**J. Todd Conner**
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**DTE Energy**

10 CFR 50.71(e)

November 5, 2012
NRC-12-0067

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington D C 20555-0001

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

Subject: Submittal of Revision 18 to the Fermi 2 Updated Final Safety Analysis Report, 10 CFR 50.59 Evaluation Summary Report, Commitment Management Report and Revisions to the Technical Requirements Manual and the Technical Specifications Bases

Pursuant to 10 CFR 50.71(e) and 10 CFR 50.4(b)(6), Detroit Edison hereby submits an electronic version (CD) of Revision 18 to the Fermi 2 Updated Final Safety Analysis Report (UFSAR).

In accordance with 10 CFR 50.71(e), Revision 18 of the UFSAR reflects changes made as a result of license amendments and other changes made under the provision of 10 CFR 50.59. Revision 18 includes plant configuration changes made through the end of the fifteenth refueling outage which concluded on May 5, 2012.

Sections, Tables and Figures that have been changed in Revision 18 are marked "REV 18 10/12" in the lower right hand corner of each page and are annotated by revision bars in the appropriate margin.

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This submittal includes five enclosures as described below:

Enclosure 1 provides the 10 CFR 50.59 Evaluation Summary Report including brief descriptions of 10 CFR 50.59 Evaluations performed since the previous report submitted with UFSAR Revision 17. This report is being submitted to meet the requirements of 10 CFR 50.59(d)(2).

Enclosure 2 provides the Commitment Management Report which contains brief summaries of commitments that have been deleted or changed since the previous report submitted with UFSAR Revision 17. Detroit Edison's Fermi 2 administrative programs and procedures are consistent with the Nuclear Energy Institute's (NEI) "Guidelines for Managing NRC Commitment Changes" NEI 99-04 Revision 0, dated July 1999.

Enclosure 3 provides revised pages of Volume I of the Technical Requirements Manual (TRM) issued since the previous report submitted with UFSAR Revision 17. The TRM is incorporated by reference in the UFSAR; therefore, these pages are being submitted in accordance with 10 CFR 50.71(e).

Enclosure 4 provides revised pages of the Technical Specifications Bases (TSB) issued since the previous report submitted with UFSAR Revision 17. These pages are being submitted in accordance with the TSB control program in Technical Specification Section 5.5.10.

Enclosure 5 provides a summary of changes made to remove excessive detail from the UFSAR. The removed information was determined to be redundant or obsolete and has been removed in accordance with the guidance contained in NEI 98-03, Revision 1, "Guidelines for Updating Final Safety Analysis Reports", and Regulatory Guide 1.181.

Should you have any questions or require additional information, please contact Mr. Zackary W. Rad, Manager - Nuclear Licensing at (734) 586-5076.

Sincerely,

Enclosures:

1. 10 CFR 50.59 Evaluation Summary Report
2. Commitment Management Report
3. Summary of Revisions to Technical Requirements Manual, Volume 1, and the Revised Pages
4. Summary of Revisions to Technical Specifications Bases and the Revised Pages
5. Summary of Excessive Detail Removed from the Fermi 2 UFSAR

(A CD with Revision 18 of the Fermi 2 UFSAR is also enclosed)

cc: NRC Project Manager
NRC Resident Office
Reactor Projects Chief, Branch 4, Region III
Regional Administrator, Region III
Michigan Department of Environmental Quality
Resource Management Division – Radiological Protection Section

I, J. Todd Conner, 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.

J. Todd Conner
Site Vice President

On this 5 day of November, 2012 before me personally appeared J. Todd Conner, being first duly sworn and says that he executed the foregoing as his free act and deed.

Sharon S. Marshall
Notary Public

SHARON S. MARSHALL
NOTARY PUBLIC, STATE OF MI
COUNTY OF MONROE
MY COMMISSION EXPIRES Jun 14, 2013
ACTING IN COUNTY OF Monroe

**ENCLOSURE 1 TO**
**NRC-12-0067**

**10 CFR 50.59 Evaluation Summary Report**

## 50.59 EVALUATION SUMMARY

| | | | |
|---|---|---|---|
| **50.59 Evaluation No:** | 09-0210 Rev 0 | **UFSAR Revision No.** | 18 |
| **Reference Document:** | EDP-35607 Rev. B<br>LCR-09-051-UFS | **Section(s)** | 2.4.2.2.3, 2.4.2.2.4, 3.3.2.1, 3.3.2.2, 3.4.4.3, 3.5.1.3.1, 3.5.2, 3.5.2.1, 3.5.4.4, 3.5.4.7, 3.5 Missile Protection References, 3.7.2.1.3.1, 3.7.2.1.3.2, 3.12.3.2.3, 8.3.1.1.8.1, 8.3.1.1.8.2, 9A.4.3.2, 9A.4.7.7.1, 9A.4.7.7.2, 9A.4.7.7.3, A-1.76 |
| | | **Table(s)** | 3.5-2 |

**Figure Change** ☒ Yes ☐ No

**Title of Change: Replacement of Cables and Duct Banks for EDG Feeder Cables to 4.16kV Buses**

Fermi's design has the Emergency Diesel Generators (EDG) located remotely from the Reactor/Auxiliary Bldg in the Residual Heat Removal (RHR) Complex. The existing EDG Class IE 4160 V cables are routed between the RHR Complex and the Auxiliary building in underground duct banks and are being replaced along with new duct banks to address the non-conforming issue of the cables not being designed for continuous under water service. The existing duct banks are below the water table and the ducts slope away from their manholes, which result in the cables being susceptible to continuous immersion in water.

Two new class IE safety related electrical underground duct banks, one for each division, are installed above the water table and provide three new manholes for each division with adequate sump pumps to remove any water intrusion. This includes an above grade concrete reinforced electrical vault for each ductbank to allow entry into the RHR Complex East wall.

This change is specifically evaluated for changes in UFSAR described methodologies.

This change can be made without prior NRC approval because using Regulatory Guide 1.76, Rev. 1 and NUREG 0800, Section 3.5.3, Rev. 3 is a generically approved design basis tornado analysis method for the purpose it was approved. The design of the new underground ductbanks, manholes and cable vaults meet all the limitations and restrictions; therefore, it is not a "departure" from a method of evaluation described in the UFSAR.

This change can be made without prior NRC approval because the seismic methodology used for the underground ductbanks was consistent with the guidance in NUREG 0800, Section 3.7.3, Rev. 3, for underground ducts which is similar to the method used in the original design and the results are slightly more conservative for the analysis of bends. Therefore, it is not a "departure" from a method of evaluation described in the UFSAR and NRC prior approval is not required.

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## 50.59 EVALUATION SUMMARY

| | | | |
|---|---|---|---|
| **50.59 Evaluation No:** | 11-0168 Rev. 0 | **UFSAR Revision No.** | 18 |
| **Reference Document:** | EDP 35607 Rev. B | **Section(s)** | 3.7.2.1.3.2, 3.7 References |
| | DC-6423, Vol. I | | 3.8.4.1, 3.8.4.1.3, 3.8.4.2, |
| | DC-6426 Vol. I | | 3.8.4.4.3, 3.8.4.5.3, 3.8.4.6.1, |
| | 3071-396 | | 3.8.4.6.3, 3.8.4.6.4, A.1.142 |
| | 3071-397 | **Table(s)** | 3.2-1, 3.8-4, 3.8-20 , 3.8-21 |
| | LCR-11-040-UFS | | |

**Figure Change** ☐ **Yes** ☒ **No**

**Title of Change: Use of ACI 349-01 to Design the Replacement 4160-V Concrete Ductbanks Between the RHR Complex and the Auxiliary Buildings**

EDP 35607 designs and implements new Category I underground duct banks, manholes and cable vaults for routing new EDG Feeder Cables to the 4160-V buses. Construction of the new Category I concrete underground duct banks, manholes and cable vaults was per Fermi Design Specifications 3071-396 & 397 which were issued to support the EDP. Their design was per calculations DC-6423 Vol. I and DC-6426 Vol. I.

These two specifications invoke ACI 349 (Code Requirements for Nuclear Safety Related Concrete Structures) for construction. Additionally, Design Calculations DC-6423, Vol. I (New Electrical Duct Banks and Electrical Manholes) along with DC-6426 Vol. I (Electrical Cable Vaults at RHR Complex) use the methodologies and acceptance criteria in ACI 349-01 and RG 1.142, Rev. 2 to design the new Category I concrete structures.

This change is specifically evaluated for changes in UFSAR described methodologies.

This change can be made without prior NRC approval because using ACI 349-01 and R.G. 1.142, Rev. 2 is endorsed by the NRC for this type of application and the methodology is similar with that of ACI 318-71, which is described in the UFSAR. Therefore, this is not a "departure" from a method of evaluation described in the UFSAR.

Additionally, this change can be made without prior NRC approval because the use of the guidance in ACI 318-05 for the epoxy anchors is not contained in any of the other codes and the analysis utilizes similar guidance and methodology provided in ACI 349-01 and ACI 318-71, which is endorsed by the NRC. Therefore, this is not a "departure" from a method of evaluation described in the UFSAR.

