UFSAR Revision No. 5

Reference Document: EDP 12176 Section(s) 9A.4

Table(s) 9A.6.1-1

Figure Change [X] Yes [ ] No

Title of Change: Modification of Division I Switchgear Room Smoke Detection System

SUMMARY:

This modification installs a single smoke detector in the ceiling of a Division II cable enclosure located in the division I switchgear room. The lack of a detector in this enclosure had been declared a limiting condition for operation and a roving fire watch patrol had been assigned as a compensatory measure. Installation of the smoke detector provides the fire detection instrumentation required by section II.D of NUREG-0798, Supplement 5, "Safety Evaluation Report Related to the Operation of Fermi-2" and eliminates the need for the fire watch.

The extension of the switchgear room fire detection system enhances safe plant operation and shutdown. The installation of the new detector decreases the probability and consequences of a fire as evaluated in the UFSAR fire hazards analysis. The cable enclosure does not contain any equipment other than cable trays and conduit. Therefore, the only type of accident that could occur in this room is a cable fault fire. The new detector and its associated conduits are supported in accordance with seismic II/I criteria.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0025 UFSAR Revision No. 5

Reference Document: EDP 12080 Section(s) 10.4

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Main Condenser Retube

SUMMARY:

This evaluation justifies revising the description of the main condenser in the UFSAR due to the retubing modification performed during the second refueling outage. The original admiralty brass and 70/30 copper/nickel tubes were replaced with titanium tubes. The original tubes experienced stress corrosion cracking, tube pitting, and were a significant contributor to the amount of dissolved copper in the feedwater system. The interaction of copper with the zircaloy fuel rod cladding results in crud induced localized corrosion (CILC). The corrosion can occur when the reactor is operated above 85% power.

The main condenser is not required for safe shutdown of the reactor or operation of emergency safety features equipment. The new tube material does not change the function of the condenser. Removal of the original condenser tubes removes a major source of copper and is expected to reduce the possibility of CILC. Reduction of CILC mitigates the potential for fuel failure caused by corrosion of the fuel rod zircaloy cladding. The titanium tubes do not react with the zircaloy cladding. The effects of condenser uplift, tube vibration, and tube sheet stress were analyzed and the results were found to be acceptable.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0027 UFSAR Revision No. 5

Reference Document: EDP 12122 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Main Steam Safety Relief Valve (MSSRV) Solenoid Valve Modifications

SUMMARY:

This modification changed the solenoid arrangement for MSSRVs B2104F013A through H, J through N, P, and R from a double to a single solenoid valve arrangement. The second solenoid valve was non-functional and did not affect the operation of the MSSRVs. It was originally added to accommodate the proposed General Electric Prompt Relief Trip System (PRT). The PRT system was never implemented at Fermi 2. Removal of the spare solenoid valves is a maintenance enhancement in that it eliminates unnecessary EQ refurbishment costs associated with each spare, non-functional solenoid valve.

The MSSRVs continue to function and operate as originally designed and as described in the UFSAR. This modification does not affect the qualification of the MSSRVs because the solenoid valves are qualified in both the single and dual solenoid configurations.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0033	UFSAR Revision No. 5
Reference Document: EDP 12175	Section(s) N/A
	Table(s) N/A
	Figure Change [X] Yes [ ] No

Title of Change: Alternate Drywell Head Storage Location on UFSAR Figure

SUMMARY:

This evaluation justifies revising UFSAR Figure 3.8-31, Sheet 1 to illustrate an alternate drywell head storage location over the dryer/separator storage pool. The support assemblies, consisting of beams and plates located in four places at the edge of the storage pool, are shown in Detroit Edison drawing 6C721-2348, revision U.

Storing the drywell head at a location other than the designated laydown area on the refueling floor does not involve any equipment or affect the performance of any system. The function of the dryer/separator and dryer/separator storage pool are unaffected by storing the drywell head over the pool. Seismic analysis shows that the support assemblies will not move and are capable of handling drywell head loads during a design basis seismic event. Compliance with NUREG-0612 ensures that a load handling accident will not occur.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0034 UFSAR Revision No. 5

Reference Document: EDP 11983 Section(s) 11.3

Table(s) 3.2-1; 11.3-4

Figure Change [X] Yes [ ] No

Title of Change: Ring Water Vacuum Pump N6200C003 Changeout

SUMMARY:

This modification provided two new ring water vacuum pumps. One pump replaced the original north ring water vacuum pump. The original north pump will become a spare for the south ring water pump. The second new ring water pump is a spare for the new north ring water pump. The original Nash model H-4N is obsolete and no longer available. The replacement pump is a Nash model H-4. The new pump's performance characteristics are the same as those of the original pump. However, there are several minor differences between the two models:

1. The original pump design, fabrication, and material conformed to ASME III, class 3. The new pump meets the manufacturer's standards only. This is acceptable per Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light Water Reactors".

2. The new pump material is cast iron whereas the original pump material is stainless steel. Cast iron is acceptable for handling the demineralized water and gasses that the pump is exposed to.

3. The suction and discharge flange size of the new pump is smaller. However, the system design pressure is less than the design pressure of the replacement pump and is therefore acceptable.

4. The full load current is somewhat higher than the original pump. However, the existing cable and motor starter ratings are sufficient to handle the higher current.

Replacement of the north ring water pump does not change the function, configuration, or operation of the offgas system. System reliability is maintained with the new ring water pump.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0038 UFSAR Revision No. N/A

Reference Document: EDP 8127 Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Personnel Airlock Handwheel Shaft Seal and Equalizing Valve Replacement

SUMMARY:

This modification replaced the personnel airlock handwheel teflon shaft seals with graphite seals. In addition, the 3" equalizing valves were replaced to eliminate the teflon seats and valve stem seals. The new 2" equalizing valves use Nordel EPDM seats and stem seals. The environmentally qualified graphite and Nordel components, having greater radiation resistance, are required to ensure containment integrity during long term design basis accident radiation exposure in combination with calculated normal plant exposure levels.

The difference in frictional characteristics between the materials is within the design characteristics of the handwheel linkage assembly. The reduction in flow area due to the difference in the replacement valves' size does not affect the pressure equalization performance of the airlock. Therefore, the change in materials does not alter the function or operation of the personnel airlock and containment integrity is assured.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0039 UFSAR Revision No. 5

Reference Document: EDP 7121 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Permanent Installation of Condenser Sample Line

SUMMARY:

This modification provided for the permanent installation of a sample line connecting the condenser west outlet tap CT-N71-L005B to sample sink #11. It replaced the tygon tubing installed by temporary modification 88-0055 with stainless steel tubing. The temporary modification was installed to ensure that representative circulating water samples would be obtained.

Installation of the permanent sample line ensures an accurate sample. Accurate water samples ensure that the proper water treatment chemical dose rate is determined resulting in the biological organism control and the prevention of excessive chemical induced corrosion. Replacement of the tygon tubing with stainless steel tubing allows the sample line to be returned to its original design basis. The sample line routing, stress analysis, support design, and fabrication conform to the requirements of Detroit Edison Specifications and ANSI B31.1.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0044	UFSAR Revision No. 5
Reference Document: EDP 12234	Section(s) N/A
	Table(s) N/A
	Figure Change [X] Yes [ ] No

Title of Change: Seal Water Return Tank Sparger Installation and Vent Line Modification

SUMMARY:

The purpose of this modification is to prevent turbulence and tank pressure induced air carry-over from the seal water return tank to the condenser. This modification installed spargers in the seal return tank and increases the seal return tank vent size from 1/2" to 2". The spargers decrease the turbulence by dispersing the drain flow and the increased vent line size eliminates tank pressurization.

This modification enhances the operation of the seal return tank. It does not change the function of the seal return tank and does not interface with or challenge any safety related equipment or systems. This modification has been prepared in accordance with ANSI B31.1.0.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0047 UFSAR Revision No. 5

Reference Document: EDP 12199 Section(s) N/A

Table(s) 6.2-2; 6.2-13; 6.2-15

Figure Change [X] Yes [ ] No

Title of Change: Division I Core Spray Minimum Flow/Recirculation Isolation Valve E2150F031A Replacement

SUMMARY:

This modification replaced the Division I core spray minimum flow/recirculation isolation valve E2150F031A. The existing Limitorque operator was reinstalled on the new valve. A body test tap was also installed as a field modification to allow Appendix J LLRT's. The replacement valve is the same make and model as the original valve.

The replacement valve is identical to the original valve in terms of seismic qualification; code design and manufacture; materials of construction; actuation mode; and performance characteristics. This modification does not change the performance and operation of the core spray Division I equipment and does not change the configuration of the plant.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0051 REV 2   UFSAR Revision No. 5

Reference Document: EDP 12331   Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Station Air and Interruptible Control Air Header Connections for the Future Relocation of the Instrument Air System (IAS) Dryers and Receiver Tank

SUMMARY:

This modification added: (1) a 3" connection, valve, and cap on the 8" header downstream of the receiver tanks located on the first floor of the turbine building and (2) a 3" connection, valve, and cap on the IAS supply to the Residual Heat Removal (RHR) complex downstream of the IAS isolation valve, P5000F360. These air system tie-ins allow the on-line relocation of the IAS dryers and receiver tank to make room for the installation of an additional demineralizer. The demineralizer is scheduled to be installed during the third refueling outage.

This modification does not impact any component or system related to the safe shutdown of the reactor. The connections are similar to existing system components and are installed to B31.1 requirements.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0057 UFSAR Revision No. 5

Reference Document: EDP 12300 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Heater Drain Pump Seal Water Piping Pressure Breakdown Orifice Installation

SUMMARY:

This modification installed pressure breakdown orifices in the heater drain pumps seal water inlet piping. Three orifices are installed in series to minimize cavitation and noise as the pressure is reduced. The orifices are sized to maintain adequate (2gpm) seal flow. This modification reduces the heater drain pump mechanical seal cooling water pressure from 700 psi to the seal manufacturer's recommended pressure of 250 psi. This pressure reduction is expected to increase seal life and reduce the probability of seal failure.

The heater drain pumps are not required for safe shutdown, accident mitigation, or for recovery after an accident. The addition of the orifices does not adversely affect the loss of feedwater accident analyses in the UFSAR.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0059 UFSAR Revision No. 5

Reference Document: EDP 12301 Section(s) N/A

Table(s) 6.2-2; 6.2-15

Figure Change [X] Yes [ ] No

Title of Change: Torus Water Management System (TWMS) Outboard Containment Isolation Valve G5100F605 Replacement

SUMMARY:

This evaluation justifies the replacement of the TWMS outboard isolation valve G5100F605 with a new valve. The valve was replaced because it could not meet the refurbishment requirements for local leak rate testing seat leakage integrity. The new valve is the same make and model as the former valve. The differences between the two valves are material specifications and wedge type. The new valve was forged and the old valve was cast. Some of the valve components in the new valve are made of different materials. The material substitutions meet Detroit Edison Fermi 2 specifications. The wedge for the new valve is flexible whereas the original valve's wedge was solid. The flexible wedge represents an improvement and has no adverse affect on the performance of the valve.

This modification does not change the design bases, function, or operation of the TWMS. It does not have any affect on the operation or reliability of interfacing plant systems. The new valve stroke time is slightly longer the original valve stroke time. However, it is still within the existing stroke time design range.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0064 UFSAR Revision No. 5

Reference Document: EDP 9188 Section(s) 9A.5.f.17; 10.4

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: North Circulating Water Cooling Tower Modifications

SUMMARY:

This modification incorporated miscellaneous modifications to the north circulating water cooling tower to restore it to its original condition and provide the required circulating water flow and temperature measurement points. This safety evaluation only addresses the modifications which affect the UFSAR. These changes include the installation of pitot tube isolation valves, installation of temperature indicators, and replacement of the existing fill with combustible fill.

