RAI
3.7.2-2

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the Main Steam Line Break (MSLB) inside containment, as discussed in the FSAR, Section 14.18 (Ref. 2). It is also influenced by the accident analysis of the MSLB events presented in the FSAR, Section 14.14 (Ref. 3). There are three different limiting cases, one for fuel integrity and two for containment analysis (one for containment temperature and one for containment pressure). The limiting case for ~~fuel integrity and~~ containment temperature is the hot full power MSLB inside containment following a turbine trip. At hot full power, the stored energy in the primary coolant is maximized. With the most reactive control rod assumed stuck in the fully withdrawn position, there is an increased possibility that the core will return to power. The core is ultimately shut down by a combination of doppler feedback, steam generator dryout, and borated water injection delivered by the Emergency Core Cooling System.

ed
MSLB
INSERT
that have been evaluated
ed

and fuel integrity

The limiting case for the containment analysis for containment pressure is the hot zero power MSLB inside containment. At zero power, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Reverse flow due to the open MSIV bypass valves, contributes to the total release of the additional mass and energy.

The accident analysis compares several different MSLB events against different acceptance criteria. The MSLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The MSLB inside containment at hot full power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following a turbine trip.

With offsite power available, the primary coolant pumps continue to circulate coolant through the steam generators, maximizing the Primary Coolant System (PCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Safety Injection (HPSI) pumps, is delayed.

2-a

INSERT

The MSIVs are swing disc check valves. The inherent characteristic of this type of valve allows for reverse flow through the valve on a differential pressure even if the valve is closed. In the event of an MSLB, if the MSIV associated with the unaffected steam generator fails to close, both steam generators may blowdown. This failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed. The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a double steam generator blowdown event, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modifications would not provide a cost beneficial improvement to plant safety.

SECTION 3.7

INSERT 1

.....assuming the normally closed MSIV bypass valves are closed. The MSIV bypass valves do not receive an isolation signal and might be open during zero power conditions.

RAI - 3.7.2-2

INSERT 2

ed

There are three different limiting MSLB cases ~~,~~ that have been evaluated, one for fuel integrity and two for containment analysis (one for containment temperature and one for containment pressure). The limiting case for ~~fuel integrity and~~ containment temperature is the hot full power MSLB inside containment following a turbine trip. At hot full power, the stored energy in the primary coolant is maximized. With the most reactive control rod assumed stuck in the fully withdrawn position, there is an increased possibility that the core will return to power. The core is ultimately shut down by a combination of doppler feedback, steam generator dry out, and borated water injection delivered by the Emergency Core Cooling System. ed

moved to INSERT 1 on Page B 3.7-8

The MSIVs are swing disc check valves. The inherent characteristic of this type of valve allows for reverse flow through the valve on a differential pressure even if the valve is closed. In the event of an MSLB, if the MSIV associated with the unaffected steam generator fails to close, both steam generators may blowdown. This failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed. The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a double steam generator blowdown event, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modifications would not provide a cost beneficial improvement to plant safety.

NRC REQUEST:

3.7.3 Main Feedwater Isolation Valves (MSIVs) [and [MFIV] bypass valves]
3.7.3-1 CTS 4.2, Table 4.2.2, Item 15
ITS 3.7.3, Applicability, Required Action A.1, and Bases Background
JFD #1, #4, and #10

Comment: (Contractor comment 3.7.3-1, issue #1) The Bases Background presents the MFRV and MFRV Bypass Valves configuration as a design which is clearly different from the standard design assumed in the STS. The fact that the MFRVs are non-safety and not in safety grade locations should be discussed in the ITS Bases.

Consumers Energy Response:

The Background discussion in the Bases for ITS 3.7.3 has been revised to state that the MFRVs and MFRV Bypass valves are non-safety grade valves located on non-safety grade piping.

Affected Submittal Pages:

Att 2, ITS 3.7.3, page B 3.7.3-1
Att 5, NUREG 3.7.3, page B 3.7-13 insert

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves

BASES

BACKGROUND

The MFRVs and MFRV bypass valves in conjunction with feed pump speed, control Main Feedwater (MFW) flow to the steam generators for level control during normal plant operation. The valves also isolate MFW flow to the secondary side of the steam generators following a High Energy Line Break (HELB). Closure of the MFRVs and MFRV bypass valves terminates flow to both steam generators. Closure of the MFRV and MFRV bypass valve effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for Main Steam Line Breaks (MSLBs) inside containment, and reducing the cooldown effects.