This change can also be made without prior NRC approval because the use of the guidance provided in ACI 318-77 Handbook for analyzing combined axial tension and shear loads on the ductbanks does not appear in ACI 318 (1963 or 1971) code cited in UFSAR Table 3.8-4 or in the

ACI 349-01 code. Therefore, it is not considered a method described in the UFSAR or a "departure" from a method of evaluation described in the UFSAR.

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## 50.59 EVALUATION SUMMARY

| | | | |
|---|---|---|---|
| **50.59 Evaluation No:** | 11-0225 Rev. 0 | **UFSAR Revision No.** | N/A |
| **Reference Document:** | Temp Mod 11-0026 | **Section(s)** | |
| | | **Table(s)** | |
| | | **Figure Change** | ☐ Yes ☒ No |

**Title of Change:** **Opening Turbine Building Roof Vents due to High Temperatures**

Temporary Modification TM 11-0026 is prepared to open up to four Turbine Building (TB) roof vents namely U2200-01, U2200-02, U2200-08 and U2200-09 (Ref. I-2741-31) to alleviate the high temperatures in the Turbine Building.

In addition to Turbine Building Heating Ventilation and Air-conditioning (TBHVAC) exhaust stack radiation monitoring, Turbine Building 3rd floor environment will be continuously monitored with temporary air sampling monitors for evaluating any potential release. Open roof vents will be covered with a screen to protect TB areas from foreign material intrusion, animals, birds, and other potential debris. Any of the open roof vents will be closed per Temporary Modification (TM) 11-0026 if any of the following occur: the TBHVAC system trips, the normally negative Turbine Building pressure with respect to the outside is lost, rain, thunderstorms, or severe weather is imminent, the TB Stationary Particulate Iodine and Noble Gas (SPING) is out of service or a valid alarm is received, TB area temperature is approaching 65 deg F, or anytime Radiation Protection (RP)/ ALARA (As Low as Reasonably Achievable) declares an event where there is a valid airborne condition in the TB that would challenge the prudency of the open roof vents, anytime a major Mayfly hatch occurs, or as directed from the Operations Shift Manager based on operational conditions.

The total post Loss Of Coolant Accident (LOCA) dose to the control room operators transported via TBHVAC stack and up to four open TB roof vents has been evaluated to remain within the previously completed analysis noted in DC-6132 Vol. I. Post LOCA dose to the Technical Support Center (TSC) is evaluated in DC-6133 Vol. I based on the TB exhaust stack and vents as a representative release location and is not affected. The total dose to the control room remains bounded by the licensing basis dose as described in the UFSAR.

The change has no impact on the radiological release to the atmosphere because no changes are proposed to the normal operation of the Turbine exhaust path nor the radiation detection system located in the exhaust stack. Additional air sampling and radiation detection equipment will be installed at TB3 near the north and south end of the building to detect and monitor potential releases to the atmosphere through open vents. Any release through the roof vents will be evaluated based on the air sampling data. It is not expected that the activity would be different from releases through the vents compared to TB stack.

No change to the roof vent structure is being performed under the TM 11-0026; therefore, the roof vent sealing surfaces are not impacted. Open vents are provided with red light indication in the main control room.

Opening the vents does not affect their smoke removal function due to a fire, because they will already be in the fail safe position.

NRC Regulatory Guide 1.183 describes the set of assumptions, methodologies, and acceptance criteria that have been used to evaluate the radiological consequences associated with the design basis accidents known as the Alternate Source Term. There is no change to the Alternate Source Term (AST) methodology as a result of this activity.

TM 11-0026 has no adverse impact on fuel cladding, reactor coolant system pressure boundary and the containment nor on operator ability to maintain these barriers. As such, design basis limit for Fission Product Barriers as described in the UFSAR are not exceeded or altered.

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**50.59 EVALUATION SUMMARY**

| | | | |
|---|---|---|---|
| **50.59 Evaluation No:** | 11-0325 Rev. 0 | **UFSAR Revision No.** | 18 |
| **Reference Document:** | EDP-36567 Rev. A<br>TSR-36741 Rev. 0 | **Section(s)** | 1.2.2.7, 1.2.2.15.5, 8.2.1.2<br>8.2.2.1, 8.2.1.4, 8.2.2.3 |
| | | **Table(s)** | |
| | LCR-11-078-UFS | **Figure Change** | [X] Yes [ ] No |

**Title of Change:** **Add Third Row Breaker Equipment to the 345KV Switchyard**

TSR 36741 documents physical changes to the 345KV switchyard implemented by ITC while EDP 36567 modifies plant aspects to incorporate the ITC changes into plant operations. ITC added a third row of breakers in the switchyard between the 301 and 302 Buses incorporating two new breakers: BM and BT (hereby identified as the "B" row). The Brownstown #3 incoming line to the switchyard was relocated from the existing "C" row of breakers (between breakers CM & CT) to the new "B" row of breakers at position BQ between breakers BM and BT. This revised configuration eliminates the need for breaker CT which was removed. The revised configuration also affects the method of synchronizing Fermi's main turbine generator to the grid. The source signal for generator synchronization is taken from Bus 302 in lieu of the Brownstown #3 incoming line, which reflects the revised configuration of the "C" line breakers and disconnects and the fact that the 302 Bus is energized from offsite power during synchronization of the Fermi main generator to the grid. Synchronization to the 302 Bus also provides the proper functional configuration where the 302 Bus is powered from the Brownstown #2 line, the Brownstown #3 line, or both. In addition, line relaying now associated with the CM and CT breakers is relocated to the BM and BT breakers and incorporated into the overall logic. The new BM and BT breakers, along with the CF breaker, utilize the SF6 type design. ITC's practice is to also maintain exclusive control of breakers in their switchyards, except where a licensee has to retain specific control (e.g. - the CF and CM breakers), and have removed the Fermi 2 control room's remote control function for breakers DF and DM on the "D" row of breakers through which the Brownstown #2 incoming line is located at position Dl.

The modifications to the 345KV switchyard involve only non-Q, non-seismic, and non-safety related components that are not required to establish and maintain a safe shutdown condition nor required for subsequent mitigation and restoration from the effects of an accident or transient. The increase in the number of breakers from five to six does not adversely affect the frequency of occurrence of accidents or malfunctions previously evaluated. There are no adverse effects to previously evaluated accidents or malfunctions, no potential for the creation of a new type of event, no adverse impacts to fission product barriers, and no impact to evaluation methodologies as described in the UFSAR, therefore, prior NRC approval of this change is not required.

## 50.59 EVALUATION SUMMARY

| | | | |
|---|---|---|---|
| **50.59 Evaluation No:** | 12-0126 Rev. 0 | **UFSAR Revision No.** | 18 |
| **Reference Document:** | TSR-36828 Rev. 0 | **Section(s)** | 6.3.2.14, 6.3.2.15, 6.3.2.7 |
| | | **Table(s)** | |
| | LCR-12-029-UFS | **Figure Change** | ☐ Yes ☒ No |

**Title of Change: Reevaluation of Reflective Metal Insulation [RMI] Debris Head Loss in Design of RHR and CS Torus Suction Strainers**

Under TSR-36828, the DC-5979 analysis of RMI transport is revised to re-evaluate RMI transport under all postulated post-accident operating modes. The new quantities of RMI delivered to the RHR and CS suction strainers were used in revisions to the DC-0230 Vol I (CS) and DC-0367 Vol I (RHR) system hydraulic calculations to re-evaluate the mode-dependent debris head loss and the associated impacts on pump deliverable flow, pump net positive suction head, and anti-vortex minimum submergence. Some effects of these changes were determined to be adverse with respect RHR pump NPSH and anti-vortex minimum submergence as described in the UFSAR. Specifically:

**1. Anti-Vortex Minimum Submergence**
RMI debris head loss and anti-vortex minimum submergence requirements for the CS and RHR suction strainers were reevaluated for the higher flow operating modes of these systems subject to considerations of TS allowed EDG frequency tolerances and the more restrictive anti-vortex criteria of NUREG/CR-2772. The redefinition of ECCS pump torus suction anti-vortex minimum submergence based on NUREG/CR-2772 rather than the UFSAR Section 6.3.2.14 described methodology of Springer and Lubin is a change in a UFSAR described method and as such, screened in.

**2. RHR Runout**
DC-0367 Vol I Revision P predicts the onset of a small amount of air ingestion during this scenario where none was previously predicted for the UFSAR Section 6.3.2.15 LPCI Runout evaluation. Its occurrence has the effect of reducing available NPSH and pumping capacity. Although there is no effect on function, it is conservatively screened in.

**3. MODE 4/5 LPCI Injection**
DC-0367 Vol I Revision P minimum submergence analysis for Mode 4 & 5 LPCI operation with the suppression pool level at the Tech Spec minimum Level of minus 66 inches predicts that the existing EOP directed action to throttle the flow in accordance with the restrictions identified on 29.100.01 sheet 6 must be credited to prevent significant air ingestion. Previously, the minimum

submergence determined based on the method of Springer and Lubin established no throttling was required. This change does not affect procedural actions, but recognizes the need to credit them for prevention of air ingestion.