The circulating water system is not required for the safe shutdown the plant. The new components do not interact with or support safety related equipment. The installation of combustible fill does not invalidate the fire protection analysis because the circulating water cooling towers are located where a fire cannot affect any safety related equipment and the cooling tower basins are not used as a water supply for the ultimate heat sink or the fire protection system. Installation of the pitot tube isolation valves and temperature indicators have no impact on upon existing accident scenarios contained in the UFSAR.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0073 UFSAR Revision No. 5

Reference Document: EDP 12428 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Installation of Fuel Pool Cooling and Cleanup System (FPCCS) Hydrolasing Connection and Drain Line

SUMMARY:

This modification added a hydrolasing connection and a drain line to the reactor well or dryer/separator storage pit to waste surge tank drain line in the FPCCS. A low spot in this piping located in the main hallway of the second floor of the reactor building has had very high radiation levels (300-600 mr/hr) due to plate out. This has caused the Division II emergency equipment service water and residual heat removal radiation monitors to stay in the high alarm mode. The hydrolasing connections and drain line have been installed to clean the piping section so that radiation levels can be kept to a minimum. This modification also downgrades the modified piping section from Class C to Class D piping. The turbine building loop seal was filled with water to maintain secondary containment integrity during the implementation of this modification. This loop seal will also be maintained during subsequent hydrolasing evolutions.

This modification does not adversely affect the FPCCS or any other plant system. When not in use, the hydrolasing connection will be blind flanged and the drain valve will remain closed and capped. This modification does not affect secondary containment leakage and does not affect the ability of the standby gas treatment system to establish a negative pressure of 0.25" wg within 10 minutes after a LOCA. Administrative controls are in place to ensure that turbine building loop seal is flooded to provide secondary containment integrity when hydrolasing is in progress. The piping downgrade is in accordance with Regulatory Guide 1.26.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0110 UFSAR Revision No. 5

Reference Document: EDP 12749 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Condensate Pressure Transmitter Installation

SUMMARY:

This change installed a quick response pressure transmitter on the condensate side of the north number 3, 4, and 5 feedwater heaters. This transmitter is used in conjunction with General Electric Transient Analysis Recording System (GETARS) to record the transient condensate pressure response during reactor scrams. The pressure data will be used to investigate the cause of condensate thermal relief valve lifting problems experienced during reactor scrams. The response of the normal condensate pressure transmitter is too slow for adequately recording of the pressure transients.

The transmitter and tubing construction; material; and installation is similar to the existing configuration. This modification does not affect safety related plant systems. The condensate system and GETARS are not required to support equipment required for safe shutdown. Failure of the pressure transmitter tubing is less severe than the outside containment feedwater line failure analyzed in the UFSAR.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0111 UFSAR Revision No. 5

Reference Document: EDP 12778 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Relocation of Heater Drain Pump Discharge Vent Connection

SUMMARY:

This modification relocated the heater drain pump discharge vent connection for each heater drain pump from the heater drain pump discharge nozzle to the discharge piping. The former threaded connection provided by the pump manufacturer had failed on all three pumps resulting in steam leaks that contaminated the heater drain pump rooms. The new vent lines are connected to their respective heater drain pump discharge line with welded sockolet connections. The new installation reduces stress at the vent/discharge pipe connection resulting in a reduction in the likelihood of future connection failures.

The new installation meets the requirements of the ANSI B31.1 Power Piping Code. It has no impact on the function of the heater drain pump vents, heater drain pumps, instrumentation, or other heater drain pump system components. The new vent connection locations are at the same elevation as the former discharge nozzle vent connection locations. The new locations do not impact the discharge piping stress analysis.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0114 UFSAR Revision No. 5

Reference Document: EDP 12772 Section(s) 8.1; 8.2

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Main Transformer 2A Replacement

SUMMARY:

This change replaced the original Ferranti main unit transformer 2A with an ABB transformer. The original transformer was replaced due to an increase in combustible gas generation within the transformer. This is indicative of transformer oil breakdown due to localized overheating. The mounting configuration of the two transformers is similar. However, the flex shunt bushing was modified to accommodate an approximate two inch gap between the low voltage bushing and the isophase bus.

The replacement transformer has the same MVA and voltage ratio rating. Engineering considerations such as load sharing and circulating current between transformers 2A and 2B; voltage regulation; short circuit current; directional overcurrent relay settings; power feeder rating; power uprate impact; and fire protection deluge system adequacy were reviewed and found acceptable. The operability of various components and circuits associated with the main transformers was reviewed and either accepted or modified to accommodate the change.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0120 UFSAR Revision No. 5

Reference Document: EDP 12785 Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Elimination of Separator Seal Tank North and South Outlet Level Control Valve Bypass Valve Leaks

SUMMARY:

This modification: (1) removed the motor operators on separator seal tank north and south outlet level control valve bypass valves, N2200F687 and N220F686 and (2) welded a plate to the valve neck of each valve to eliminate body to bonnet gasket leakage. Prior to this modification, on-line leak repairs were unsuccessful. Each plate provides a more leakproof boundary and improves the integrity of the system. These valves were previously abandoned by engineering design packages EDP 8938 and EDP 10778.

This modification does not change the function of the heater drains system or any equipment required for safe shutdown or accident mitigation because these valves were already abandoned. The plates were sized using the guidelines of ASME Section VIII, Part UG-34, Equation 1. The plate installations utilized full penetration welds to provide the necessary strength to maintain system integrity.

END OF EDP SECTION

FERMI 2

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SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0049 UFSAR Revision No. 5

Reference Document: LCR 90-074-UFS Section(s) 7.4

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Clarification of Reactor Core Injection Cooling (RCIC) and High Pressure Coolant Injection (HPCI) Suction Transfer Logic

SUMMARY:

This evaluation justifies revising UFSAR subsection 7.4.1.1.3.8 to state that a condensate storage tank (CST) failure which results in a loss of inventory and/or loss of the current signal from either CST level transmitter will cause an automatic RCIC/HPCI suction transfer. The original UFSAR text stated that a complete failure of the CST and/or transmitter system would result in an automatic RCIC/HPCI suction transfer. Design Basis Task Force (DBTF) item E51-026 questioned whether dual upscale failures should be postulated that could disable the RCIC/HPCI suction transfer capability. Dual upscale failures are not credible because each transmitter system is redundant and the postulated failure would have to involve simultaneous failures of both transmitter systems. UFSAR subsection 7.4.1.1.3.8 was clarified to remove any implication that dual upscale failures of the CST level transmitters will result in a RCIC/HPCI suction transfer.

This change clarifies the description of the RCIC/HPCI suction transfer logic. It does not alter the the design, function, or operation of RCIC, HPCI, or the CST transmitters. The Fermi 2 Safety Evaluation Report accurately describes the RCIC/HPCI suction transfer. It does not state that a complete failure of the CST transmitter systems causes a suction transfer.

SAFETY EVALUATION SUMMARY

Safety Evaluation No:	90-0071	UFSAR Revision No.	5
Reference Document:	LCR 90-100-UFS	Section(s)	5.5; 9.1; 15.6
		Table(s)	9.1-1
		Figure Change	[ ] Yes [X] No

Title of Change: Residual Heat Removal (RHR) and Core Spray (CS) Systems UFSAR Description Corrections

SUMMARY:

This evaluation justifies correcting UFSAR discrepancies found by the Design Basis Task Force. These changes are as follows:

1) UFSAR subsection 5.5.7.3.2 has been revised to reflect the RHR system flushing methodology specified in system operating procedure (SOP) 23.205, "Residual Heat Removal System". The UFSAR originally stated that the RHR system was flushed through the minimum flow line and the test line prior to initiating shutdown cooling. SOP 23.205 specifies that flushing is accomplished using the warm up line.

2) The RHR fuel pool cleanup (FPCU) assist flowrate in UFSAR subsection 9.1.3.2 has been changed from the original 5400 gpm to 3500 gpm to reflect the actual flowrate achieved during startup testing and the results of a design calculation.

3) UFSAR subsection 15.6.5.5.1.1 has been revised to state that the RHR FPCU assist piping is seismic category I and that the non-seismic keep-fill piping connects to the core spray piping. This subsection has also been revised to reference UFSAR subsection 3.7.3.13 for the design methodology for non-seismic piping that is connected to safety-related piping. The UFSAR originally stated that the RHR FPCU assist piping is non-seismic and that there is no non-seismic piping connected to the core spray piping.

The plant has not been modified by these changes. The RHR FPCU cooling capacity still exceeds the maximum fuel pool heat load and the seismic design of systems that interface with the CS and RHR systems is in accordance with UFSAR section 3.7.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0073 UFSAR Revision No. 5

Reference Document: LCR 90-103-UFS Section(s) 5.5; 7.4

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Reactor Core Isolation Cooling (RCIC) Suction Transfer Logic Description Correction

SUMMARY:

This evaluation justifies correcting the description of the RCIC suction transfer logic in UFSAR subsections 5.5.6.3.3 and 7.4.1.1.3.8. The description originally stated that the automatic RCIC suction transfer to the torus on low condensate storage tank (CST) level was single failure proof. Design Basis Task Force Item E51-027 questioned the accuracy of this statement and identified components that would prevent RCIC suction transfer if they failed. As a result, UFSAR sections 5.5.6.3.3 and 7.4.1.1.3.8 were revised to correctly state that the automatic RCIC suction transfer is accomplished by utilizing the deenergize to operate trip logic within the Division II HPCI system from redundant analog CST level transmitters and trip units.

This change makes UFSAR subsections 5.5.6.3.3 and 7.4.1.1.3.8 reflect the as-built condition of the plant. RCIC is not part of the emergency core cooling system (ECCS) and is not required to achieve safe shutdown of the reactor. In addition, no credit is taken for RCIC in accident mitigation. Failure of the RCIC suction to align with the torus on CST low level does not affect other equipment involved in the accidents evaluated in the UFSAR.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0081 UFSAR Revision No. 5

Reference Document. DBTF B31-022 thru 025-M

Section(s) 5.5

Table(s) 5.5-1

Figure Change [ ] Yes [X] No

Title of Change: Reactor Recirculation System (RRS) Pump UFSAR Text and Table Revisions

SUMMARY:

This evaluation justifies revising portions of the UFSAR text and tables that deal with the RRS pump seals, seal cooling, motor, and required net positive suction head (NPSH) to address the concerns in Design Basis Task Force items B31-022-M through 025-M. The changes are as follows:

1. The operability life of the RRS pump seals has been changed from "1 year" to "1 operating cycle" in UFSAR subsection 5.5.1.

2. UFSAR subsection 5.5.1 has been clarified to specify that the RRS pump motor is a standard AC induction motor capable of being operated with a power supply of varying frequency over the specified range and that the motor starts when the motor-generator excitation field breaker is closed.

3. UFSAR subsection 5.5.1 has been clarified to state that the reactor building component cooling water system supplies component cooling water to the RRS pumps via the divisional emergency equipment cooling water (EECW) system and that there are no RRS pump trips associated with the loss of RRS component cooling.

4. The RRS pump NPSH has been changed from 115 ft to 135 ft. This change reflects the actual operating point for hot conditions.

These changes do not alter the RRS pump assembly system or components including their design bases criteria, function, operation, or control. There is no change to the analyzed or postulated failure modes or effects for the RRS pump assembly and its associated support systems and facilities.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0089 REV 1 UFSAR Revision No. 5

Reference Document: LCR 90-005-UFS Section(s) 3.5; 10.2

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: UFSAR Revisions Concerning Turbine Missile Barriers

SUMMARY:

This safety evaluation was written to address changes to the UFSAR concerning turbine missile barriers. During the development of a proposed licensing amendment to delete the turbine overspeed technical specification, inconsistencies were found in the turbine missile barrier support documentation for several UFSAR sections. These inconsistencies were evaluated in a design calculation. The design calculation confirms that the original design level of conservatism as currently stated in the UFSAR is maintained.

These changes to the UFSAR do not introduce a new mode of plant operation or involve a physical modification to the plant. No technical specifications, assumptions, safety limits or limiting safety system setpoints are affected or changed.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0103 UFSAR Revision No. 5

Reference Document: LCR 90-135-UFS Section(s) 4.3; 10.4; 15.17; B15.5
Table(s) B15.1-1

Figure Change [ ] Yes [X] No

Title of Change: Revision to UFSAR Discussion of Core Power Distribution and Cold Water Injection Effects

SUMMARY:

This evaluation justifies the following:

1. UFSAR subsection 4.3.2.5 has been revised to eliminate the statement that the core power distribution goal is that of a Haling shape and remove the implication that adjustments are needed when control rods become inoperable. The core power distribution is constrained by average planar linear heat generation rate, linear heat generation rate, and critical power ratio. The control rod patterns are adjusted whenever necessary to ensure conformance to the target exposure distribution and to thermal limits for a variety of reasons including but not limited to the presence of inoperable control rods.