The MFRVs and MFRV bypass valves isolate MFW ~~in~~ to the event of a secondary side pipe rupture inside containment limit the quantity of high energy fluid that enters containment through the break. Controlled addition of Auxiliary Feedwater (AFW) is provided by a separate piping system. cd X

One MFRV and one MFRV bypass valve are located on each MFW line outside containment. The piping volume from the valves to the steam generator must be accounted for in calculating mass and energy releases following an MSLB.

The MFRVs and MFRV bypass valves close on receipt of a isolation signal generated by either; steam generator low pressure from its respective steam generator, or containment high pressure. These isolation signals also actuate the Main Steam Isolation Valves (MSIVs) to close. The MFRVs and MFRV bypass valves may also be actuated manually. ~~The MFRVs and MFRV bypass valves fail "as is" on a loss of air. However, only the MFRVs are equipped with a handwheel for local operation. In addition to the MFRVs and MFRV bypass valves, a check valve outside containment is available to isolate the feedwater line penetrating containment.~~ RAI 3.7.3-1 & RAI 3.7.3-2

A description of the MFRVs and MFRV bypass valves is found in the FSAR, Section 10.2.3 (Ref. 1).

Insert

INSERT

The MFRVs and MFRV Bypass valves are non-safety grade valves located on non-safety grade piping that fail "as-is" on a loss of air. If required, MFW isolation can be accomplished using manually operated valves located upstream or downstream of the MFRVs and MFRV Bypass valves. In addition, each MFRV is equipped with a handwheel that can be used to isolate this MFW flowpath.

SECTION 3.7

INSERT 1

.....and MFRV bypass valves in conjunction with feed pump speed, control Main Feedwater (MFW) flow to the steam generators for level control during normal plant operation. The valves also

INSERT 2

X RAI 3.7.3-1 & RAI 3.7.3-2

~~The MFRVs and MFRV Bypass valves fail "as-is" on a loss of air. However, only the MFRVs are equipped with a handwheel for local operation.~~

The MFRVs and MFRV Bypass valves are non-safety grade valves located on non-safety grade piping that fail "as-is" on a loss of air. If required, MFW isolation can be accomplished using manually operated valves located upstream or downstream of the MFRVs and MFRV Bypass valves. In addition, each MFRV is equipped with a handwheel that can be used to isolate this MFW flowpath.

***NRC REQUEST*:**

3.7.3-2 CTS 4.2, Table 4.2.2, Item 15
ITS 3.7.3, Applicability, Required Action A.1, and Bases Background
JFD #1, #4, and #10

Comment: (Contractor comment 3.7.3-1, issue #2) The STS is based upon redundant isolation valves in the flow path, in addition to the main feedwater regulating, control or bypass valves on a closed system to containment (three valves in the flow path). The Bases needs additional discussion of the fact that manual valves typically are relied on to isolate the flow paths and that for main feedwater the valves providing containment isolation are check valves.

***Consumers Energy Response*:**

The Background discussion in the Bases for ITS 3.7.3 has been revised to explain that if necessary, main feedwater isolation can be accomplished using manually operated valves located upstream or downstream of the MFRVs and MFRV Bypass valves.

The safety analysis described in the ISTS differs from the safety analysis at Palisades. Specifically, the Background discussion in the Bases for ISTS 3.7.3 discusses the availability of a check valve inside containment used to isolate the feedwater line penetrating containment. In the ISTS, the MFIVs and MFIV Bypass valves are credited in the FWLB analysis. As such, in the event of an FWLB outside containment, the check valves in the feedwater lines provide a leak tight barrier between the steam generators and the ruptured feedwater line outside containment.

In Palisade's safety analysis, an FWLB is not separately analyzed but rather, it is bounded by the Steam Line Rupture Incident, Loss of External Load Event, and Loss of Normal Feedwater Event. Thus, all information in the Bases of ISTS 3.7.3 pertaining to an FWLB has been removed from the Bases of ITS 3.7.3. The check valves in the feedwater lines near the containment penetration perform a containment isolation function, and as such, are addressed by ITS 3.6.3, "Containment Isolation Valves."