Note: The above adverse results were also affected by the need to revise the analyzed RHR torus suction from 4.82 ft above the bottom of the torus to 5.38 ft, consistent with that specified when the current RHR strainers were installed under EDP-29024. This is a plant physical characteristic input to these analyses that acted to make the results adverse relative to 10CFR50.59 screen criteria.

**4. Long Term Post-Accident Operation of LPCI and Torus Cooling**
Depending on torus temperature, if torus cooling is initiated with the other division operating in LPCI injection mode, manual operator action is required to either secure or throttle flow in the division that is operating in LPCI injection mode in order to maintain adequate pump NPSH. Like the previous item (Mode 4/5 LPCI Injection), the required throttling is performed per the existing EOP guidance (29.100.01), but it is an action that must be credited in order to meet the single failure criterion to preserve the division so that it can be used for torus cooling at a later time if necessary. Previously, the design basis did not recognize the need to protect NPSH for a division that might remain operating in LPCI injection mode.

**Bases:**
1. While the torus suction anti-vortex minimum submergence based on NUREG/CR-2772 was screened in as a new methodology, it and the method of applying NPHS penalties for air ingestion are endorsed under Reg Guide 1.82. Furthermore, the requirements are more restrictive than the licensed methodology.

2. Although some limited air ingestion is predicted to occur during the LPCI runout scenario, the LPCI function itself is not credited to meet post-LOCA Emergency Core Cooling System (ECCS) cooling requirements and the amount of NPSH margin available is sufficient to ensure one or more RHR pumps remain capable of performing the credited function for long-term post-LOCA torus cooling, considering the application of a 1% penalty on available NPSH per Reg. Guide 1.82.

3. Mode 4/5 LPCI injection is a manually initiated mode of operation for which there is no defined design or Tech Spec required minimum flow capability. Thus, the existing EOP-specified anti-vortex limits are sufficient to ensure the operators align the system and throttle flow such that excessive air ingestion will not occur and required net positive suction head will be preserved.

4. 29.100.01 sheet 6 already establishes limitations on RHR pump flow as a function of torus temperature to protect ECCS pump NPSH. The minimum time required for this action is greater than or equal to 20 minutes depending on torus temperature which is consistent with that within which torus cooling must be initiated following the DBA large break LOCA under the exiting

licensing basis (UFSAR Section 6.3.2.14). LPCI flow control is implemented by throttling the E1150F017A/B from the control room using Class 1E circuits and power. RHR flow indication (E11R801A/B) in the control room is Class 1E powered and fully qualified specifically to support post-accident operator action to manually throttle RHR (RACTs 2597, Reg. Guide 1.97). The only credible malfunction would occur due to failure of the operators to throttle flow in accordance with the EOPs. As such the probability of this malfunction is very low and effectively equivalent to the probability of failing to perform the manual action to initiate the mode itself. In the event that the operators did fail to throttle LPCI flow and that division of RHR was assumed to be lost, the failure is equivalent to postulating an existing analyzed single failure – namely the loss of a division of ECCS. Since the additional LPCI injection was not required anyway the accident response relative to required ECCS injection is unaffected.

## 50.59 EVALUATION SUMMARY

| | | | |
|---|---|---|---|
| **50.59 Evaluation No:** | 12-0172 Rev.0 | **UFSAR Revision No.** | -- |
| **Reference Document:** | EDP-36982 Rev. D | **Section(s)** | |
| | EDP-36984 Rev. E | | |
| | 22.000.03 Rev. 88 | **Table(s)** | |
| | LCR-12-037-UFS | **Figure Change** | [X] **Yes** [ ] **No** |

**Title of Change:** **Operation at Reduced Power with the South RFP and RFPT Out of Service**

On 6/25/12 at 13:30, Condenser vacuum was lost due to a catastrophic failure of the South Reactor Feed Pump/Turbine [RFP/T]. Operation at 100% full rated power is not possible until the south RFP/T is repaired or replaced. EDP-36982 installs mechanical devices (e.g. flanges with blank plates) to positively isolate the south RFP/T and selected auxiliaries from the balance of the plant to facilitate prolonged maintenance. Similarly, EDP-36984 electrically isolates power supplies and I&C components to facilitate maintenance. The UFSAR recognizes operation with a single RFP/T, permitting operation at reduced power levels. Operations procedure 22.000.03, Power Operation 25% to 100% to 25%, currently starts a second RFP/T at 60%. This procedure is modified to allow long-term plant operation on one RFP/T up to 70% power. Hardware aspects related to isolation of the RFP/T were found to screen out under 10CFR50.59. Evaluation of potential impacts to the license bases for accident response and operation above current pump procedural guidelines are considered in this evaluation.

This evaluation reviewed operating parameters of the North RFP/T and associated components at a load corresponding to 70% reactor power (increased from 60%) to assure that they will remain within design ratings. Based on being within design ratings, it is concluded that the North RFP/T and associated components will not have a significant increase in probability of malfunction. In addition, other BOP and safety related SSCs are reviewed for continuous operation at approximately 70% power (reduced from - normal full power) to assure reliable operation. UFSAR accidents were individually reviewed to assure that operation at reduced power would be bounded by the original analyses performed at 100% power and would not result in new accidents or different consequences. Based on the responses to 50.59 evaluation criteria; it was determined that continuous operation of the plant at 70% of rated thermal power while using only the North RFP can proceed without prior NRC approval.

ENCLOSURE 2 TO
NRC-12-0067

Commitment Management Report

## Commitment Management Report

Fermi 2 administrative programs and procedures are consistent with NEI's "Guidelines for Managing NRC Commitment Changes," NEI 99-04, Revision 0, dated July 1999. These guidelines discuss the need for a report to be submitted either annually or along with the UFSAR updates required by 10 CFR 50.71(e).

This report involves changes that have been made in the Fermi 2 commitment management database (referred to as the Regulatory Action Commitment and Tracking System or RACTS).

The changes being reported do not affect or change commitments or descriptions included in the UFSAR. Commitment changes are included in the following two tables:

- Table 1: Commitments that have been deleted from the Fermi 2 RACTS database because they are no longer applicable.
- Table 2: Commitments that have been revised in the RACTS database. The table includes the original commitment and reference document in addition to a brief description of the change.

Table 1

Regulatory Commitment Deleted From the Regulatory Commitment Tracking System (RACTS)

| RACTS NO. | ORIG. DATE | REFERENCE DOCUMENT | DESCRIPTION OF COMMITMENT | BASIS FOR DELETION |
|---|---|---|---|---|
| 87611 | 12/28/87 | IR 87-044 | Inspection Report (IR) 87-044 Included the following statement:<br>"The licensee should develop acceptance criteria to establish at what point corrective maintenance will be performed, and that this information will be trended and compared to the other accumulator check valves. This is considered an Open Item (341/87044-05)." | Deleted - Changed to one-time.<br>The open item was closed in Inspection Report 88-018 with the following statement:<br>"(Closed) Open Item (341/87044-05): Develop acceptance criteria and trend leaky control rod drive (CRD) to accumulator check valves. The licensee added the following criteria to Surveillance Procedure 44.010.201: Initiate a Work Request for all accumulators that alarmed in the first two minutes. The trending of leaky accumulator valves was being performed in conjunction with the Nuclear Plant Reliability Data System. The licensee's corrective actions were considered acceptable."<br>(DCR 11-1377) |
| 20118 | 01/07/2003 | Security Order | Revise procedures to require reinvestigation of all persons in accordance with the guidance contained in the order. Security will complete necessary reinvestigations to comply with the order. | Deleted - This commitment was intended to be one time closed.<br>The security order was incorporated into procedure MGA22, Rev: 5.<br>(DCR 11-1423) |

Table 1

Regulatory Commitment Deleted From the Regulatory Commitment Tracking System (RACTS)

| RACTS NO. | ORIG. DATE | REFERENCE DOCUMENT | DESCRIPTION OF COMMITMENT | BASIS FOR DELETION |
|---|---|---|---|---|
| 87296 | 08/21/1987 | NRC-89-0219 | In response to NOV 87-022-01, Fermi committed in NRC-87-0140 to replace carbon steel shafts and wafers for safety related Jamesberry butterfly valves with stainless steel shafts and wafers. | Deleted – changed to one time. Commitment has been fulfilled. (DCR 12-0738) |
| 90298 | 06/06/1990 | NRC-90-0095 | From RG 1.97 Letter responding to NRC SER, Fermi committed to upgrading the position indication circuit (position switches, interface terminal blocks, cables, and power supply) to Category 1 Criteria. | Deleted – changed to one time. Commitment has been fulfilled. (DCR 12-0738) |
| 94293 | 08/15/1994 | LER 94-003 | A set of electrical surveillance overlap drawings will be created and maintained. | Deleted – changed to one time. A Fermi Business Practice (FBP)-94 was developed to control and maintain the overlap drawings. (MLS10002 dated 10/05/2012). |