2. UFSAR subsection B15.5.1 has been corrected to state that the cold water injection from an inadvertent high pressure coolant injection (HPCI) event results in a decrease in inlet enthalpy and a consequent increase in power. This original subsection incorrectly stated that the above scenario resulted in a decrease in inlet subcooling.

The changes in UFSAR subsection 4.3.2.5 do not result in changes to the facility or method of plant operation. Conservative bounds on the amount of axial power distribution and restrictions on the manner by which the axial power distribution may vary are imposed. This distribution is monitored throughout the cycle within a pre-established licensing range. By staying within these limits, the consequences of any accident in UFSAR Chapter 15 are not increased.

The correction to UFSAR subsection B15.5.1 is an administrative clarification that only corrects the confusion between enthalpy and subcooling. It does not impact the function or operation of HPCI, has no effect on plant operation, and cannot affect the probability or consequences of any radiological release.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0144 UFSAR Revision No. 5

Reference Document: LCR 90-208-UFS Section(s) 9.1

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: New Fuel Assembly Transfer to the Spent Fuel Pool (SFP) Storage Racks Using the Fuel Preparation Machine (FPM)

SUMMARY:

The purpose of this safety evaluation is to justify using the FPM to place new fuel assemblies in the SFP storage racks, correcting the description of FPM operation in UFSAR sections 9.1.2.2.1 and 9.1.4.2.3, and revising UFSAR figure 9.1-8. The FPM is being used to lower the new fuel assemblies in the SFP storage racks to avoid contaminating the crane. This is accomplished by setting the FPM full up end stop set so that when a fuel assembly is placed in the FPM the bail handle is just above the surface of the SFP. This ensures that when a fuel assembly is placed in the FPM the crane hook will not get wet. UFSAR sections 9.1.2.2.1 and 9.1.4.2.3 have been revised to eliminate reference to the use of used fuel channels (NRC commitment per DECO letter #NRC-90-0078, dated 4/25/90) and to state that the FPM can be used for new fuel receipt/transfer activities. UFSAR figure 9.1-8 has been replaced with a simplified diagram of the FPM.

This change does not alter the fuel handling accident analysis as the fuel movement addressed in this safety evaluation is constrained to the SFP and the fuel assemblies are not irradiated. The consequences of dropping a fuel assembly onto a SFP storage rack during pool transfer were evaluated. This change does not increase the likelihood of SFP storage rack damage because established administrative procedures require that the fuel assemblies be placed in and taken out of the SFP in areas which do not contain fuel storage racks. The FPMs are not used to mitigate the effects of any accidents or specifically postulated to cause any accidents evaluated in the UFSAR. The possibility of inadvertent personnel radiation exposures by placing an irradiated fuel assembly in the FPM with the full up end stop set to the new fuel receipt/transfer level (lack of adequate shielding) is prevented by administrative controls.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0148 UFSAR Revision No. 5

Reference Document: LCR 90-215-UFS Section(s) 8.3

Table(s) 8.3-1

Figure Change [ ] Yes [X] No

Title of Change: Deletion of UFSAR Cable Ampacity Table 8.3-1

SUMMARY:

The evaluation justifies: (1) Deleting UFSAR Table 8.3-1 which defines the ampacity for cables used at Fermi 2 and references to the table in UFSAR subsections 8.3.1.1.1 and 8.3.1.4.2.1 and, (2) Revising UFSAR subsection 8.3.1.4.2.1 to state that cable ampacities are controlled by design instruction. Table 8.3-1 is limited in its applicability in that it does not contain all cable sizes and types used at Fermi 2 and it contains no provisions for conditions that deviate from those upon which the table is based. The design instruction consolidates all ampacity calculations previously utilized at Fermi 2 including those from which UFSAR Table 8.3-1 was derived.

Both UFSAR Table 8.3-1 and the design instruction are based on the same criteria and industry standards. This revision does not change the plant. Comparison of the ampacity values in the design instruction and UFSAR Table 8.3-1 verify that no existing cables are affected by these changes. Proper application of this design instruction precludes cable failure, one of the potential causes of a single electrical component failure, from occurring as a result of cable overheating.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0150 REV 1 UFSAR Revision No. 6

Reference Document: LCR 90-219-UFS Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Control Center Pressure Boundary Shield Block and Plank Wall Modifications

SUMMARY:

This evaluation justifies revising UFSAR figure 9.1-3, sheet 1 to reflect the changes made by engineering design package EDP-10125. This change made structural modifications to three removable shield block and plank walls on the fifth floor and one shield block and plank wall on the third floor of the auxiliary building. This modification upgraded the walls from seismic category II/I to category I. The modification involved the addition of a continuous one eighth inch steel cover plate over the walls in the standby gas treatment system rooms. The other shield plank walls were structurally reinforced with steel plate and angles.

These walls are structurally superior to the original design. This modification complies with seismic category I design and implementing criteria. This ensures control center pressure integrity in that the barriers are structurally stable and capable of performing their safety function during a seismic event. The modification to the shield walls does not change anything associated with the reactor building refueling floor or its activities as discussed in UFSAR section 9.1.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 90-0154 UFSAR Revision No. 5

Reference Document: COLR 3.0 Section(s) B.4; B.5; B.6; B.15

Table(s) B.4.3-1; B.5.1-1; B.6.1-1; B.15.0-1; B.15.0-2; B.15.0-3; B.15.1.2-1; B.15.1.2-2; B.15.1.2-3; B.15.1.2-4; B.15.1.2-5; B.15.1.3-1; B.15.1.3-2; B 15.1.3-3; B.15.1.3-4

Figure Change [ ] Yes [X] No

Title of Change: Revision to Core Operating Limits Report

SUMMARY:

This evaluation justifies changes to the Core Operating Limits Report (COLR). These changes include:

1. Changing the methodologies for calculating thermal limits as a result of acquiring average power range monitor and rod block monitor technical specifications (ARTS) and Maximum Extended Operating Domain (MEOD). The maximum average linear heat generation rate (MAPLHGR) and minimum critical power ratio (MCPR) have become a function of reactor power and core flow. As a result of MEOD, the rod block monitor (RBM) trip setpoints and allowable values have become cycle specific and have been added to COLR-3.0.

2. Adding the thermal limits for the new fuel types used in cycle 3. 228 GE6 fuel bundles were replaced with 224 GE9B fuel bundles and 4 SVEA-96 lead fuel assemblies (LFA). MAPLHGR and and linear heat generation rate (LHGR) limits for the fuel bundles loaded for the first and second cycles remain the same.

3. Using 80 mil fuel channels on the new GE9B fuel bundles. The fuel bundles loaded during the first and second fuel cycles were enclosed in 100 mil fuel channels.

The addition of the MAPLHGR, LGHR, and operating MCPR limits to COLR does not change the method of plant operation or result in any modifications to the facility.

Safety Evaluation No. 90-0154 (continued):

Use of the GE9B fuel design has been generically approved by the NRC. Its application at Fermi 2 has been analyzed using the approved methodologies in GESTAR II.

The performance and function of the SVEA-96 LFAs are similar to that of the GE9B reference bundle. They comply with the criteria in GESTAR II. Use of the LFAs does not change the operation of the plant or require modifications to equipment important to safety. The LFAs are geometrically compatible with existing fuel assemblies, control rods, neutron instrumentation, spent fuel storage racks, and fuel handling equipment. The GE cycle 3 reload analysis using the GE9B reference bundle remains valid because: (1) the shutdown margin is slightly improved with the SVEA-96 LFAs; (2) the peak cladding temperatures are 50 to 150 degrees F lower than an 8x8 bundle operating at the same LHGR and maximum oxidation rates are far below the 17% design limit; and (3) the differences in void coefficient, fuel rod thermal time constant, and scram reactivity will not affect the overpressurization transient because of the small number of LFAs in the core.

The purpose and performance of the 80 mil channels are identical to that of the 100 mil channels. The 80 mil channels do not require any modification to equipment important to safety and do not change the method of operation. The 80 mil channels are geometrically compatible with existing fuel assemblies, control rods, neutron instrumentation, spent fuel storage racks, and fuel handling equipment.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0001 UFSAR Revision No. 5

Reference Document: LCR 91-001-UFS Section(s) 9.1

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Fuel Assembly Drop Height Over the Spent Fuel Pool Revision to UFSAR

SUMMARY:

This revision changes the fuel assembly drop height in UFSAR section 9.1.2.1.1 for a drop in the spent fuel pool from 6 ft 6 in. to 17 ft 7 in. Detroit Edison re-evaluated the integrity of the spent fuel pool liner for a postulated fuel assembly drop of 17 ft 7 in.; the maximum height currently permitted by the refueling bridge. The results of the evaluation indicate that the spent fuel pool liner is capable of sustaining a fuel assembly drop from a height of 17 ft 7 in. without penetrating the liner. These results are documented in a design calculation.

No new design is required and no modifications were made as a result of this change. The spent fuel pool fuel assembly drop accident discussed in UFSAR Chapter 9 is unaffected by this change because the spent fuel pool liner integrity is still maintained. The radiological consequences of the spent fuel pool fuel assembly drop are not affected because they are bounded by the fuel assembly drop over reactor core accident.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0005 UFSAR Revision No. 5

Reference Document: LCR 91-010-UFS Section(s) 9A.6

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Clarification of Surveillance Requirements for CO_2 Fire Protection System Valve Verification

SUMMARY:

This evaluation justifies adding a clarification note to UFSAR section 9A.6.4.2 which states that the valve lineup for the CO_2 fire protection system is verified by verifying that the CO_2 tank level and pressure are within the acceptance criteria of the applicable surveillance tests. This methodology is required because the position of these valves cannot be determined by visual inspection. This clarification was previously contained in Technical Specification Clarification TSC 89-022. However, the fire protection requirements have been removed from the technical specifications and placed in the UFSAR. As a result, the clarification has been incorporated into the UFSAR.

This change is a document clarification and no physical changes were made to any component or system in the plant. This clarification does not impact any surveillance or operating procedures.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0012

UFSAR Revision No. 5

Reference Documents: 67.000.501
78.000.69

Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Alternate Method of Grab Sampling the Primary Containment Atmosphere

SUMMARY:

This evaluation justifies the use of an alternate method for grab sampling the primary containment atmosphere. The alternate grab sampling scheme draws a sample from the drywell through the Division I H2/O2 cabinet H21P282 sample line into a manual sample rig. The sample is returned through the Division I H2/O2 Cabinet H21P282 sample return line. This sampling method violates containment integrity when samples are being taken because the sample suction containment isolation valves are remote manual isolation valves and do not automatically isolate on a containment isolation. In addition, the manual sample rig is not expected to withstand the theorized primary containment pressures and temperatures that would exist following a LOCA. The normal method of sampling the primary containment does not violate primary containment integrity because the sample taps are located outside the primary containment boundary and isolated by the primary containment radiation monitoring skid (PCRMS) automatic isolation valves. If one of the four PCRMS automatic isolation valves has to be deenergized closed or fails closed, the normal sampling method introduces the risk of not meeting the required sampling frequency for drywell purge and venting. The inability to draw grab samples could require a plant shutdown if the PCRMS automatic isolation valves are not restored to operability within the Technical Specification time limits.

In order to meet the functional requirements of primary containment, the alternate method of grab sampling requires the following administrative controls:

1. The leakage area of the alternate sampling taps has been reduced by installing 0.25 in. diameter fittings.

2. A method of providing primary containment isolation within the normal containment isolation time frame is used while alternate grab samples are being taken. This method dedicates personnel to monitoring control center instrumentation for LOCA conditions and to deenergizing the power supply to the alternate sampling remote manual valves when LOCA conditions are detected.

Safety Evaluation No. 91-0012 (continued):

3. The alternate grab sample taps are controlled as locked valves.

4. The grab sample tap locations were selected such that their primary containment isolation valves are the same model as the PCRMS automatic isolation valves.