Affected Submittal Pages:

Att 2, ITS 3.7.3, page B 3.7.3-1
Att 2, ITS 3.7.3, page B 3.7.3-2
Att 5, NUREG 3.7.3, page B 3.7-13 insert
Att 5, NUREG 3.7.3, page B 3.7-14

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves

BASES

BACKGROUND

The MFRVs and MFRV bypass valves in conjunction with feed pump speed, control Main Feedwater (MFW) flow to the steam generators for level control during normal plant operation. The valves also isolate MFW flow to the secondary side of the steam generators following a High Energy Line Break (HELB). Closure of the MFRVs and MFRV bypass valves terminates flow to both steam generators. Closure of the MFRV and MFRV bypass valve effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for Main Steam Line Breaks (MSLBs) inside containment, and reducing the cooldown effects.

The MFRVs and MFRV bypass valves isolate MFW ~~in~~ to the event of a secondary side pipe rupture inside containment limit the quantity of high energy fluid that enters containment through the break. Controlled addition of Auxiliary Feedwater (AFW) is provided by a separate piping system. cd x

One MFRV and one MFRV bypass valve are located on each MFW line outside containment. The piping volume from the valves to the steam generator must be accounted for in calculating mass and energy releases following an MSLB.

The MFRVs and MFRV bypass valves close on receipt of a isolation signal generated by either; steam generator low pressure from its respective steam generator, or containment high pressure. These isolation signals also actuate the Main Steam Isolation Valves (MSIVs) to close. The MFRVs and MFRV bypass valves may also be actuated manually. ~~The MFRVs and MFRV bypass valves fail "as is" on a loss of air. However, only the MFRVs are equipped with a handwheel for local operation. In addition to the MFRVs and MFRV bypass valves, a check valve outside containment is available to isolate the feedwater line penetrating containment.~~ RAI 3.7.3-1 & RAI 3.7.3-2

Insert

A description of the MFRVs and MFRV bypass valves is found in the FSAR, Section 10.2.3 (Ref. 1).

INSERT

The MFRVs and MFRV Bypass valves are non-safety grade valves located on non-safety grade piping that fail "as-is" on a loss of air. If required, MFW isolation can be accomplished using manually operated valves located upstream or downstream of the MFRVs and MFRV Bypass valves. In addition, each MFRV is equipped with a handwheel that can be used to isolate this MFW flowpath.

BASES

Containment response analysis

APPLICABLE SAFETY ANALYSES

Closure an assumption in

~~The design basis~~ of the MFRVs is ~~established by~~ the ~~analysis for the~~ MSLB. Closure of the MFRVs and MFRV bypass valves ~~may also be relied on to mitigate a steam break for core response analysis.~~ is also assumed in the MSLB core response (DNB) analysis.

Failure of an MFRV or MFRV bypass valve to close following a MSLB can result in additional mass and energy to the steam generators contributing to cooldown. This failure also results in additional mass and energy releases following an MSLB event.

RAI 3.7.3-5 X

Insert RAI 3.7.3-2

The MFRVs and MFRV bypass valves satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

This LCO ensures that the MFRVs and MFRV bypass valves will isolate MFW flow to the steam generators following an MSLB. This LCO requires that both MFRVs and both MFRV bypass valves be OPERABLE. The MFRVs and MFRV bypass valves are considered OPERABLE when the isolation times are within limits, and are closed on an isolation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an MSLB inside containment.

APPLICABILITY

All MFRVs and MFRV bypass valves must be OPERABLE, or either closed and deactivated, or isolated by closed manually actuated valves, whenever there is significant mass and energy in the Primary Coolant System and steam generators.

In MODES 1, 2, and 3, the MFRVs or MFRV bypass valves are required to be OPERABLE, except when both MFRVs and both MFRV bypass valves are either closed and deactivated, or isolated by closed manually actuated valves, in order to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are either closed and deactivated, or isolated by closed manually actuated valves, they are already performing their safety function.

INSERT

However, this failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed. The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a single steam generator blowdown event with continued feedwater, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modification would not provide a cost beneficial improvement to plant safety.

SECTION 3.7

INSERT 1

.....and MFRV bypass valves in conjunction with feed pump speed, control Main Feedwater (MFW) flow to the steam generators for level control during normal plant operation. The valves also

X RAI 3.7.3-1 & RAI 3.7.3-2

INSERT 2

~~The MFRVs and MFRV Bypass valves fail "as-is" on a loss of air. However, only the MFRVs are equipped with a handwheel for local operation.~~

The MFRVs and MFRV Bypass valves are non-safety grade valves located on non-safety grade piping that fail "as-is" on a loss of air. If required, MFW isolation can be accomplished using manually operated valves located upstream or downstream of the MFRVs and MFRV Bypass valves. In addition, each MFRV is equipped with a handwheel that can be used to isolate this MFW flowpath.