Table 2

Regulatory Commitment Revised in the Regulatory Commitment Tracking System (RACTS)

| RACTS NO. | ORIG. DATE | REFERENCE DOCUMENT | DESCRIPTION OF COMMITMENT | BASIS FOR COMMITMENT CHANGE |
|---|---|---|---|---|
| 99060 | 12/22/1999 | NRC-99-0104 | Per DECO Commitment in Letter NRC-99-0104, Fermi will comply with all three phases of the Motor Operated Valve Periodic Verification Joint Owner's Group (JOG). | Revised - commitment expanded to cover periodic verification of 5 additional Class D MOVs, not originally included in the JOG commitment. Fermi will apply alternative methods to address service-related degradation for five additional MOVs that were not included in the NRC-approved JOG MOV periodic verification program. (MLS10002 dated 09/07/2012) |

**ENCLOSURE 3 TO**
**NRC-12-0067**

**Summary of Revisions to Technical Requirements Manual, Volume I, and the Revised Pages**

## Summary of the Technical Requirements Manual (TRM) Volume I Changes

| | |
|---|---|
| Revision 101<br>04/29/2011 | Revised Core Operating Limits Report (COLR) for Cycle 15, Revision 1*. |
| Revision 102<br>09/20/2011 | Revised ionization fire detectors in Zone 17, Fourth Floor Reactor Building in Table TR3.12.1-1. |
| Revision 103<br>04/04/2012 | Revised Core Operating Limits Report (COLR) for Cycle 16, Revision 0*. |

The following pages are information only copies of the revised TRM pages for the above revisions.

* Pages of the COLR from Revision 101 and 103 are not attached. The COLR is included in Volume I of the TRM for convenience and ease of reference; however, it is not part of the information incorporated by reference into the UFSAR.

TABLE TR3.12.1-1 (Page 1 of 3)
Fire Detection Instrumentation

| FUNCTION | FIRE DETECTION ZONE | TOTAL NUMBER OF INSTRUMENTS[a] IONIZATION (X/Y) | PHOTOELECTRIC (X/Y) | THERMAL (X/Y) | INFRARED (X/Y) |
|---|---|---|---|---|---|
| 1. Reactor Building[b] | | | | | |
| a. Torus area | 1 | 8/0 | | | |
| b. NW corner rooms RHR pump | 2 | 4/0 | | | |
| c. SW corner rooms RHR pump | 3 | 4/0 | | | |
| d. SE corner rooms CRD HPCI | 4 | 9/0 | | | |
| e. NE corner rooms RCIC | 5 | 5/0 | | | |
| f. First floor | 7 | 31/0 | | 8/0 | |
| g. EECW system area second floor | 10 | 21/0 | | | |
| h. Third floor | 15 | 23/0 | | | |
| i. Fourth floor | 17 | 21/0 | | 2/0 | |
| j. Refueling area, fifth floor | 17 | 1/0 | | | 10/0 |
| 2. Auxiliary Building | | | | | |
| a. Basement N control air equipment | 4 | 6/0 | | | |
| b. Corridors, 562 ft, 563 ft | 5 | 2/0 | 2/0 | | |
| c. First floor mezzanine cable tray, 583 ft, 603 ft | 6 | 17/0 | | | |
| d. Switchgear room, corridor area second floor | 9 | 10/0 | | | |
| e. Cable tunnel | 9 | 10/0 | | | |

(continued)

(a) (X/Y) X is number of Function A (early-warning fire detection and notification only) instruments.
Y is number of Function B (actuation of fire suppression system and early warning and notification) instruments.

(b) The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

**ENCLOSURE 4 TO**
**NRC-12-0067**

**Summary of Revisions to Technical Specifications Bases**
**and the Revised Pages**

**Summary of Technical Specification Bases (TSB) Changes**

| | |
|---|---|
| Revision 52<br>09/02/2011 | Revised Sections B 3.3.6.2, B 3.3.7.1, B 3.6.4.1, B 3.6.4.2, B 3.6.4.3, B 3.7.3, B 3.7.4, B 3.8.2, B 3.8.5 and B 3.8.8 to reflect that the 37 day requirement for recently irradiated fuel no longer applies since it was applicable to GE11 only and GE11 fuel is no longer in the core. |
| Revision 53<br>09/29/2011 | Revised Sections B 3.4.6 to implement License Amendment 186. |
| Revision 54<br>03/22/2012 | Revised Sections B 3.8.1 and B 3.8.3 to implement License Amendment 188. |
| Revision 55<br>08/09/2012 | Revised Sections B 3.5.1 (SR Bases 3.5.1.2) as a CARD 07-22776-01 Enhancement, B 3.2.1, B 3.2.2, B 3.2.3 to reflect a change in Transient Analysis Methodology, B 3.8.1 to reflect ITC addition of third row of breakers and associated modifications in 345kV switchyard, B 3.0.6 to incorporate TSTF-71 and portion of TSTF-494, B 3.3.3.1 to incorporate TSTF-539, B 3.4.10 and B 3.7.4 for editorial changes. |

The following pages are information only copies of the revised TSB pages for the above revisions.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

individual monitor whose trip outputs are assigned to an isolation channel. Four channels of Fuel Pool Ventilation Exhaust Radiation-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Fuel Pool Ventilation Exhaust Radiation-High Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, this Function is not required. In addition, the Function is also required to be OPERABLE during OPDRVs and movement of recently irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. Due to radioactive decay, this Function is only required to isolate secondary containment during fuel handling accidents involving recently irradiated fuel. "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated.

4. Manual Initiation

The Manual Initiation push button channels introduce signals into the secondary containment isolation and SGTS initiation logic that are redundant to the automatic protective instrumentation channels and provide manual isolation capability. There is no specific UFSAR safety analysis that takes credit for this Function. It is retained for the overall redundancy and diversity of the secondary

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

ventilation exhaust). Four channels of Fuel Pool Ventilation Exhaust Radiation-High Function are available (two channels per trip system) and are required to be OPERABLE to ensure that no single instrument failure can preclude CREF System initiation. The Allowable Value was selected to ensure that the Function will promptly detect high activity that could threaten exposure to control room personnel.

The Fuel Pool Ventilation Exhaust Radiation-High Function is required to be OPERABLE in MODES 1, 2, and 3 and during movement of recently irradiated fuel assemblies in the secondary containment and operations with a potential for draining the reactor vessel (OPDRVs), to ensure that control room personnel are protected during a LOCA, fuel handling event, or vessel draindown event. During MODES 4 and 5, when these specified conditions are not in progress (e.g., OPDRVs, the probability of a LOCA is low; thus, the Function is not required. Also due to radioactive decay, this Function is only required to initiate the CREF system during fuel handling accidents involving recently irradiated fuel. "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated.

4. Control Center Normal Makeup Air Radiation-High

The control center normal makeup air radiation monitors measure radiation levels before filtration in the inlet ducting of the MCR. A high radiation level may pose a threat to MCR personnel; thus, automatically initiating the CREF System.

The Control Center Normal Makeup Air Radiation-High Function consists of two independent monitors. Two channels of Control Center Normal Makeup Air Radiation-High are available and are required to be OPERABLE to ensure that no single instrument failure can preclude CREF System

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

initiation. The Allowable Value was selected to ensure protection of the control room personnel.

The Control Center Normal Makeup Air Radiation-High Function is required to be OPERABLE in MODES 1, 2, and 3 and during OPDRVs and movement of recently irradiated fuel assemblies in the secondary containment, to ensure that control room personnel are protected during a LOCA, fuel handling event, or vessel draindown event. During MODES 4 and 5, when these specified conditions are not in progress (e.g., OPDRVs), the probability of a LOCA is low; thus, the Function is not required. Also due to radioactive decay, this Function is only required to initiate the CREF system during fuel handling accidents involving recently irradiated fuel. "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated.

# B 3.6 CONTAINMENT SYSTEMS

## B 3.6.4.1 Secondary Containment

BASES

### BACKGROUND

The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment.

The secondary containment is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure (e.g., due to pump and motor heat load additions). To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."

### APPLICABLE SAFETY ANALYSES

There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a loss of coolant accident (LOCA) (Ref. 1) and a fuel handling accident involving recently irradiated fuel inside secondary containment (Ref. 2). "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated. The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment.

Secondary containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

An OPERABLE secondary containment provides a control volume into which fission products that bypass or leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, can be diluted and processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement

BASES

APPLICABILITY (continued)

of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, secondary containment is only required to be OPERABLE during fuel handling involving recently irradiated fuel. "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated.

ACTIONS

A.1

With a Secondary Containment railroad bay access door inoperable there remains a redundant access door in an OPERABLE status. This door is capable of maintaining the Secondary Containment function. Therefore, the 7 day Completion Time gives a reasonable period of time to correct the problem given the availability of the other access door and the low probability of an event occurring that will challenge the Secondary Containment during this time period.