There is no loss in the level of protection provided by any of the fission product barriers. The administrative controls for the alternate sampling method provide the capability of isolating the sample lines within the 60 second primary containment isolation time. An analysis of the frequency of use and the probability that the administrative controls would be used concluded that the probability of a LOCA occurring while alternate sampling is in progress is below that of normal regulatory concern. Therefore, the probability of UFSAR/SER analyzed accidents and transients is unaffected.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0015 REV 1 UFSAR Revision No. N/A

Reference Document: DER 88-2071 Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Temporary Storage of Mixed Waste in the Onsite Storage Facility (OSSF)

SUMMARY:

The purpose of this evaluation is to justify the temporary storage of mixed waste at the OSSF. The mixed waste consists of freon (2% by volume) contaminated waste body oil generated as a result of dry cleaning laundry operations. The mixed waste is stored in DOT-17E 55 gallon steel drums. The storage area is located in the northeast section of the OSSF and occupies an area 15 ft by 30 ft. A containment berm is installed in the storage area to prevent mixed waste from entering the floor drain system in the event of a storage drum rupture. Mixed waste is being stored at the OSSF because it is currently not accepted at low level radioactive disposal sites or hazardous waste disposal facilities. This is a temporary condition that will last until regulations allow the waste to be processed into separate hazardous and radioactive constituents.

The mixed waste is non-explosive and essentially non-volatile. Siting the mixed waste storage facility in the OSSF will result in radiation exposure consistent with ALARA guidelines. The isotopic concentrations average less than 10% MPC and dose rates on the outside of the drums are essentially nonexistent. Storage of mixed waste at the OSSF will not increase the radioactivity inventory of the radwaste system. The OSSF is designed to prevent the release of radioactive materials to the environment and meets the requirements of the Resource Conservation and Recovery Act contained in 40CFR265 for the hazardous waste components of the mixed waste. The storage drums are designed to Department of Transportation DOT 17E specifications. Therefore, no severe drum damage due to a seismic event or fall should occur. Any release of mixed waste within the OSSF will be contained by the berm. The applicable portions of Regulatory guides 8.8 and 1.143; NUREG-0800; IE Circular 80-18; 10CFR20.206; 10CFR50.34A; and 10CFR50 Criteria 60, 63, and 64 were examined and the storage of mixed waste in the OSSF was found to be acceptable.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0018 UFSAR Revision No. 5

Reference Document: LCR 91-051-UFS Section(s) 9A.4.7

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Division I and II Underground Safety Related Ducts Fire Hazards Analysis

SUMMARY:

This evaluation justifies: (1) a revision to the UFSAR fire hazard analysis of equipment and structures in the yard area to address the Division I and II underground safety related cable ducts between the residual heat removal (RHR) complex and the auxiliary building (AB) cable vault; and (2) corrective action for a QA audit finding generated during the September, 1990 fire protection triennial audit. The QA audit finding indicated that flammable liquids (from a tank truck oil spill fire) may pose a threat to the Division I and II safety related cable ducts running between the RHR complex and the AB. The concern was that the burning liquid would enter both division manhole covers and damage the safe shutdown cables in both divisions. The cable ducts contain safe shutdown cables for the RHR service water system, emergency equipment service water system, and the emergency diesel generators. At the time of the finding, one manhole was completely covered with soil and gravel. It was determined that the as-found soil and gravel cover and the physical separation of the division cable ducts had provided adequate protection to ensure that one division was available for safe shutdown during and after a fire. The audit finding corrective action included: (1) covering the manhole covers with soil and gravel to provide a physical barrier against the flammable liquid and heat; (2) erecting a rope barrier and signs to control the storage of combustibles on or between the manholes; and (3) inspecting annually to assure that the rope barrier and signs are in place.

The fire hazards analysis used the guidance included in the NRC's response to question 3.1.4 of Generic letter 86-10. The analysis concluded that at least one safe shutdown train remains available during and after a fire. The soil and gravel overlay and barrier are separated from the safe shutdown circuits. The addition of soil and gravel over the manhole covers reduces the amount of fire damage to the safe shutdown cables by functioning as a flame arrester. The probability of a fire within the area of concern is significantly reduced because storage is not permitted in the area. The corrective actions have an insignificant effect on seismic, flooding, and tornado design.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0019 UFSAR Revision No. 5

Reference Document: 82.000.18 Section(s) 9.1

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Additional Control Rod Storage Rod Storage in the Spent Fuel Pool (SFP)

SUMMARY:

This evaluation justifies the addition of 40 control rod curb hangers in the SFP to provide enough storage for the control rods replaced during the second refueling outage. The hangers are designed to be mounted on the four inch high curb around the SFP. Each hanger has two lifting/storage rods of different lengths and each lifting/storage rod supports one control rod. The staggered hanger rods maximize the use of the available space in the SFP. The lifting/storage rods have a welded tab for position indication purposes. With the reactor building crane auxiliary hoist in its full up position, the tab is set to the top of the SFP curb using a chainfall. A lock is attached to the chainfall to prohibit further upward travel of an engaged control rod. This ensures that the control rod is maintained at greater than 6 ft 6 in. below the refueling platform tracks as required by Technical Specifications.

The curb hangers store the control rods at a lower elevation than the highest installed wall hooks. Seismic and structural load analyses indicate that a curb hanger with a control rod attached will withstand a design basis earthquake. The fuel handling accident analysis does not address a control rod drop in the spent fuel pool. However, the impact energy from a postulated control rod drop is less than that analyzed for a fuel assembly drop. Administrative controls and hoist blocks ensure that the 6 ft 6 in. minimum height restriction of Technical Specification 3.9.6 is met. The load limit for crane travel over the SFP stated in Technical Specification 3.9.7 is 1100 lbs greater than the combined weight of a control rod, curb hanger, and chainfall.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0026   UFSAR Revision No. N/A

Reference Document: SOE 91-01   Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Simulation of Drywell Gas Sampling Techniques Using Sequence of Events Procedure SOE 91-01

SUMMARY:

This evaluation supported the SOE 91-01 procedure for mass spectrometer testing of the primary containment atmosphere (PCA) grab sampling technique used by radiation protection from 9/88 to 9/90. It was believed that this sampling technique introduced air from the reactor building atmosphere into the sampling system which diluted the samples. It was also believed that a back-flow condition caused by the inleakage stopped flow through the oxygen and hydrogen sensors. As a consequence, it was thought possible that the primary containment monitoring system (PCMS) and, possibly, the primary containment radiation monitoring system (PCRMS) were inoperable during the duration of the sampling interval. SOE 91-01 simulated the sampling technique and traced the air inleakage flowpath through the sampling system. The test data was used to determine PCMS operability, PCRMS operability, and whether representative sampling of the PCA prior to venting or purging the drywell was occurring.

The performance of this test was limited to operational conditions 4 and 5. In operational conditions 4 and 5 the PCMS and PCRMS are not required for accidents previously evaluated in the UFSAR. This test made no structural design changes to the PCMS or PCRMS as described in the UFSAR. Helium was used as the tracer gas. Its concentration in primary containment was negligible and it is not corrosive to primary containment or to the equipment within containment.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0030 UFSAR Revision No. 5

Reference Document: LCR 90-068-UFS Section(s) 6.3

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Clarification of UFSAR Description for Emergency Core Cooling System (ECCS) Discharge Line Fill System Low Pressure Alarms

SUMMARY:

UFSAR section 6.3.2.2.5 has been clarified to specify that fill system low pressure alarms exist for the low pressure coolant injection (LPCI) and core spray (CS) systems. The original description stated that the high pressure coolant injection system (HPCI) also had a fill system low pressure alarm. The HPCI system discharge piping is kept full by the head of water from the condensate storage tank (CST) and relies on the CST low level alarms for warning of an inadequately filled discharge line.

This change does not create any physical modifications to the plant and does not change any operating procedures. The function and operation of the ECCS discharge line fill system is not changed by this clarification. Operability of the HPCI system is assured by the physical design of the HPCI fill system in which water and pressure is supplied by the head of the CST. The requirements provided in the basis for the Technical Specifications, UFSAR, and Safety Evaluation Report are not affected by this change.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0031 UFSAR Revision No. N/A

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

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Processing Chemical Waste Tank (CWT) Contents Through the Condensate Phase Separators (CPS)

SUMMARY:

This evaluation justifies processing the contents of the CWT in the CPSs. This method consists of transferring 2000 gallons or less of neutralized chemical waste to a CPS containing 100 cubic feet or more of partially expended ion exchange resins. The phase separator contents is circulated for one hour minimum. Contact between the resins and chemical waste results in the removal of ionic and organic impurities. The supernatant contains less than detectable radionuclides and a significant reduction in total organic carbon (TOC) is achieved. Normally, the contents of the CWT are processed by the following methods:

1. The CWT contents are transferred to the floor drain collector tank (FDCT) and subsequently processed through the floor drain filter (FDF), evaporator feed surge tank (EFST), and floor drain and waste demineralizers (FDD & WD) for release from the waste sample tanks (WST).

2. The CWT contents are processed through a portable activated carbon bed and mixed bed demineralizer. The effluent is transferred to the FDCT and processed as described above.

3. The CWT contents is transferred to the FDCT and processed through the waste collector filter for release from the WSTs.

Method 1 creates additional solid radwaste and an unacceptable TOC in the WSTs; method 2 is cost prohibitive; and method 3 is contrary to the goals of the Fermi 2 zero discharge program. Processing the CWT contents in the CPS supports the zero discharge and solid radwaste reduction programs.

Safety Evaluation No. 91-0031 (continued):

No equipment important to safety is used to process chemical waste and no equipment important to safety is located in the radwaste building. The chemical waste has no effect on the radwaste system components because it is neutralized prior to transfer to the CPSs. Since the chemical waste is processed in the radwaste system prior to release to the WSTs, there is no effect on releases to unrestricted areas. This method does not affect the Fermi 2 process control program. The calculated offsite doses that would result due to the use of this process and as a result of a hypothetical radwaste accident are greater than those previously calculated in UFSAR section 15.7.3 but are still much less than the limits in Appendix B of 10 CFR20.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0035 UFSAR Revision No. 5

Reference Document: LCR 91-079-UFS Section(s) 7.1; 7.6

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Revising the T50 Primary Containment Radiation Monitoring System (PCRMS) Functional Description in UFSAR Sections 7.1 and 7.6

SUMMARY:

This evaluation justifies revising UFSAR subsections 7.1.2.1.22.1(A) and 7.6.1.12.1.6 to make the T50 PCRMS functional description consistent with UFSAR Table 6.2-16; subsections 7.3.2.2.7.1 and 7.3.2.2.7.6; and Technical Specification 3.4.3.1. The new wording eliminates any discussion that states that the T50 PCRMS is required: (1) when the reactor is shut down (i.e., cold shutdown or refueling), (2) when personnel enter containment, or (3) when the standby gas treatment system (SGTS) operation is required. This revision clearly states that the T50 PCRMS functions as part of the leakage detection system (LDS) during operational conditions 1,2, and 3 only.

This revision does not change the function of the T50 PCRMS as it is currently utilized at Fermi 2. The criticality radiation monitors fulfill the criteria of NUREG 8.12 (UFSAR Appendix A.8.12) for a refueling dropped rod accident. In case of an accident, these monitors are supplemented by the control center normal makeup air radiation monitor and the control center direct radiation monitor to fulfill General Design Criteria 13. The Fermi 2 Radiation Protection Program as established in UFSAR Section 12.5 fulfills the radiation protection requirements set forth in 10CFR 19 and 20; the applicable regulatory guides; and the Technical Specifications. Administrative controls, portable radiation monitoring instrumentation, and the area radiation monitoring system ensure that personnel exposure is kept as low as reasonably achievable (ALARA). Therefore, this change does not increase the the onsite or offsite dose received by personnel following an accident. The PCRMS is not qualified to operate under post accident conditions. The containment area high range monitors perform primary containment monitoring during and after an accident and the SGTS effluent SPINGS are used to monitor post accident effluents.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0041 UFSAR Revision No. N/A

Reference Document: DER 91-0102 Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Control Center Heating, Ventilation, and Air Conditioning (CCHVAC) Ductwork Leak Test Acceptance Criteria

SUMMARY:

This evaluation justifies modifying the acceptance criteria for leak testing the CCHVAC ductwork based on revised assumptions for unfiltered inleakage and for reduced ingress and egress inleakage from the present control center door configuration. Reanalysis of the control center personnel exposure received in radiological emergencies provides the basis for changing the acceptance criteria. The allowable personnel dose is 30 rem to the thyroid over a 30 day period in accordance with General Design Criterion 19 and Standard Review Plan NUREG-0800. The original dose assumptions assumed a constant unfiltered control center infiltration rate of 10 cfm over the entire 30 day period. The original calculations resulted in a thyroid dose to control center personnel of 16.1 rem. The new assumptions assume an unfiltered control center infiltration rate of 32 cfm for the first 30 minutes due to a postulated damper failure in the operating CCHVAC division and subsequent switch-over to the other division within 30 minutes; an unfiltered control center infiltration rate of 9 cfm for the balance of the 30 days; and a constant 3 cfm for operating personnel ingress and egress over the entire 30 day period. The results of the reanalysis indicate that the thyroid dose to control center personnel would be 18.7 rem.