BASES

BACKGROUND (continued)

AND MFRV BYPASS VALVES

A description of the MFIVs is found in the FSAR, Section [10.~~4.7~~] (Ref. 1). 1 2.3

MSLB containment response analysis
an assumption in

RAI 3.7.3-5

APPLICABLE SAFETY ANALYSES

Closure 8 2 MITIGATE

~~The design basis~~ of the MFIVs is ~~established by~~ the analysis for the ~~large~~ SLB. ~~It is also influenced by the accident analysis for the large FWLB.~~ Closure of the MFIVs and ~~their~~ bypass valves ~~may~~ also ~~be~~ relied on to ~~terminate~~ a ~~steam break for core response~~ analysis ~~and an excess feedwater flow event upon receipt of a MSIS on high steam generator level~~.

MFRV

assumed in the MSLB core response (DNB) analysis 6

OR MFRV 2

8 M

Failure of an MFIV ~~and the~~ bypass valve to close following an SLB~~, FWLB, or excess feedwater flow event~~ can result in 4 additional mass and energy to the steam generators contributing to cooldown. This failure also results in additional mass and energy releases following an SLB ~~or FWLB~~ event.

M 8

INSERT RAI 3.7.3-2

2

The MFIVs satisfy Criterion 3 of the ~~NRC Policy Statement~~.

AND MFRV BYPASS VALVES 10 CFR 50.36(c)(2)

BOTH BOTH MFRV

LCO

2 8 M

This LCO ensures that ~~the~~ MFIVs and ~~the~~ bypass valves will isolate MFW flow to the steam generators, following an ~~FWLB or~~ SLB. ~~these valves will also isolate the nonsafety related portions from the safety related portions of the system.~~ This LCO requires that [~~two~~] MFIVs and [~~MFIV~~] bypass valves ~~in each feedwater line~~ be OPERABLE. The MFIVs and ~~the~~ bypass valves are considered OPERABLE when the isolation times are within limits, and are closed on ~~an~~ isolation ~~actuation~~ signal.

10 BOTH 1 MFRV 4 10 BOTH MFRV

8 M 6

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB ~~or FWLB~~ inside containment. ~~If an MSIS on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.~~

MFRV EITHER

APPLICABILITY

The MFIVs and ~~the~~ bypass valves must be OPERABLE whenever there is significant mass and energy in the ~~Reactor~~ Coolant

4 PRIMARY

(continued)

2 , CLOSED AND DEACTIVATED, OR ISOLATED BY CLOSED MANUALLY ACTUATED VALVES,

5-f

SECTION 3.7

INSERT

However, this failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed. The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a single steam generator blowdown event with continued feedwater, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modifications would not provide a cost beneficial improvement to plant safety.

***NRC REQUEST*:**

3.7.3-3 CTS 4.2, Table 4.2.2, Item 15
ITS 3.7.3, Applicability, Required Action A.1, and Bases Background
JFD #1, #4, and #10

Comment: (Contractor comment 3.7.3-1, issue #3) The ITS Applicability is modified to isolate the feedwater line with a "manually actuated valve" which is different from the STS text of "a closed manual valve." The valve difference is not explained and no reason given why this new wording is required. Provide this explanation.

***Consumers Energy Response*:**

"Manually actuated" power operated valves are typically used to isolate the MFRV flow paths rather than "manual valves." These valves are air operated gate valves (CV-0742 and CV-0744) located a few feet down stream of the MFRVs (CV-0701 and CV-0703). They are controlled from the control room. Many manual valves would also be available to isolate the flow path if needed. The air operated gate valves do not isolate the MFRV bypass flow paths. Manual valves would be used if those flow paths needed to be isolated. The existing degree of isolation for the main feedwater lines has previously been found acceptable by the NRC as transmitted in their Safety Evaluation of February 28, 1986, titled "Main Steam Line Breaks - Single Failures."

***Affected Submittal Pages*:**

No page changes.

***NRC REQUEST*:**

3.7.3-4 CTS 4.2, Table 4.2.2, Item 15
ITS SR 3.7.3.1 and Bases
DOC M.2 and JFD #1

Comment: (Contractor comment 3.7.3-3) ITS SR 3.7.3.1 appears to be acceptable; however, the specific basis for the closure time of 22 seconds should be stated in the SR Bases.