B.1

If secondary containment is inoperable for reasons other than Condition A, it must be restored to OPERABLE status within 4 hours. The 4 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

C.1 and C.2

If secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To

BASES

APPLICABLE SAFETY ANALYSES (continued)

previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated. The secondary containment performs no active function in response to either of these limiting events, but the boundary established by SCIVs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment.

SCIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

SCIVs form a part of the secondary containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs.

The power operated automatic isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO, along with their associated stroke times, are listed in Reference 3.

The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves and blind flanges are closed, or open in accordance with appropriate administrative controls. These passive isolation valves or devices are listed in plant procedures.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the

BASES

APPLICABILITY (continued)

secondary containment. Therefore, the OPERABILITY of SCIVs is required.

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant radioactive releases can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the secondary containment. Moving recently irradiated fuel assemblies in the secondary containment may also occur in MODES 1, 2, and 3. Due to radioactive decay, SCIVs are only required to be OPERABLE during fuel handling involving recently irradiated fuel. "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated.

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

The second Note provides clarification that for the purpose of this LCO separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SCIV. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SCIVs are governed by subsequent Condition entry and application of associated Required Actions.

BASES

BACKGROUND (continued)

The moisture separator is provided to remove entrained water in the air, while the electric heater reduces the relative humidity of the airstream to less than 70% (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter removes fine particulate matter and protects the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter collects any carbon fines exhausted from the charcoal adsorber.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, both charcoal filter train fans start. Upon verification that both subsystems are operating, the redundant subsystem is normally shut down.

APPLICABLE SAFETY ANALYSES

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents involving recently irradiated fuel (Ref. 2). "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated. For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The SGT System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE

BASES

LCO (continued)

subsystems ensures operation of at least one SGT subsystem in the event of a single active failure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System in OPERABLE status is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, the SGT System is only required to be OPERABLE during fuel handling involving recently irradiated fuel. "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated.

ACTIONS

A.1

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status in 7 days. In this Condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the

BASES

BACKGROUND (continued)

automatically switches to the recirculation mode of operation to prevent infiltration of contaminated air into the control room. A part of the recirculated air is routed through the emergency recirculation filter train. Outside air is taken in at one of two emergency outside air ventilation intakes and is passed through the emergency makeup filter train before being mixed with recirculated air. The air mixture is then returned to the control room.

The CREF System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA without exceeding 5 rem whole body dose or its equivalent to any part of the body. The recirculation mode will pressurize the control room to about 0.250 ± 0.125 inches water gauge to prevent infiltration of air from surrounding buildings. CREF System operation in maintaining control room habitability is discussed in the UFSAR, Chapters 6 and 9 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The ability of the CREF System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the UFSAR, Chapters 6 and 15 (Refs. 1 and 3, respectively). The recirculation mode of the CREF System is assumed to operate following a loss of coolant accident, fuel handling accident involving recently irradiated fuel, main steam line break, and control rod drop accident, as discussed in the UFSAR (Ref. 3). "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject of the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated. The radiological doses to control room personnel as a result of the various DBAs are also summarized in Reference 3. No single active failure will cause the loss of outside or recirculated air from the control room.

The CREF System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

APPLICABILITY (Continued)

In MODES 4 and 5, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the CREF System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During operations with potential for draining the reactor vessel (OPDRVs); and

b. During movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, the CREF System is only required to be OPERABLE during fuel handling involving recently irradiated fuel. "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated.

ACTIONS

A.1

With one CREF subsystem inoperable, the inoperable CREF subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CREF subsystem is adequate to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced CREF System capability. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

BASES

APPLICABILITY (continued)

a. During operations with a potential for draining the reactor vessel (OPDRVs); and

b. During movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, the Control Room AC System is only required to be OPERABLE during fuel handling involving recently irradiated fuel. "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated.

ACTIONS

A.1

With one control center AC subsystem inoperable, the inoperable control center AC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE control center AC subsystem is adequate to perform the control center air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the control center air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining subsystem can provide the required protection, and the availability of alternate safety and nonsafety cooling methods.

B.1 and B.2

If both Control Center AC subsystems are inoperable, the Control Center AC System may not be capable of performing its intended function. Therefore, the control room area temperature is required to be monitored to ensure that temperature is being maintained low enough that equipment in

# B 3.8 ELECTRICAL POWER SYSTEMS

## B 3.8.2 AC Sources - Shutdown

BASES

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BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources-Operating."

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APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 4 and 5 and during movement of recently irradiated fuel assemblies ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident involving recently irradiated fuel. Due to radioactive decay, AC electrical power is only required to mitigate fuel handling accidents involving recently irradiated fuel. "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated.

In general, when the unit is shut down the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or loss of all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, and 3 have no specific analyses in MODES 4 and 5. Worst case bounding events are deemed not

BASES

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the secondary containment to provide assurance that:

a. Systems providing adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident involving recently irradiated fuel are available. "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated.

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

AC power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.1.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1

An offsite circuit is considered inoperable if it is not

# B 3.8 ELECTRICAL POWER SYSTEMS

## B 3.8.5 DC Sources-Shutdown

BASES

| | |
|---|---|
| BACKGROUND | A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources-Operating." |

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the emergency diesel generators (EDGs), emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 4 and 5 and during movement of recently irradiated fuel assemblies ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident involving recently irradiated fuel. Due to radioactive decay, DC electrical power is only required to mitigate fuel handling accidents involving recently irradiated fuel. "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated

BASES

APPLICABLE SAFETY ANALYSIS (Continued)

fuel, as long as any fuel in the core or fuel pool is recently irradiated.

The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

At least one DC electrical power subsystem consisting of two 130 VDC batteries in series, two battery chargers, and the corresponding control equipment and interconnecting cabling is required to be OPERABLE to support required DC distribution subsystems required OPERABLE by LCO 3.8.8, "Distribution Systems - Shutdown." In addition, when the redundant division of the Class 1E DC electrical power subsystem is required by LCO 3.8.8, the other DC source subsystem, consisting of either a battery or a battery charger, the corresponding control equipment and interconnecting cabling, is required to be OPERABLE. This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving recently irradiated fuel and inadvertent reactor vessel draindown).

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the secondary containment provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

b. Required features needed to mitigate a fuel handling accident involving recently irradiated fuel are available. "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated.

# B 3.8 ELECTRICAL POWER SYSTEMS

## B 3.8.8 Distribution Systems-Shutdown

BASES

| | |
|---|---|
| BACKGROUND | A description of the AC and DC electrical power distribution system is provided in the Bases for LCO 3.8.7, "Distribution Systems-Operating." |

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC and DC electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC and DC electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC and DC electrical power sources and associated power distribution subsystems during MODES 4 and 5, and during movement of recently irradiated fuel assemblies in the secondary containment ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident involving recently irradiated fuel. Due to radioactive decay, AC and DC electrical power is only required to mitigate fuel handling accidents involving recently irradiated fuel. "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated.

The AC and DC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of Technical Specifications required systems, equipment, and components- both specifically addressed by their own LCO, and implicitly required by the definition of OPERABILITY.

In addition, during the shutdown conditions applicable to this LCO, cross-tie breakers between redundant safety related power distribution systems may be closed.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving recently irradiated fuel and inadvertent reactor vessel draindown).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the secondary containment provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

BASES

APPLICABILITY (continued)

b. Systems needed to mitigate a fuel handling accident involving recently irradiated fuel are available. "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated.

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC and DC electrical power distribution subsystem requirements for MODES 1, 2, and 3 are covered in LCO 3.8.7.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant divisions of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem division may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, recently irradiated fuel movement, and operations with a potential for draining the reactor vessel. By allowing the option to declare

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.6 RCS Leakage Detection Instrumentation

BASES

### BACKGROUND

GDC 30 of 10 CFR 50, Appendix A (Ref. 1), requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45, Revision 0, (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Limits on LEAKAGE from the reactor coolant pressure boundary (RCPB) are required so that appropriate action can be taken before the integrity of the RCPB is impaired (Ref. 2). Leakage detection systems for the RCS are provided to alert the operators when leakage rates above normal background levels are detected and also to supply quantitative measurement of leakage rates. In addition to meeting the OPERABILITY requirements, the monitors are typically set to provide the most sensitive response without causing an excessive number of spurious alarms. The Bases for LCO 3.4.4, "RCS Operational LEAKAGE," discuss the limits on RCS LEAKAGE rates.

Systems for separating the LEAKAGE of an identified source from an unidentified source are necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action.

LEAKAGE from the RCPB inside the drywell is detected by at least one of two or three independently monitored variables, such as sump level changes and drywell gaseous radioactivity level. The primary means of quantifying unidentified LEAKAGE in the drywell is the primary containment sump flow monitoring system.