The CCHVAC system will still provide an acceptable post-accident environment with the new duct and door inleakage assumptions. The calculated thyroid dose to control center personnel is well below the 30 rem limit. Changing the leakage criteria does not change the design or function of the control center or CCHVAC.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0053 UFSAR Revision No. N/A

Reference Document: DER 91-0330 Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Justification for Using the Damaged Refueling Bridge Mast During the Second Refueling Outage

SUMMARY:

This evaluation justifies using the refueling bridge mast during the second refueling outage with a horizontal bracing member on the outermost mast truss removed. This structural member was damaged during load testing when the mast assembly was raised too high. It was removed for engineering observation and evaluation. The structure was returned to its original configuration after the second refueling outage.

The structural configuration of the refueling bridge mast during the second refueling was acceptable for the following reasons:

1. The forces acting on the outermost mast truss are due to the weight of the mast and the drag generated as the mast is being moved through the water. The weight of the mast is carried by the vertical truss tube sections. The horizontal drag forces are uniformly distributed over the length of the mast and, as such, the load is redistributed over the remaining structural members. The adjacent horizontal and diagonal members easily compensate for the loss of the horizontal member. The connection between the grapple and bail is not considered to be rigid. Therefore, there are no significant bending moments carried by the mast structure.

2. A confirmatory test was performed to determine acceptable operation of the mast. A dummy fuel bundle was lifted out of the spent fuel pool, placed in the reactor cavity, and returned to its original position in the spent fuel pool. The acceptance criteria of this test verified that the truss sections move up and down without binding and that the grapple satisfactorily performs its latching and unlatching function as intended.

Safety Evaluation No. 91-0053 (continued):

The mast is only used as a cable guide. The UFSAR does not describe an accident dealing directly with the fuel mast. Other accidents pertaining to fuel storage and handling described in the UFSAR are not affected by the structural configuration described here. The missing mast member did not adversely affect the function of the grapple. Use of the mast in this configuration did not affect the operating condition of the refueling platform; hoists or cranes for handling control rods and fuel assemblies; core internals; or the pressure vessel.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0058 UFSAR Revision No. N/A

Reference Document: COLR 3.1 Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Revision to Core Operating Limits Report (COLR)

SUMMARY:

This evaluation justifies the following changes described in COLR 3.1:

1. The maximum average planar linear heat generation rate (MAPLHGR) and the linear heat generation rate (LHGR) limits for the SVEA-96 lead fuel assemblies (LFA) have been revised to accommodate a change to the process computer model. The process computer database assumes the LFAs have 60 fuel pins. This is the same number of pins that GE9B fuel bundles have. The LFAs have 96 pins and, as a result, the calculated heat generation rates are 60% higher than the actual values. To allow for the proper calculation of the thermal limit margins, the MAPLHGR and LGHR limits have been raised 60%.

2. Rod block monitor (RBM) filter time constants and RBM setpoints have been revised. Specific RBM filter time constants have been added to ensure the effectiveness of the filter is not degraded. The RBM licensing basis supports any combination of time delays and filter time constants that are allowed by the system hardware. However, some combinations reduce the filters effectiveness. An ineffective filter, combined with the more limiting setpoints required to use it, results in more frequent unnecessary rod blocks than an unfiltered system.

   The RBM setpoint allowable values have been adjusted to correct a small variance between the COLR 3.0 values and the values based on a new Fermi 2 specific design calculation. The original RBM setpoints were obtained using a generic analysis which is applicable to Fermi 2. The analytical limits and nominal trip setpoints are the same in both the calculation and the generic analysis. However, the allowable values for the power and trip setpoints are slightly different.

Safety Evaluation No. 91-0058 (continued):

Thermal limits calculations have been performed using the NRC approved methods in GESTAR II. The thermal limit changes do not invalidate the reload licensing analysis. Revision of the LFA thermal limits does not result in a modification to the facility and does not change plant operation.

The filter time constants, and allowable setpoint values specified in COLR 3.1 do not invalidate the rod withdrawal error analysis and have no effect on any other transient. The specifications in COLR 3.1 do not result in any modification to the facility.

SAFETY EVALUATIO SUMMARY

Safety Evaluation No: 91-0061 UFSAR Revision No. 5

Reference Document: LCR 91-089-UFS Section(s) 9A4.1; 9A4.2; 9A4.10

Table(s)

Figure Change [X] Yes [ ] No

Title of Change: Control Room Fire Hazards Analysis

SUMMARY:

This evaluation justifies expanding the discussion in the UFSAR hazards analysis for the control room and reactor building fifth floor fire zone. This discussion includes the air conditioning room on elevation 677'-6" of the auxiliary building in the control room fire area and the auxiliary building stairwell (columns F9 to F12) in the reactor building fifth floor fire area. It also provides justification for having no fire detection instrumentation in these rooms. The Appendix R safe shutdown analysis considers the air conditioning room as part of the control room fire area and the auxiliary building stairwell as part of the reactor building fifth floor fire zone. A justification of the lack of fire protection in these two rooms is required because they contain safety-related cables.

The air conditioning room does not contain any cables or components required for the safe shutdown of the plant. This room does not contain any exposed combustible materials as all combustible materials are encased in metal housings (motor housings, cables in conduit). The room is heavily congested with conduit precluding the future storage of combustibles. As a result, no fire is postulated in the air conditioning room.

The auxiliary building stairwell contains two Division II reactor and auxiliary building HVAC system cables which are required to achieve and maintain safe shutdown. The safe shutdown analysis documents the fact that the loss of these cables due to a postulated fire would not prevent the safe shutdown of the plant because of the availability of Division I systems outside of the fire area. All combustible materials in the room are encased in metal enclosures (a gang box, cables in conduit) and are not readily accessible. Plant procedures do not allow storage of combustibles in stairwells. As a result, no fire is postulated in the stairwell.

This revision to the UFSAR fire hazards analysis does not affect the conclusions of the fire hazards analysis or the Appendix R safe shutdown analysis. There are no physical plant changes made by these changes.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0065 UFSAR Revision No. 5

Reference Document: DER 91-0428 Section(s) A.1.52

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Control Center Filtration System Charcoal Adsorber Testing Requirements

SUMMARY:

The purpose of this evaluation is to justify a revision to the UFSAR in which compliance with Regulatory Guide 1.52 is modified to reference the requirements of Table 5.1 of ANSI/ASME N509-1980. The requirements of Table 5.1 of ANSI/ASME N509-1976 were originally referenced in the UFSAR. These standards address the physical properties and performance requirements for the control center adsorber filter charcoal. The performance requirements of the new standard are slightly less stringent than the older 1976 standard. The differences are as follows:

1. The 1980 standard for methyl iodine requires testing at 30 degrees C whereas the 1976 standard requires testing at 25 degrees C.

2. The 1980 standard for elemental iodine requires a retention acceptance criteria of 99.5% whereas the retention acceptance criteria of the 1976 standard is 99%.

For the methyl iodine test requirements, increasing the test temperature to 30 degrees C increases the removal efficiency of the carbon. However, 30 degrees matches the actual charcoal service conditions and is the industry standard approved by the NRC for testing control room emergency filtration systems. The new charcoal performance standards are equal to or better than the older requirements for the elemental iodine test requirements.

There is no change to plant equipment or charcoal. The newly purchased replacement charcoal essentially meets the same design criteria as the used charcoal but is tested to slightly different criteria. The use of the newer charcoal testing criteria will assure that the control center filtration system charcoal adsorbers will perform as designed. The new testing criteria supports the same service as the old testing did.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0066 UFSAR Revision No. 5

Reference Document: DER 91-0160 Section(s) 3.3

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Revision to UFSAR Tornado Loading Discussion

SUMMARY:

This evaluation justifies rewording UFSAR subsection 3.3.2.3.7.2 to allow temporary non-seismic structures within 100 ft of the south wall of category 1 structures. This agrees with the current plant practice of locating temporary trailers and staging of construction materials outside of the reactor building during outages. Rewording UFSAR subsection 3.3.2.3.7.2 eliminates the need to write a safety evaluation every time temporary structures or material is placed near the plant.

Category 1 structures have been designed to resist the impact forces of tornado-generated missiles per Regulatory Guides 1.76 and 1.176. The Fermi 2 Safety Evaluation Report concludes that the plant design for externally generated missiles is acceptable. The missile protection design does not take credit for not having any structures within 100 ft of the the south wall of Category 1 structures. The analysis in UFSAR section 3.5.4, Barrier Design Procedures, envelopes the potential missiles that could be generated from temporary trailers and other outage related materials. Engineering probabilistic analyses of tornado missile hazards due to penetrations and openings in the reactor building and auxiliary building walls, tornado missile damage to the residual heat removal complex cooling towers, and spent fuel pool tornado protection are not changed by the addition of temporary trailers and construction materials outside of the reactor building.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0070 REV 1 UFSAR Revision No. 5

Reference Document: LCR 91-126-UFS Section(s)

Table(s) 8.3-2; 8.3-3; 8.3-4; 8.3-6; 8.3-7

Figure Change [ ] Yes [X] No

Title of Change: UFSAR Emergency Diesel Generator (EDG) Load Table Changes

SUMMARY:

This evaluation justifies revising the UFSAR EDG load tables to reflect changes in calculated EDG loads. The new calculation utilizes load data from other design calculations and applicable design documents. In addition, Cable losses (I^2R) were added to all loads and the swing bus loads (automatic loads from MCC 72-CF) were added to EDG 14. As a result, the peak loads for the following scenarios changed as follows:

1. For a loss of offsite power (LOOP) at 0 to 10 minutes, the highest calculated EDG load increased from 707 KW to 777 KW.

2. For a LOOP after 10 minutes, the highest calculated EDG load increased from 2649 KW to 2705 KW.

3. For a LOOP coincident with a loss of coolant accident (LOCA) at 0 to 10 minutes, the highest calculated EDG load increased from 3030 KW to 3124 KW.

4. For a LOOP coincident with a LOCA after 10 minutes, the highest calculated EDG load increased from 2846 KW to 2902 KW.

The highest calculated load, 3124 KW, occurs during a LOOP coincident with a LOCA at 0 to 10 minutes on EDG 14 when all EDGs are available. This load is within the short time rating (3135 KW) of the EDGs and is, therefore, in compliance with paragraph C.2 of Regulatory Guide 1.9, revision 2 and item 3.7.2 of IEEE Standard 387-1977.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0074 UFSAR Revision No. 5

Reference Document: LCR 91-130-UFS Section(s) 9A.4

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Residual Heat Removal (RHR) Complex and Condensate Storage Tank (CST) Yard Fire Analysis

SUMMARY:

This evaluation justifies a revision to subsections 9A.4.3.1 and 9A.4.7.2.1 of the UFSAR Fire Hazards Analysis to include analyses of a yard fire near the RHR complex and the CSTs. The previously approved fire hazards analysis did not address exterior hazards to safety related buildings and tanks. The RHR complex analysis demonstrates that an oil spill fire is the worst case scenario for a yard fire and that it will not adversely affect the RHR complex. The CST analysis includes a discussion of the diked area around the CSTs and how it prevents an exposure fire in the yard from affecting the tanks.