The drywell floor drain sump flow monitoring system monitors the LEAKAGE collected in the drywell floor drain sump. This unidentified LEAKAGE consists of LEAKAGE from control rod drives, valve flanges or packings, floor drains, the closed cooling water systems, and drywell air cooling unit condensate drains, and any LEAKAGE not collected in the drywell equipment drain sump. The drywell floor drain sump has transmitters that supply level indications in the main control room.

BACKGROUND (continued)

The drywell floor drain sump flow monitoring system uses four basic leak detection methods to monitor the drywell floor drain sump. As the water in the sump is pumped out, the flow is metered by a flow integrator. Level switches are used to set fill time and pump-out time periods using adjustable reset timing devices. If the nominal pumping out or filling time for the sump is exceeded, an alarm is generated in the control room. In addition, if both pumps automatically start to handle the flow into the sump, an alarm is generated.

The supplementary drywell floor drain level monitor provides a continuous analog level measurement of the drywell floor drain level. This sump level monitor provides a rate-of-change measurement and alarm. The monitor has the sensitivity to detect a 1 gpm leak integrated over a 1 hour period (Ref. 3).

The primary containment atmosphere gaseous radioactivity monitoring system continuously monitors the primary containment atmosphere for airborne gaseous radioactivity. A sudden increase of radioactivity, which may be attributed to RCPB steam or reactor water LEAKAGE, is annunciated in the control room.

APPLICABLE SAFETY ANALYSES

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. 4 and 5).

A control room alarm allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. 6). Therefore, these actions provide adequate response before a significant break in the RCPB can occur.

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide confidence that small amounts of unidentified LEAKAGE are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

The LCO requires 3 instruments to be OPERABLE.

The drywell floor drain sump flow monitoring system is required to quantify the unidentified LEAKAGE rate from the RCS. Thus, for the system to be considered OPERABLE, it must be capable of determining the leakage rate. The identification of an increase in unidentified LEAKAGE will be delayed by the time required for the unidentified LEAKAGE to travel to the drywell floor drain sump and it may take longer than one hour to detect 1 gpm increase in unidentified LEAKAGE, depending on the origin and magnitude of the LEAKAGE. This sensitivity is acceptable for containment sump flow monitoring OPERABILITY.

The reactor coolant contains radioactivity that, when released to the primary containment, can be detected by the gaseous primary containment atmosphere radioactivity monitor. The airborne radioactivity detection system is included for monitoring gaseous activities because of its sensitivity and rapid response to RCS LEAKAGE, but has recognized limitations. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding containment or cladding defects. If there are few fuel element cladding defects and low levels of activation products, it may not be possible for the gaseous primary containment atmosphere radioactivity monitor to detect 1 gpm increase within 1 hour during normal operation. However, the gaseous containment primary atmosphere radioactivity monitor is OPERABLE when it is capable of detecting a 1 gpm increase in unidentified LEAKAGE within 1 hour given an RCS activity equivalent to that assumed in the design of the monitor (Ref. 7).

The drywell floor drain sump level monitoring system is capable of quantifying the unidentified LEAKAGE rate from the RCS. The system is considered to be OPERABLE, when it is capable of detecting an inleakage of 1 gpm within 1 hour. The identification of an increase in unidentified LEAKAGE will be delayed by the time required for the unidentified LEAKAGE to travel to the drywell floor drain sump and it may take longer than one hour to detect 1 gpm increase in

LCO (continued)

unidentified LEAKAGE, depending on the origin and magnitude of the LEAKAGE. This sensitivity is acceptable for containment sump level monitoring OPERABILITY.

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the drywell floor drain sump flow monitoring system, in combination with the gaseous primary containment atmosphere radioactivity monitor, and the drywell floor drain sump level monitoring system provides an acceptable minimum.

APPLICABILITY

In MODES 1, 2, and 3, leakage detection systems are required to be OPERABLE to support LCO 3.4.4. This Applicability is consistent with that for LCO 3.4.4.

ACTIONS

A.1

With the drywell floor drain sump flow monitoring system inoperable, the plant has lost one means to quantify leakage. However, the primary containment atmosphere gaseous radioactivity monitoring system and the drywell floor drain sump level monitoring system will provide indication of changes in leakage.

With the drywell floor drain sump flow monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every 12 hours (SR 3.4.4.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still available.

B.1

With the primary containment atmosphere gaseous radioactivity monitoring system inoperable, grab samples of the primary containment atmosphere must be taken and analyzed to provide periodic leakage information. Provided a sample is obtained and analyzed every 24 hours, the plant may continue operation since at least one other form of drywell leakage detection (i.e., drywell floor drain sump level monitoring system) is available.

The 24 hour interval provides periodic information that is adequate to detect LEAKAGE.

ACTIONS (continued)

C.1

With the drywell floor drain sump level monitoring system inoperable, SR 3.4.6.1 must be performed every 8 hours to provide periodic information of activity in the primary containment at a more frequent interval than the routine Frequency of SR 3.4.6.1. The 8 hour interval provides periodic information that is adequate to detect LEAKAGE and recognizes that other forms of leakage detection are available. However, this Required Action is modified by a Note that allows this action to be not applicable if the primary containment atmosphere gaseous radioactivity monitoring system is inoperable. Consistent with SR 3.0.1, Surveillances are not required to be performed on inoperable equipment.

D.1, D.2, D.3.1, and D.3.2

With the drywell floor drain sump flow monitoring system and the drywell floor drain sump level monitoring system inoperable, the only means of detecting LEAKAGE is the primary containment atmosphere gaseous radiation monitor. A Note clarifies this applicability of the Condition. The primary containment atmosphere gaseous radiation monitor typically cannot detect a 1 gpm leak within one hour when RCS activity is low. In addition, this configuration does not provide the required diverse means of leakage detection. Indirect methods of monitoring RCS leakage must be implemented. Grab samples of the primary containment atmosphere must be taken and analyzed and monitoring of RCS leakage by administrative means must be performed every 12 hours to provide alternate periodic information.

Administrative means of monitoring RCS leakage include monitoring and trending parameters that may indicate an increase in RCS leakage. There are diverse alternative mechanisms from which appropriate indicators may be selected based on plant conditions. It is not necessary to utilize all of these methods, but a method or methods should be selected considering the currant plant conditions and historical or expected sources of unidentified leakage. The administrative methods are pressure and temperature in the primary containment, equipment drain and floor drain sump pump activity, drywell floor drain sump level, cooling water differential temperature of the closed cooling water system, reactor water level, and containment atmosphere radioactivity. These indications, coupled with the atmospheric grab samples, are sufficient to alert the

ACTIONS (continued)

operating staff to an unexpected increase in unidentified LEAKAGE.

The 12 hour interval is sufficient to detect increasing RCS leakage. The Required Action provides 7 days to restore another RCS leakage monitor to OPERABLE status to regain the intended leakage detection diversity. The 7 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period.

E.1 and E.2

With both the primary containment atmosphere gaseous radioactivity monitoring system and the drywell floor drain sump level monitoring system inoperable, the only means of detecting LEAKAGE is the drywell floor drain sump flow monitoring system. This condition does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period.

F.1 and F.2

If any Required Action of Condition A, B, C, D or E cannot be met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the actions in an orderly manner and without challenging plant systems.

G.1

With all required monitors inoperable, no required automatic means of monitoring LEAKAGE are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR is for the performance of a CHANNEL CHECK of the required primary containment atmosphere gaseous radioactivity monitoring system. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.6.2

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.6.3

This SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Operating experience has proven this Frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, Revision 0, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.
3. UFSAR, Section 5.2.7.1.3.
4. GEAP-5620, April 1968.
5. NUREG-75/067, October 1975.
6. UFSAR, Section 5.2.7.4.3.3.
7. NUREG/CR-6861, December 2004

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.4

This SR provides verification that there is an adequate inventory of fuel oil in the day tank to support the EDG operation for a minimum of one hour at full load. The volume of fuel oil equivalent to one hour supply is 210 gallons.

The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and facility operators would be aware of any large uses of fuel oil during this period.

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day tanks once every 31 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during EDG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 10). This SR is for preventive maintenance. The presence of water does not necessarily represent a failure of this SR provided that accumulated water is removed during performance of this Surveillance.

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. It is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the

# B 3.8 ELECTRICAL POWER SYSTEMS

## B 3.8.3 Diesel Fuel Oil and Starting Air

BASES

### BACKGROUND

Each emergency diesel generator (EDG) is provided with a storage tank having a fuel oil capacity sufficient to operate that EDG for a period of 7 days while the EDG is supplying maximum continuous load discussed in UFSAR, Section 9.5.4 (Ref. 1) and Regulatory Guide 1.137 (Ref.2). This onsite fuel oil capacity is sufficient to operate the EDGs for longer than the time to replenish the onsite supply from outside sources.

Fuel oil is transferred from storage tank to day tank by either of two transfer pumps associated with each storage tank. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve, or tank to result in the loss of more than one EDG.

For proper operation of the standby EDGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195 (Ref. 3). The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level.

Each EDG has an air start system with adequate capacity for five successive start attempts on the EDG without recharging the air start receiver(s).

### APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in UFSAR, Chapter 6 (Ref. 4), and Chapter 15 (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The EDGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

BASES

ACTIONS (continued)

A.1

In this Condition, the 7 day fuel oil supply for a required EDG is not available. However, the Condition is restricted to fuel oil level reductions that maintain at least a 6 day supply. The fuel oil level equivalent to a 6 day supply is 30,240 gallons. These circumstances may be caused by events such as:

a. Full load operation required for an inadvertent start while at minimum required level; or

b. Feed and bleed operations that may be necessitated by increasing particulate levels or any number of other oil quality degradations.

This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of the fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the EDG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

B.1

This Condition is entered as a result of a failure to meet the acceptance criterion for particulates in one or more required EDG storage tanks. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, since particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and since proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated EDG inoperable. The 7 day Completion Time allows for further evaluation, resampling, and re-analysis of the EDG fuel oil.

BASES

ACTIONS (continued)

C.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.2 for new fuel that has already been added to a required EDG storage tank not within the required limits, a period of 30 days from the time of obtaining new fuel oil sample results is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combination of these procedures. Even if a EDG start and load was required during this time interval and the fuel oil properties were outside limits, there is high likelihood that the EDG would still be capable of performing its intended function.

D.1

With a Required Action and associated Completion Time not met, or the stored diesel fuel oil or starting air subsystem not within limits for reasons other than addressed by Conditions A through C, the associated EDG may be incapable of performing its intended function and must be immediately declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks of each required EDG to support each EDG's operation for 7 days at full load. The fuel oil level equivalent to a 7 day supply is 35,280 gallons when calculated in accordance with References 2 and 3. The required fuel storage volume is determined using the most limiting energy content of the stored fuel that meets the plant design basis requirements. Using the most limiting energy content as verified by direct energy content measurement or the known correlation of diesel fuel oil absolute specific gravity or API gravity to energy content, the required diesel generator output, and the corresponding fuel consumption rate, the onsite fuel storage volume required for 7 days of operation can be determined. The 7 day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note to indicate that when this test results in LPCI inoperability solely for performance of this required Surveillance, or when the LPCI swing bus automatic throwover scheme is inoperable due to EDG-12 being paralleled to the bus for required testing, entry into associated Conditions and Required Actions may be delayed for up to 12 hours until the required testing is completed. Upon completion of the Surveillance or expiration of the 12 hour allowance the swing bus must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. The LPCI swing bus automatic throwover scheme is typically not inoperable when EDG-12 is paralleled to the bus for testing purposes.

SR 3.5.1.3

The flow path piping has the potential to develop voids and pockets of entrained air. Maintaining the pump discharge lines of the HPCI System, CS System, and LPCI subsystems full of water ensures that the ECCS will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent a water hammer following an ECCS initiation signal. One acceptable method of ensuring that the lines are full is to vent at the high points. The 31 day Frequency is based on the gradual nature of void buildup in the ECCS piping, the procedural controls governing system operation, and operating experience.

SR 3.5.1.4

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a non-accident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves

BASES

APPLICABLE SAFETY ANALYSES (continued)

which turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent standard APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and AOOs. A complete discussion of the analysis code is provided in References 9 and 11.

LOCA analyses are then performed to ensure that the above determined APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference 10. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly.

The APLHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

The APLHGR limits specified in the COLR are the result of the fuel design, DBA, and transient analyses. The limit is determined by multiplying the smaller of the $MAPFAC_p$ and $MAPFAC_f$ factors times the exposure dependent standard APLHGR limits.

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APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 7) and operating experience have shown that as power is reduced, the margin to the required fuel design limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the fuel design limits; thus, this LCO is not required.

BASES

REFERENCES

1. NEDO-24011-P-A "General Electric Standard Application for Reactor Fuel" (latest approved version).

2. UFSAR, Chapter 4.

3. UFSAR, Chapter 6.

4. UFSAR, Chapter 15.

5. MDE-56-0386, "Fermi 2 Single Loop Operation Analysis," April 1987, and NEDC-32313-P, "Enrico Fermi Energy Center Unit 2 Single Loop Operation," September 1994.

6. NEDC-31515, Rev. 1, "Maximum Extended Load Line Limit and Feedwater Heater Out-of-Service Analysis for Enrico Fermi Atomic Power Plant Unit 2," August 1989.

7. NEDC-31843, "Maximum Extended Operating Domain Analysis for Detroit Edison Company Enrico Fermi Energy Center Unit 2," July 1990.

8. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

9. TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses, NEDE-32906P-A, Revision 3, September 2006.

10. NEDC-31928, "Fermi 2 SAFER/GESTR-LOCA, Loss-of-Coolant Accident Analysis," July 1991, Erratta and Addenda, April 1992.

11. Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients, NEDE-32906P Supplement 3-A, Rev. 1, April 2010.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 6, 7, and 8). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 9) to analyze slow flow runout transients. The operating limit is dependent on the maximum recirculation scoop tube mechanical stop setting in the Recirculation Flow Control System.

Power dependent MCPR limits ($MCPR_p$) are determined mainly by the three dimensional transient code (Refs. 10 and 11). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.

Transients involving increase in pressure and power are sensitive to the size of the steam volume and the availability of this steam volume to accommodate the reactor steam production. Larger steam volumes and longer or earlier availability result in less severe pressure transients. Thus operation of the turbine generator bypass valves and the availability of the moisture separator reheater have an effect on the transient results. For this reason the COLR contains MCPR limits for when the turbine bypass valves and/or moisture separator reheater are out-of-service (refer to LCO 3.7.6, "The Main Turbine Bypass System and Moisture Separator Reheater").

The MCPR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The MCPR operating limits specified in the COLR are the result of the design basis transient analysis. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to the MCPR safety limit at low power levels.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of $\tau$, which is a measure of the actual scram speed distribution compared with the assumed distribution. For $\tau > 0$, the MCPR operating limit is then determined based on an interpolation between the applicable limits for TRACG Option A (scram times of LCO 3.1.4,"Control Rod Scram Times") and TRACG Option B (realistic scram times) analyses. The parameter $\tau$ and MCPR operating limit must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in $\tau$ expected during the fuel cycle.

REFERENCES

1. NUREG-0562, June 1979.
2. NEDO-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).
3. UFSAR, Chapter 4.
4. UFSAR, Chapter 6.
5. UFSAR, Chapter 15.

BASES

REFERENCES (continued)

6. MDE-56-0386, "Fermi 2 Single Loop Operation Analysis," April 1987, and NEDC-32313-P, "Enrico Fermi Energy Center Unit 2 Single Loop Operation," September 1994.

7. NEDC-31515, Rev. 1, "Maximum Extended Load Line Limit and Feedwater Heater Out-of-Service Analysis for Enrico Fermi Atomic Power Plant Unit 2," August 1989.

8. NEDC-31843, "Maximum Extended Operating Domain Analysis for Detroit Edison Company Enrico Fermi Energy Center Unit 2," July 1990.

9. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

10. TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses, NEDE-32906P-A, Revision 3, September 2006.

11. Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients, NEDE-32906P Supplement 3-A, Rev. 1, April 2010.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limits specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for AOOs, plus an allowance for densification power spiking.

LHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during the limiting AOOs (Ref. 1). Flow dependent LHGR limits are determined using the three dimensional BWR simulator code (Ref. 4) to analyze slow flow runout transients. The flow dependent multiplier, $LHGRFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the recirculation scoop tube mechanical stop in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multiplier, $LHGRFAC_p$, is also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closure and turbine control valve fast closure scram trips are bypassed (≅ 30% RTP), both high and low core flow $LHGRFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent standard LHGR limits are reduced by $LHGRFAC_p$ and $LHGRFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and AOOs. A complete discussion of the analysis codes are provided in References 5,6, and 7.

The LHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LHGR is a basic assumption in the fuel design and transient analyses. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limits to accomplish this objective are specified in the COLR and are determined by multiplying the smaller of the $LHGRFAC_p$ and $LHGRFAC_f$ factors times the exposure dependent standard LHGR limits.

BASES

REFERENCES

1. UFSAR, Chapter 15.
2. UFSAR, Chapter 4.
3. NEDO-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).
4. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
5. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.
6. TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses, NEDE-32906P-A, Revision 3, September 2006.
7. Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients, NEDE-32906P Supplement 3-A, Rev. 1, April 2010.

BASES

LCO (continued)

result of, or coincident with the loss of power generated by the nuclear power unit. The 345 kV breaker alignment must be maintained such that this criteria continues to be met. The GDC-17 criteria are met when either BM or DF breaker is closed when the main generating unit is online. An example of not meeting the criteria is with both BM and DF breakers open, a main generator trip would open breakers CM and CF and cause a loss of the 345 kV preferred power source. Thus, an offsite circuit must be declared inoperable when the breaker alignment is such that a loss of the main generator could lead to a loss of the respective offsite circuit.