Revising the Fire Hazards Analysis by documenting the effects of a postulated fire in the yard adjacent to both the RHR complex and the CSTs is in accordance with the Generic Letter 86-10 guidance on yard fire analysis. The original Appendix R analysis remains valid and unchanged. These discussions provide additional information to demonstrate the accuracy of the Appendix R yard fire assumptions. The RHR complex analysis concludes that a yard fire cannot spread to the 590' elevation of the RHR complex and, therefore, will not damage safe shutdown equipment or circuits in the RHR complex. The CST analysis concludes that the CSTs are adequately protected from a potential exposure fire in the yard by the diked area surrounding them. In addition, the suppression pool can be used as an alternate water source if the CSTs are damaged in a fire.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0075 UFSAR Revision No. 5

Reference Document: LCR 91-135-UFS Section(s) N/A

Table(s) 7.5-2

Figure Change [ ] Yes [X] No

Title of Change: Corrections to UFSAR Table 7.5-2, "Safety-Related and Power Generation Display Instrumentation"

SUMMARY:

This evaluation justifies miscellaneous corrections to UFSAR Table 7.5-2, "Safety-Related and Power Generation Display Instrumentation" to make it agree with other Fermi 2 base configuration design documents (BCDD). These changes include corrections to instrument numbers, design class QA level/seismic categories, number of channels, alarm setpoints, instrument ranges, and instrument accuracies.

No physical modifications to the plant were required as a result of these corrections. These corrections do not affect any BCDDs or the postulated accidents and accident analyses described in UFSAR Chapters 6 and 15. There is no reduction in the margin of safety as stated in the UFSAR, NRC Safety Evaluation Report, or Technical Specifications.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0082 UFSAR Revision No. 5

Reference Document: DC 4388 Section(s) 8.3

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Deletion of 4160V Ground Fault Relay Settings in UFSAR

SUMMARY:

This evaluation justifies deleting the 4160V ground fault relay settings in UFSAR subsection 8.3.1.1.12.2. During a review of design calculations, an inconsistency between UFSAR subsection 8.3.1.1.12.2, "Circuit Protection", and the actual 4160V ground fault relay settings was identified. The UFSAR stated that the relays are calibrated to operate with 0.5 amps secondary current, which is equivalent to 5 amps primary current. In practice, the relays are calibrated to operate at 15 amps primary current. This change eliminates the conflict between the UFSAR and the field settings and removes extraneous and misleading information.

This change allows the use of a more realistic calibration setpoint higher than the value previously stated in the UFSAR. The operation of equipment important to safety is enhanced because less equipment trips due to nuisance ground fault trips will be experienced. The field settings are still adequate to protect the power source and do not promote common mode failures such as overcurrent induced fires or a major fault.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0083 UFSAR Revision No. 5

Reference Document: LCR 91-149-UFS Section(s) 8.2

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: 120 KV Bus Maximum Continuous Voltage Change

SUMMARY:

This evaluation justifies reducing the 120 KV Bus maximum continuous voltage stated in UFSAR subsection 8.2.2.5.1 from 128 KV to 126 KV. This change is based on recent short circuit studies identified by the Detroit Edison Transmission Planning Department. These studies indicate that the maximum voltage may be reduced due to changes in the system load; network and capacitor changes that have made higher voltage excursions less possible; and improved system voltage control.

This revision will not change the available voltage on the downstream busses because the transformers will maintain bus voltage. The load flow and stability analyses previously evaluated in the UFSAR remain unchanged. The results of a design calculation indicate that all safety related busses are capable of performing their safety functions within design limits. Reducing the maximum overvoltage value reduces the potential for equipment overvoltage and overcurrent. The new maximum voltage is within the previously evaluated 120 KV/4.160 KV transformer SS 64 load tap changer setpoints and the reduction in the maximum source voltage does not have any impact on the 4160 V undervoltage trip setpoints.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0085 UFSAR Revision No. N/A

Reference Document: NPP 20.107.01 Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Procedure Change to Allow Manual Initiation of Standby Feedwater (SBFW) System During Loss of Reactor Feedwater Pump (RFP) Reactor Recirculation Pump (RRP) Runback Concurrent With Loss of Heater Drains

SUMMARY:

This evaluation justifies the revision of abnormal operating procedure (AOP) 20.107.01, "Loss of Feedwater or Feedwater Control", to allow manual initiation of the SBFW system during a RRP runback caused by the loss of one RFP, a coincident loss of heater drains, and reactor power greater than 70%. An evaluation of acceptance test data for the design change that allows the Fermi 2 to be operated in the maximum extended load line limit (MELLL) region indicates that when the reactor is operating at 100% power and 75% core flow (the upper end of the MELLL region) a RRP runback caused by the loss of one RFP and heater drains will run back power to 73%. Since the maximum feedwater flow of a single operating RFP can only maintain 70% reactor power, the steam/feedwater flow mismatch will lead to a level 3 scram in approximately 6 minutes.

In order to avoid a scram, operating procedure AOP 20.107.01 has been revised to direct the operators to:

1. Use SBFW to avoid a reactor pressure vessel low water level scram on a loss of one RFP, a coincident loss of heater drains, and reactor power greater than 70%.

2. Insert the cram array control rods to the extent necessary to reduce reactor power to 68%.

3. Terminate SBFW on any inadvertent high pressure coolant injection (HPCI) initiation.

Safety Evaluation No. 91-0085 (continued):

The use of SBFW reduces the probability of a reactor scram that would challenge equipment important to safety. The injection of SBFW at 1200 gpm is a cold water addition which results in a positive reactivity insertion. However, the margin of safety is not reduced because this reactivity addition is bounded by the 5000 gpm cold water (40 degrees F) HPCI reactivity addition analysis. The procedural step to terminate SBFW if HPCI is inadvertently initiated ensures the reactor will be operated within the HPCI analysis. If reactor core injection cooling (RCIC) is inadvertently initiated while SBFW is in operation, the HPCI addition analysis is still the bounding analysis since RCIC will only inject an additional 600 gpm. The limiting transients for a reactivity insertion are the turbine generator trip at 100% power and the feedwater controller failure at power levels below 100%. Therefore, the use of SBFW as described in this procedure change is not a limiting transient and has no adverse impact on safety.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0086 UFSAR Revision No. 5

Reference Document: LCR 91-152-UFS Section(s) 9.5

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Revision of Emergency Diesel Generator Fuel Oil System Operation in UFSAR Subsection 9.5.4.2

SUMMARY:

This evaluation justifies revising UFSAR subsection 9.5.4.2. The following changes have been made:

1) The description of the startup of the alternate fuel oil transfer pump and alarm has been changed to indicate that they do not initiate at the same level. The transfer pump start signal occurs at a higher level than the alarm.

2) The statement that the EDG fuel oil transfer system strainers are changed out has been changed to state that the filters are blown down. The strainers are Leslie self-cleaning "Y" strainers. When the strainer blowdown valve is opened, the strainers are designed to allow fuel oil to flush out the sediment collected on the screen.

3) The time available to take corrective actions has been changed from the running time of a full day tank to the running time of the fuel inventory at the low alarm level in the day tank. This reflects the time available to take action for the worst case scenario; both strainers plugged and the day tank low level alarm in.

This revision does not change the design, function, or operation of the EDGs. Long term sustained operation of the EDGs is maintained. The oil used to blow down the EDG fuel oil transfer system strainers cannot be used as fuel for the EDGs. However, the amount of oil used is insignificant (1 to 2 gallons per cleaning) compared to the 35,280 gallon fuel supply provided to maintain operation for seven days. Therefore, blowing down the strainers does not affect the ability of the EDGs to run for the required time interval.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0089 UFSAR Revision No. 5

Reference Document: LCR 91-158-UFS Section(s) 8.2

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Removal of the Minimum/Maximum Bus Voltage Limit Table in UFSAR Subsection 8.2.2.5.1 and Grid Configuration Study Year Revision

SUMMARY:

This evaluation justifies removing the minimum/maximum bus voltage limit table from UFSAR subsection 8.2.2.5.1 and revising the year that the latest grid configuration study was performed. The minimum/maximum bus voltage limit table was removed for the following reasons:

1. Calculations show that each bus has a different type of loading, total loading, and feeder length. Therefore, different voltage limits apply.

2. Electrical loading has varied over the years and the voltage limits set during construction are not the same during plant operation. Fixed voltage limits for all buses are no longer relevant to the operating plant environment.

3. Voltage limit evaluation is an ongoing process.

4. Bus voltage limits are determined and controlled by design calculations. Design calculations evaluate plant bus voltages which allow proper operation of all safety related electrical equipment and plant process systems. Design calculations consider the minimum AC voltage and current pickup values at the 120 V levels.

The year of the gr configuration study has been changed from 1981 to 1991 to reflect the updated tudy.

The removal of the voltage limits table does not change relay types or settings. Removing the voltage limits table from the UFSAR has no technical impact on the operation of the equipment since existing calculations control the equipment voltage limits. All safety related loads remain capable of performing their safety functions. The analyses, equipment, instrumentation, and voltage limits identified in the Technical Specifications are not changed. The identification of and response to a degraded grid condition described in UFSAR subsections 8.2.2.5.2 and 8.2.2.5.3 are not affected by this revision.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0050 UFSAR Revision No. N/A

Reference Document: NPP-CRBPP-91-011, -012, and -013 Section(s) N/A

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Control Rod Blade Processing

SUMMARY:

This evaluation justifies the activities required for processing the control rod blades (CRB) removed from the reactor during the second refueling outage. This safety evaluation covers: (1) activities prior to and following processing actions such as load handling and rigging; (2) CRB processing within the spent fuel pool (SFP); and (3) the effects of the activities on SFP cooling. The CRBs were processed one at a time. The CRB stellite bearings and velocity limiters were sheared off using spent fuel pool (SFP) curb mounted equipment. After being compacted, each CRB was inverted and placed inside a liner. The filled liners are stored in the spent fuel pool.

Crane hoists were selected with adequate capacity and double the normal safety factors to provide compliance with regulatory requirements. Safe load paths were used to move the CRBs and associated process equipment within the SFP area. With the exception of the CRBs, no equipment was moved over the spent fuel racks. The effects of a CRB drop over the spent fuel racks is bounded by the SFP fuel assembly drop analysis. The impact energy is below the UFSAR analysis impact energy of 2000 ft-lbs because the CRBs are approximately one-third the weight of a fuel assembly and the minimum water depth maintained over the CRBs is limited by procedure NPP-CRBPP-012 to ensure that the CRB drop height is less than the fuel assembly drop height. Similarly, the effect of a CRB drop on the SFP liner is bounded because the fuel bundle is heavier than a CRB and the drop heights are similar. The components related to CRB processing are seismically qualified and the loading effects of these components on the SFP supporting structures is acceptable. The impact of CRB processing does not have any significant effect on SFP cooling.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0091 UFSAR Revision No. 5

Reference Document: DER 87-398 Section(s) 6.4

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: UFSAR Clarification for Control Center Heating, Ventilation, and Air Conditioning (CCHVAC)

SUMMARY:

This evaluation justifies revising UFSAR subsection 6.4.2.3.1 to clarify the fact that the CCHVAC emergency intake isolation dampers remain closed when chlorine gas is detected. The original wording stated that these dampers "would close" when chlorine gas is detected. This wording incorrectly implied that the emergency intake isolation dampers can undergo a lineup change when chlorine is detected. The emergency intake isolation dampers are required to stay closed during chlorine release accidents if the CCHVAC system is in the normal, purge, or chlorine mode at the onset of the release. If the CCHVAC system is in the recirculation mode (emergency air intake dampers open) closure of the dampers is not required to keep chlorine concentrations down to an acceptable level in the control room. Therefore, there is no emergency air intake damper closing operation associated with the detection of chlorine gas and the resultant alignment of the CCHVAC system. This UFSAR revision is the result of findings in Technical Specification Improvement Program Item #749.