Each EDG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This sequence must be accomplished within 10 seconds. Each EDG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions, such as EDG in standby with the engine hot and EDG in standby with the engine at ambient condition. Additional EDG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the EDG to revert to standby status upon restoration of offsite power.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for EDG OPERABILITY.

The AC sources must be separate and independent (to the extent possible) of other AC sources as described in UFSAR Sections 8.2 and 8.3 (Ref. 2).

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 1, 2, and 3 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 4 and 5 are covered in LCO 3.8.2, "AC Sources-Shutdown."

BASES

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for supported systems that have a support system LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the plant is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.11, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a

BASES

LCO 3.0.6 (continued)

result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

The following examples use Figure B 3.0-1 to illustrate loss of safety function conditions that may result when a TS support system is inoperable. In this figure, the fifteen systems that comprise Division I are independent and redundant to the fifteen systems that comprise Division II. To correctly use the figure to illustrate the SFDP provisions for a cross division check, the figure establishes a relationship between support and supported systems as follows: the figure shows System 1 as a support system for System 2 and System 3; System 2 as a support system for System 4 and System 5; and System 4 as a support system for System 8 and System 9. Specifically, a loss of safety function may exist when a support system is inoperable and:

a. A system redundant to system(s) supported by the inoperable support system is also inoperable (EXAMPLE B 3.0.6-1),

b. A system redundant to system(s) in turn supported by the inoperable supported system is also inoperable (EXAMPLE B 3.0.6-2), or

c. A system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable (EXAMPLE B 3.0.6-3).

For the following examples, refer to Figure B 3.0-1.

EXAMPLE B 3.0.6-1

If System 2 of Division I is inoperable and System 5 of Division II is inoperable, a loss of safety function exists in Systems 5, 10, and 11.

BASES

LCO 3.0.6 (continued)

EXAMPLE B 3.0.6-2

If System 2 of Division I is inoperable, and System 11 of Division II is inoperable, a loss of safety function exists in System 11.

EXAMPLE B 3.0.6-3

If System 2 of Division I is inoperable, and System 1 of Division II is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10 and 11.

If an evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

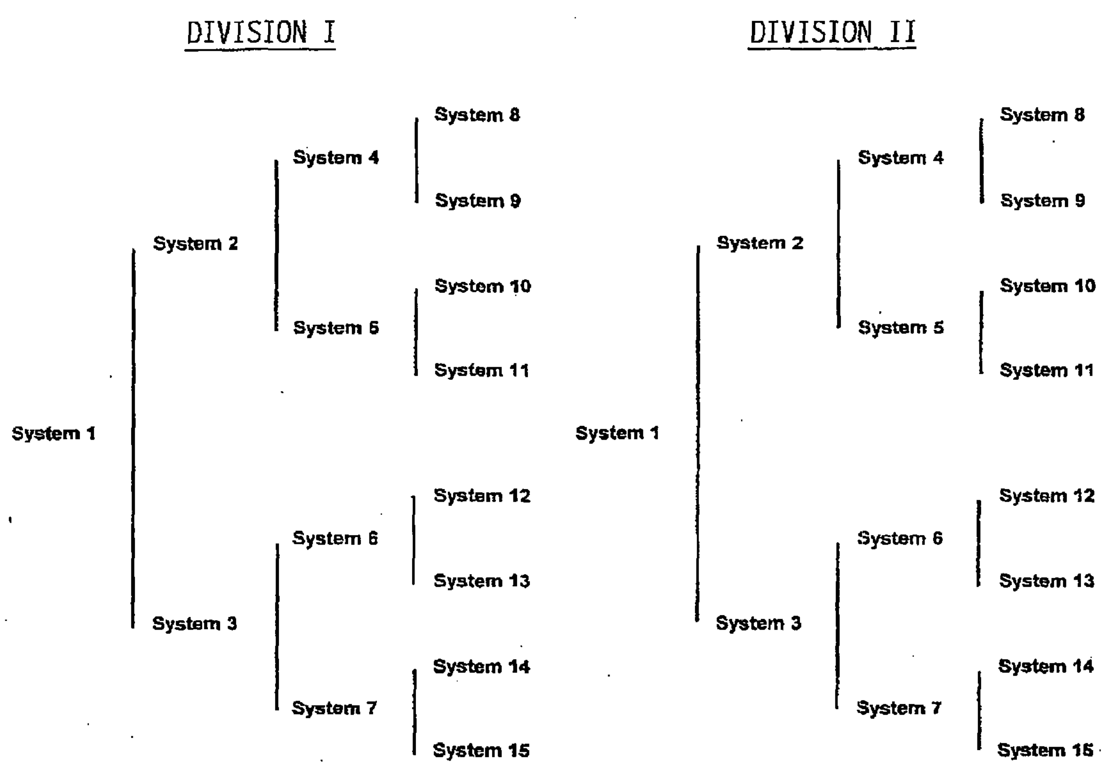


Figure B 3.0-1
Configuration of Divisions and Systems

BASES

ACTIONS

A Note has been provided to modify the ACTIONS related to PAM instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable PAM instrumentation channels provide appropriate compensatory measures for separate Functions. As such, a Note has been provided that allows separate Condition entry for each inoperable PAM Function.

A.1

When one or more Functions have one required channel that is inoperable, the required inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channels (or, in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

If a channel has not been restored to OPERABLE status in 30 days, this Required Action specifies initiation of action in accordance with Specification 5.6.7, which requires a written report to be submitted to the NRC. This report discusses the cause of the inoperability and identifies proposed restorative actions. |

BASES

SURVEILLANCE REQUIREMENTS (continued)

increase, or flow increase.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.10.4 and SR 3.4.10.6 is to compare the temperatures of the operating recirculation loop and the idle loop.

These SRs have been modified by Notes that require the Surveillance to be performed only in certain MODES. In MODE 5, the overall stress on limiting components is lower. Therefore, ΔT limits are not required for SRs 3.4.10.3 and 3.4.10.4 in MODE 5. In MODES 3, 4, and 5, THERMAL POWER increases are not possible, and recirculation flow increases will not result in additional stresses. Therefore ΔT limits are only required for SRs 3.4.10.5 and 3.4.10.6 in MODES 1 and 2. The Notes also state that the SR is only required to be met during the event of concern (e.g., pump startup, power increase or flow increase) since this is when the stresses occur.

SR 3.4.10.7, SR 3.4.10.8, and SR 3.4.10.9

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits during system heatup and cooldown. However, operations approaching MODE 4 from MODE 5 and in MODE 4 with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

The flange temperatures must be verified to be above the limits 30 minutes before and while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When in MODE 4 with RCS temperature ≤ 80° F, 30 minute checks of the flange temperatures are required because of the reduced margin to the limits. When in MODE 4 with RCS temperature ≤ 100°F, monitoring of the flange temperature is required every 12 hours to ensure the temperature is within limits.

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

BASES

APPLICABILITY (continued)

a. During operations with a potential for draining the reactor vessel (OPDRVs); and

b. During movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, the Control Center AC System is only required to be OPERABLE during fuel handling involving recently irradiated fuel. "Recently irradiated fuel" is fuel that has occupied part of a critical reactor core within the previous 6.3 days. Handling new (non-irradiated) fuel bundles over the open reactor core or the spent fuel pool is subject to the same requirements of handling recently irradiated fuel, as long as any fuel in the core or fuel pool is recently irradiated.

ACTIONS

A.1

With one control center AC subsystem inoperable, the inoperable control center AC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE control center AC subsystem is adequate to perform the control center air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the control center air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining subsystem can provide the required protection, and the availability of alternate safety and nonsafety cooling methods.

B.1 and B.2

If both Control Center AC subsystems are inoperable, the Control Center AC System may not be capable of performing its intended function. Therefore, the control room area temperature is required to be monitored to ensure that temperature is being maintained low enough that equipment in

**ENCLOSURE 5 TO**
**NRC-12-0067**

**Summary of Excessive Detail Removed from the Fermi 2 UFSAR**

# Summary Report of Excessive Detail Removal

The following is a summary report of excessive detail that has been removed in Revision 18 to the Fermi 2 UFSAR. These changes were made under the guidance of NEI 98-03, and endorsed by the NRC. The Fermi 2 UFSAR continues to adequately describe the design bases, plant safety analyses, and design and operation of Structures, Systems, and Components (SSCs).

| LCR No. | Excessive Detail Removed | Basis for Removal |
|---|---|---|
| 09-064 | Emergency Facility Augmentation Time information in Sections 7.8.3 and 7.8.4.1 | The change to the UFSAR eliminates redundancy and excessive detail. The information is available in the RERP Plan. |
| 10-044 | Excessive details about roofing material for the Turbine Building, Auxiliary Building and Reactor Building in Figure 1.2-19 | This level of detail is not required on the figure or in any section of the UFSAR. |
| 12-013 | Alternate ID for valve B2103F019 in UFSAR Table 6.2-2 | The valve is identified by the primary ID, B2103F019. |

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