This UFSAR revision clarifies the operation of the CCHVAC system. This revision does not change its design, function, or operation. The CCHVAC system still conforms to the Regulatory Guide 1.95 requirement that the control room chlorine concentration should not exceed 15 ppm within two minutes after the operators are made aware of the presence of chlorine.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0093 UFSAR Revision No. 5

Reference Document: LCR 90-178-UFS Section(s) 7.3

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Removal of Reactor Water Level 1 Isolation Level in UFSAR Subsection 7.3.2.2.7.1

SUMMARY:

This evaluation justifies removing the reactor water level 1 isolation level setpoint in UFSAR subsection 7.3.2.2.7.1. This subsection contained a sentence stating that the reactor water level 1 isolation level is approximately 14 inches above the top of active fuel. The current Technical Specifications (Table 3.3.2-2 and Figure B 3/4.3-1) show a trip setpoint of greater than or equal to 31.8 inches and an allowable value of greater than or equal to 24.8 inches. A design calculation and General Electric Specification 22A2919AB agree with the Technical Specifications. The origin of the 14 inch setpoint and whether it is meant to represent the nominal trip setpoint, allowable value, or the analytical limit is unclear. Removal of this setpoint from UFSAR subsection 7.3.2.2.7.1 provides consistency as the values for the reactor water levels 2 and 3 are not found in the text.

This revision does not have any effect on the design, function, of operation of the plant. The actual values for the reactor water level 1 isolation level remain unchanged in the Technical Specifications and design calculation.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0094 UFSAR Revision No. 5

Reference Document: LCR 91-165-UFS Section(s) 13.4

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: UFSAR Chapter 13 Revisions

SUMMARY:

This change: (1) revises UFSAR subsection 13.4.3.2 to remove the requirement that the secretary of the Nuclear Safety Review Group (NSRG) be appointed from the membership of the NSRG; and (2) revises UFSAR subsection 13.4.3.3 to remove the requirement for the Independent Safety Engineering Group (ISEG) to make detailed recommendations to the chairman of the NSRG.

Eliminating the requirement that the NSRG secretary shall be a member of the NSRG is administrative in nature. There is no change to any system, structure, or component. No new mode of plant operation is introduced. NSRG staffing requirements and all other requirements of Technical Specification 6.5.2 are not changed by this revision.

Eliminating the requirement that the ISEG make detailed recommendations to the chairman of the NSRG makes UFSAR subsection 13.4.3.3 consistent with the requirements of Technical Specification 6.2.3. As amended by NRC approved License Amendment 63, this technical specification only requires that ISEG make recommendations to the Vice President - Nuclear Engineering and Services. The ISEG staffing and reporting requirements are not altered by this change.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0099 UFSAR Revision No. 5

Reference Document: LCR 91-024-UFS Section(s) 7.7; 9.4; 10.4; 11.3

Table(s) 11.3-1; 11.3-2; 11.3-3; 11.3-4; 11.3-5; 11.3-7

Figure Change [ ] Yes [X] No

Title of Change: Incorporation of Operation of the Off-Gas System at Flowrates Greater Than 40 scfm into the UFSAR

SUMMARY:

This evaluation justifies incorporation of off-gas system operation at flowrates greater than 40 scfm into the UFSAR. Per deviation event report, DER 91-0598, an engineering functional analysis was performed to support the long term operation of the off-gas system with condenser air inleakage flow in excess of the 40 scfm UFSAR value. This analysis concluded that air flows greater than or equal to 80 scfm have no adverse impact on the functioning and performance of the equipment, associated instrumentation, and accessories. The analysis further concludes that the limiting factor for the capacity of the off-gas system is the capacity of the ring water vacuum pumps when operating both pumps in parallel. As a result, UFSAR sections 9.4, 10.4, and 11.3; and tables 11.3-1, 11.3-2, 11.3-3, 11.3-4 and 11.3-5 have been revised to state that: (1) The off-gas system is capable of processing off-gas with air flows greater than 40 scfm and, (2) The off-gas parameters will vary in the event of air flows greater than 40 scfm.

In addition, various inconsistencies and redundancies were removed from UFSAR sections associated with operation and analysis of the off-gas system. As a result, the following changes have been made:

1. UFSAR subsection 11.3.2.7.5 has been revised to change the xenon residence time in the charcoal adsorbers from 14 to 16 days. The derived residence time for xenon in the charcoal adsorber is 17.3 to 18.9 days. Therefore, it is acceptable to conclude that the residence time design basis value is 16 days.

2. UFSAR subsection 11.3.3.4 has been revised to change the offsite dose limit from 0.17 rem to 0.5 rem. The original Safety Guide 26 specified an offsite dose limit of 0.17 rem. However, the UFSAR commits Fermi 2 to the 0.5 rem limit of Regulatory Guide 1.26.

Safety Evaluation No. 91-0099 (continued):

3. UFSAR subsections 11.3.3.4 and 11.3.3.5 have been revised to delete the offsite doses that result from a total failure of the offgas system. The offgas system failure analysis is provided in UFSAR section 15.11.

4. UFSAR subsection 11.3.2.7.3.1 has been revised to state that the value of the dynamic adsorption coefficient, K_D, and the residence time are determined experimentally or derived per calculation. The UFSAR previously stated that K_D and the residence time were only determined experimentally.

5. The chiller outlet off-gas temperature in UFSAR Table 11.3-4 has been changed from less than or equal to 4 degrees F to 14 degrees F. The former number did not agree with the numbers stated in UFSAR subsections 11.3.2.7.5 and 11.3.3.3.8, operating experience, or charcoal adsorber tests.

6. The ranges of charcoal filter outlet flow transmitter N530 and off-gas charcoal units to adsorber filters pressure transmitter N525 in UFSAR Table 11.3-5 have been changed to reflect the current ranges in PDC 8471, revision C, and the central component data base.

7. UFSAR subsection has been revised to change the off-gas high flow alarm from 50 scfm to 55 to 70 scfm. This setpoint was revised per engineering design package, EDP 11816.

8. Precooler and chiller outlet temperatures have been deleted from UFSAR subsections 11.3.2.7.5 and 11.3.3.3.6 as this information is already in UFSAR Table 11.3-4. In addition, UFSAR section 11.3 has been revised to provide clarification and reference for the above changes.

The operation of the off-gas system with an air flow greater than 80 scfm has no adverse impact on the function or performance of the equipment, associated instrumentation, or accessories. The radioisotope inventory values that result from this change are still within the existing bounds in UFSAR section 11.3 and the accident analysis in section 15.11. Offsite doses will remain below NRC limits and will also be ALARA. Adequate safeguards and procedures exist for detecting and limiting offsite doses well in advance of approaching any dose limits as the Fermi 2 Offsite Dose Calculation Manual provides the methods for both release rates and offsite dose analysis; technical specifications and administrative controls limit the off-gas release rate. The Failed Fuel Action Plan provides the required ALARA safeguards.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0107 UFSAR Revision No. 5

Reference Document: LCR 91-190-UFS Section(s) 9.1

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Discussion of Fermi 2 Special Lifting Devices to the UFSAR

SUMMARY:

This evaluation justifies adding a discussion of special lifting devices and their periodic testing in UFSAR subsection 9.1.4.4. A special lifting device is a device designed specifically for handling a certain type of load. The Fermi 2 special lifting devices are the reactor pressure vessel head strongback and the dryer/separator lifting device. Previously, the UFSAR did not provide a discussion of special lifting devices or any guidance on their periodic testing. This change revises UFSAR subsection 9.1.4.4 to state that (1) the Fermi 2 special lifting devices meet the criteria of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" and, (2) Fermi 2 meets the testing guidelines of NUREG-0612 by performing the load bearing weld full non-destructive examination option of ANSI N14.6-1978 at five year intervals. A visual inspection of these devices is performed each year and before each period of use.

The addition of the discussion of special lifting devices in the UFSAR does not affect existing plant equipment or change operating procedures. No new equipment, modifications, or testing is introduced by this change. The special lifting devices and their testing are in conformance with NRC guidelines in NUREG-0612.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0109 UFSAR Revision No. 5

Reference Document: LCR 91-193-UFS Section(s) 9.1

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: High Density Spent Fuel Racks Neutron Absorber Material Surveillance Program UFSAR Reference Change

SUMMARY:

This evaluation justifies changing the UFSAR reference for the high density spent fuel racks Boraflex neutron absorber material surveillance program from TM-586, "Joseph Oat Corporation, Licensing Input on High Density Spent Fuel Racks for Fermi 2 Project" to Northeast Technology Corporation (NETCO), "Revised Boraflex Coupon Surveillance Program Document No. 073-01". This surveillance program monitors the condition of the Boraflex neutron absorbing material in the spent fuel storage racks. The new reference describes how the neutron absorber material surveillance program in currently carried out. The Joseph Oat Corporation program called for periodic measurements of the Boraflex coupons dimensions to be taken over the life of the plant and compared to baseline data. The baseline data was to be taken by the Joseph Oat Corporation during fabrication and prior to coupon irradiation. However, the baseline data has been lost. As a result, a DER was written to document that the Joseph Oat Corporation program could not be properly implemented without the baseline data. The DER corrective action contracted NETCO to take current dimensions of the Boraflex coupons. Future measurements will be compared to this new baseline data. A new in-service testing program for testing the Boraflex coupons by NETCO was developed to replace the Joseph Oat Corporation program and is now referenced in UFSAR subsection 9.1.2.2.2.

The NETCO program makes the following additional changes:

1. The method of fastening the coupon housing front, center, and back plates has been changed. Formerly, the plates were tack welded together. The front cover plate currently has deformable capture tabs that are bent around the back of the center plate. The center plate is still tack welded to the back plate.

2. Two of the three unirradiated coupons that were removed prior to the first refueling outage will be reinstalled in the spent fuel pool. These coupons will be removed prior to the fourth refueling outage and tested. These coupons will aid in determining the maximum anticipated Boraflex shrinkage in the high density fuel storage racks.

Safety Evaluation No. 91-0109 (continued):

3. The three irradiated coupons removed prior to the second refueling outage will be reinstalled in the spent fuel pool. These coupons will also aid in determining the maximum anticipated Boraflex shrinkage in the high density fuel storage racks.

4. The NETCO testing program utilizes a radioassay of the coupon surface for beta and gamma radiation. This provides an indication of the extent of water permeation in the Boraflex.

5. The NETCO program does not require neutron radiography of the coupons as this is not a quantitative measure and there are small variations in the boron-10 loading in Boraflex.

6. The NETCO program changes the number of coupons removed per testing interval from three to two. This allows the Boraflex coupons to be removed throughout the design service life of the fuel racks as opposed to the eight year period under the Joseph Oat program.

These changes are limited to the high density fuel storage rack neutron absorbing material surveillance program. These changes restore the program to a functional status and meet the program's design requirements. The NETCO surveillance program follows the EPRI guidelines for Boraflex surveillance programs. The weight of the surveillance specimen tree which holds the sample coupons remains well within the bounds of the fuel handling accident analyzed in the UFSAR. These changes do not impact the seismic qualifications of the fuel racks. Technical Specification 6.8.5.d requires that a program be established, implemented, and maintained to assure that any unanticipated degradation of the high density spent fuel racks will be detected and will not compromise the integrity of the racks. The requirements of this technical specification are not impacted by these changes.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0112 UFSAR Revision No. 5

Reference Document: LCR 91-196-UFS Section(s) 9A.6

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Revisions to UFSAR Section 9A.6, "Fire Protection Conditions for Operations"

SUMMARY:

This evaluation justifies revising UFSAR Section 9A.6, "Fire Protection Conditions for Operation" to bring fire protection systems surveillance intervals into closer conformance with National Fire Protection Association (NFPA) codes; facilitate the implementation of compensatory measures; and correct typographical errors. The changes are as follows:

1. The fire detection instrumentation functional test surveillance interval in UFSAR subsections 9A.6.1.2.1 and 9A.6.1.2.2 have been changed from six months to twelve months. The twelve month interval meets the functional testing interval requirements of the 1990 edition of NFPA 72E.

2. UFSAR subsection 9A.6.1.2.3 has been deleted. This section covered non-supervised fire detection alarms circuits. It was deleted because there are no non-supervised fire detection alarm circuits at Fermi 2.

3. Typographical errors in UFSAR subsection 9A.6.3.1 have been corrected. These changes are editorial and do not affect the Fire Protection Program.

4. A note has been added to UFSAR subsections 9A.6.3.1 (spray and sprinkler systems), 9A.6.4.1 (carbon dioxide systems), and 9A.6.5.1 (halon systems) to indicate which of these fire suppression systems are in fire zones containing redundant safe shutdown equipment. This change allows easier and more reliable identification of areas that contain redundant safe shutdown equipment and also ensures that the proper fire watch (continuous or hourly) will be assigned when a fire suppression system is declared inoperable.

Safety Evaluation No. 91-0112 (continued):

5. The manual cable spreading room sprinkler system has been removed from UFSAR subsection 9A.6.3.1. This room already has a halon suppression system and there is no requirement to have backup manual suppression systems to protect plant systems or equipment.

6. The puff test requirement for UFSAR subsections 9A.6.4.2.2 (carbon dioxide systems) and 9A.6.5.2.1 (halon systems) has been deleted. This test has been deleted because it does not demonstrate system operability and is not required by the NFPA 12 and 12A.

This revision does not impact the operability or function of any fire protection component. The changes do not affect the operation, function, or reliability of any plant system or component. The consequences of an accidental release of a fire suppression agent (water, CO_2, or halon) resulting from the failure or inadvertent operation of a fire suppression system is analyzed in UFSAR subsection 9.5.1. The consequences of fire protection system or component malfunction is addressed in UFSAR section 9A.6. These changes do not alter the level of protection provided for the plant systems addressed in the Fermi 2 Technical Specifications or Bases.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0116 UFSAR Revision No. 5

Reference Document: LCR 91-197-UFS Section(s) 8.3; 9.4

Table(s) 9.4-8; 9.4-11

Figure Change [ ] Yes [X] No

Title of Change: Miscellaneous Corrections and Clarifications to the UFSAR Discussion of the Residual Heat Removal (RHR) Complex Ventilation System

SUMMARY:

This evaluation justifies making corrections and clarifications to UFSAR sections 8.3 and 9.4 to resolve the findings of the Fermi 2 Independent Safety Engineering Group (ISEG) Report 90-011. These changes are as follows:

1. The design ambient temperature for the emergency diesel generator (EDG) rooms has been changed from 125°F to 122°F in UFSAR subsection 8.3.1.1.8. UFSAR subsection 9.4.7.1, Safety Evaluation Report section 9.4.3, and a design calculation correctly state that the EDG room has a 122°F maximum ambient design temperature.

2. The reference to EDG switchgear room ventilation system local manual control switches in UFSAR subsection 9.4.7.2.5 has been removed because these switches do not exist.

3. The reference to control room indication for EDG room temperature and high HVAC filter differential pressure in UFSAR subsections 9.4.7.2.5 and 9.4.7.3.5 has been removed because this indication does not exist.

4. UFSAR subsection 9.4.7.3.5 has been revised to correctly state that the system logic automatically starts the pump room ventilation system on high temperature or EDGs running. The original text incorrectly stated that it required both the high temperature and EDGs running to start the ventilation system.

5. The inappropriate references to the switchgear room under the row describing the pump room ventilation system in UFSAR Table 9.4-8 have been replaced with the words "pump room". The switchgear room ventilation system is described elsewhere in this table.

Safety Evaluation No. 91-0115 (continued):

6. The UFSAR Table 9.4-8 comments column for the pump room has been revised to state that the operator will take the necessary actions depending on room temperature. This table previously stated that the operator will take the necessary actions for a filter high pressure differential switch failure. However, there is no high differential pressure indication in the control room.

7. The pump room ventilation system fan capacity has been changed from 7500 scfm to 12500 scfm in UFSAR Table 9.4-11. The revised capacity is based on vendor drawings and the plant component database.

These revisions do not change the components or the design requirements of the RHR complex HVAC system. These changes only clarify and correct the UFSAR descriptions of the RHR complex HVAC system.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 91-0119 UFSAR Revision No. 5

Reference Document: LCR 91-138-UFS Section(s) 11.4

Table(s) 11.4-5; 11.4-6

Figure Change [ ] Yes [X] No

Title of Change: Removal of References to Gaseous Particulates and Iodine in the UFSAR; Additional Improvements to the Chemistry Program Described in the UFSAR

SUMMARY:

The evaluation justifies removing references to gaseous particulates and iodine and miscellaneous clarifications in order to strengthen the post accident sampling program (PASS) and improve the accuracy of the UFSAR as it pertains to the Fermi 2 chemistry program. Fermi 2 procedures do not require the use of gaseous particulate and iodine samples for alternate measurement of containment radiation or for estimation of core damage. The miscellaneous changes are intended to reference state of the art instrumentation and procedures while providing greater flexibility to utilize existing equivalent equipment. These changes also clarify the UFSAR to reflect true system capabilities and provide specific details as to the implementation of regulatory commitments.

Data gathered from a containment atmosphere particulates and iodine sample provides no input to the damage assessment calculations of Fermi 2 procedure 78.000.15, "Determination of Extent of Core Damage". Therefore, the elimination of the capability to gather gaseous particulate and iodine samples does not degrade the ability to mitigate core damage. In addition, this capability is not identified in emergency plan procedure EP 46, "Calculation of Estimated Containment High Range Radiation Monitor SGTS/AX Monitor Readings if Instruments are Inoperable or Offscale". Therefore, this change does not impact the Emergency Plan. The miscellaneous changes do not create any physical or procedural changes to the PASS system. These changes are consistent with the commitments identified in the NRC Fermi 2 SER Chapter 22, section II.B.3 and its subsequent supplements. These changes do not challenge the conclusions of the instrument line break analysis in the UFSAR. These changes do not affect the training; sampling and analysis; or maintenance requirements of Technical Specification 6.8.5.c, "Post-Accident Sampling".

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0002 UFSAR Revision No. 5

Reference Document: LCR 90-116-UFS Section(s) 8.3; 9A.5

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Revise the UFSAR to List the Exceptions to the Construction and Qualification Requirements for Special Wires and Cables

SUMMARY:

This evaluation justifies revising the UFSAR to list exceptions to the construction and qualification requirements of specification 3071-080, "Special Wires and Cables". The following exceptions have been added:

1. Balance of Plant (BOP) medium voltage cables utilized in the underground power supply

2. Wiring for the lighting, communications, and security systems

3. Internal control panel wiring in the control center

4. Vendor supplied wiring

The BOP medium voltage cables utilized in the underground power supply include some 5 kv and 15 kv rated cables that are used to interconnect transformers and load centers that are outside of the main power block. These cables were obtained from Detroit Edison stock and cable qualification documentation does not exist.

The lighting, communications, and security systems cabling is routed throughout the plant in separate enclosed raceways and generally does not enter the cable tray systems. However, some communications cables are routed in BOP cable trays. They use flexible conduit for separation from the other cables. The lighting cables are National Electrical Code NFPA type THHN or XHHW. Both insulation types are flame retardant systems. The communications cables are specialty cables specified by the manufacturer. The security system cables are a similar in design to those that meet the 3071-080 specifications but their qualifications are undocumented.

Safety Evaluation No. 92-0002 (continued):

The control center internal control panel wiring was purchased under specification 3071-080 but was not required to meet the radiation exposure and post accident (LOCA) environmental qualification requirements of IEEE-323. However, they do meet the requirements of IEEE-383. The insulation on these wires is much thinner than the wire insulation called for in specification 3071-080. However, after an accident, the control center is a milder environment and, therefore, the thinner insulation will not adversely affect the associated systems.

The qualification of the equipment internal wiring was the responsibility of the vendor or the vendor's suppliers. Detroit Edison was responsible for reviewing the vendor's qualification documentation. There were no requirements to meet the safety related equipment standards of IEEE 383 and IEEE 323 placed on the non-safety related equipment purchase specifications. The wire used was appropriate for the application. Equipment maintenance and modifications utilize wire originally used by the equipment vendor or wiring that meets the 3071-080 specifications.

All Fermi 2 cables are either certified to the flame testing requirements of IEEE 383-1974 or are fully enclosed in metallic raceway/enclosures with fire stops as required by the 3071-080 specifications. The installation techniques used at Fermi 2 do not impact the fire protection program. All safety related cables are fully qualified for their functions and for the areas in which they are located.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0009 UFSAR Revision No. 5

Reference Document: LCR 92-018-UFS Section(s) N/A

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Reactor Water Cleanup System (RWCU) Recirculation Pump B Motor Replacement

SUMMARY:

This modification replaces the motor for RWCU recirculation pump B. The motor was replaced due to excessive vibration. The new motor is rated at the same horsepower and voltage as the old motor (50 hp and 460 vac). The full load current of the new motor is 57 amperes whereas the original motor full load current is 60.5 amperes. Therefore, UFSAR figure 8.5-5 has been revised to change the full load current on bus 72E POS. 2D from 60.5 amperes to 57 amperes.

RWCU recirculation pump B is in the non-Q, seismic II/I portion of the system and does not serve a safety related function. No circuit modifications were required to accommodate the change. The small difference in full load current between the original motor and the replacement motor has been evaluated. Engineering considerations such as emergency diesel generator loading, short circuit current, power feeder rating, protective relay setting, and power uprate impact, were reviewed and found acceptable.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0010 UFSAR Revision No. 5

Reference Document: LCR 90-029-UFS Section(s) 9.2

Table(s) N/A

Figure Change [ ] Yes [X] No

Title of Change: Deletion of Lined Piping Requirements in the UFSAR

SUMMARY:

This evaluation justifies deleting the material specifications for make-up demineralizer lined pipe and sampling lines in UFSAR Subsection 9.2.3.3. This subsection originally stated that vessels and valves in contact with caustic or acid solutions are rubber lined and piping handling non-neutral flow is polypropylene lined. It also stated that sample lines are Type 304 stainless steel. In the original make-up demineralizer system design all of the non-neutral flow piping was not polypropylene lined and all valves were not rubber lined. Some piping was stainless steel or heavy walled carbon steel and some valves were polypropylene lined. As a result, the statements specifying rubber or polypropylene lined components and Type 304 stainless steel sample lines have been deleted. These statements are not necessary because alternate materials are available to handle non-neutral fluids. This change allows flexibility in choosing materials compatible with the fluids based on standard engineering practices.

This revision does not impact the function of the make-up demineralizer system. Removing these specifications has no impact upon the existing accident scenarios contained in UFSAR chapters 6 and 15. The make-up demineralizer system does not initiate these events. The make-up demineralizer system is not required for safe shutdown of the plant and no safety related equipment is located within close proximity of any of the make-up demineralizer components in question.

SAFETY EVALUATION SUMMARY

Safety Evaluation No: 92-0014 UFSAR Revision No. 5

Reference Document: LCR 92-019-UFS Section(s) 3.1; 4.5; 7.1; 7.6

Table(s) N/A

Figure Change [X] Yes [ ] No

Title of Change: Removal of Neutron Startup Sources

SUMMARY:

This evaluation justifies revising the UFSAR to remove statements that startup neutron sources are installed in the reactor core. The startup neutron sources were removed after the first refueling outage. Thev are not required in a sufficiently irradiated core and General Electric Service Information Letter SIL No. 215 recommended that the startup source holders be removed. General Electr.c based its recommendations on the fact that broken startup source holders had been discovered in seven BWRs. The earliest failure was observed after two fuel cycles.

The startup neutron sources do not impact the minimum neutron count rate required by the source range monitors for proper startup operation. The function of the startup neutron sources has been replaced by the irradiated fuel. This revision does not impact the function of the reactor core or its response to accidents. There is no impact on reactor structure, core flow, or procedures. Removal of the startup source holders precludes the potential for holder failures and the resultant dispersal of broken parts within the reactor vessel.

The following Technical Specification Amendments were incorporated into Revision 5 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
66	Use of hafnium in Control Rods	4.1 4.2 4.5
67	Rod Sequence Control System (RSCS) Removal	1.2 3.1 7.1 7.6 8.15.4 Table 7.6-12 Table 7.6-13
69	Maximum Extended Operating Domain (MEOD)	4.4 6.3 7.6 7.7 8.15.17 Table 7.6-9 Table 7.6-10 Tables 8.15.17-1 thru 8.15.17-8
74	Low Pressure Coolant Injection (LPCI) Response Times	6.3 15.6 Table 6.3-06 Table 6.3-07
77	Pressure/Temperature Curves	3.1 4.3 5.2 A1.99 Table 4.3-3 Table 4.3-4 Table 5.2-7 Table 5.2-8 Table 5.2-9 Table 5.2-10

END OF SAFETY EVALUATION SUMMARY REPORT