***Consumers Energy Response*:**

The Bases of ITS SR 3.7.3.1 has been revised to clarify that MFRV closure times are bounding values assumed in the MSLB containment response and core response (DNB) analyses. In addition, a reference to the appropriate FSAR Section has been included.

***Affected Submittal Pages*:**

Att 2, ITS B 3.7.3, page B 3.7.3-4
Att 5, NUREG B 3.7.3, page B 3.7-17

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies the closure time for each MFRV and MFRV bypass valve is ≤ 22.0 seconds on an actual or simulated actuation signal. Specific signals (e.g., steam generator low pressure and containment high pressure) are tested under Section 3.3, "Instrumentation." The MFRV and MFRV bypass valves closure times are bounding values assumed in the MSLB ~~and~~ containment response and core response (DNB) analyses (Refs. 3 and 4). This SR is normally performed upon returning the plant to operation following a refueling outage. The MFRVs and MFRV bypass valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the plant generating power. As these valves are not stroke tested at power, they are exempt from the ASME Code, Section XI (Ref. 2) requirements during operation in MODES 1 and 2. RAI 3.7.3-4 x x

The Frequency is 18 months. The 18 month Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency.

REFERENCES

1. FSAR, Section 10.2.3
2. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400 RAI 3.7.3-4
3. FSAR, Section 14.18.2 X
4. FSAR, Section 14.14 X

RAI 3.7.3-4

BASES

response and core response (DNB)

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 (continued)

AND MFRV BYPASS VALVES 2

ARE bounding Values

actuation signal. The MFIV closure time ~~is~~ assumed in the ~~accident~~ and containment analyses. This ~~Surveillance~~ is SR normally performed upon returning the ~~unit~~ to operation following a refueling outage. The MFIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the ~~unit~~ generating power. As these valves are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 2) requirements during operation in MODES 1 and 2.

MSLB

(Refs. 3 and 4)

PLANT 4

2 STROKE

The Frequency is ~~in accordance with the [Inservice Testing Program or~~ [18] months]. The [18] month Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the SR when performed at the [18] month Frequency.

1 2

REFERENCES

1. FSAR, Section ~~[10.4.7]~~. 10.2.3 1
2. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400.
3. FSAR, Section 14.18.2
4. FSAR, Section 14.14

RAI 3.7.3-4

2

2 SPECIFIC SIGNALS (E.G., STEAM GENERATOR LOW PRESSURE AND CONTAINMENT HIGH PRESSURE) ARE TESTED UNDER SECTION 3.3, "INSTRUMENTATION."

7-b

NRC REQUEST:

3.7.3-5 ITS Bases Page 3.7.3-2

Comment: The Applicable Safety Analysis states "Closure of the MFRVs and ... may also be relied on" (emphasis added). Are they or not? That statement does not appear consistent with the LCO discussion which only addresses MSLB.

Consumers Energy Response:

The ITS Applicable Safety Analyses has been revised to clarify that MFRV closure is an initial assumption used in the MSLB analysis, and that valve closure is also relied on to mitigate a steam break for core response analysis. This change establishes consistency with the LCO Bases discussion which states "...valves will isolate MFW flow to the steam generators following an MSLB."

Affected Submittal Pages:

Att 2, ITS B 3.7.3, page B 3.7.3-2
Att 5, NUREG B 3.7.3, page B 3.7-14

BASES

Containment response analysis

APPLICABLE SAFETY ANALYSES

Closure   an assumption in

~~The design basis~~ of the MFRVs is ~~established by~~ the ~~analysis for the~~ MSLB. ~~Closure of the MFRVs and MFRV bypass valves may also be relied on to mitigate a steam break for core response analysis.~~ is also assumed in the MSLB core response (DNB) analysis. RAI 3.7.3-5 X

Failure of an MFRV or MFRV bypass valve to close following a MSLB can result in additional mass and energy to the steam generators contributing to cooldown. This failure also results in additional mass and energy releases following an MSLB event.

Insert
RAI 3.7.3-2

The MFRVs and MFRV bypass valves satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

This LCO ensures that the MFRVs and MFRV bypass valves will isolate MFW flow to the steam generators following an MSLB. This LCO requires that both MFRVs and both MFRV bypass valves be OPERABLE. The MFRVs and MFRV bypass valves are considered OPERABLE when the isolation times are within limits, and are closed on an isolation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an MSLB inside containment.

APPLICABILITY

All MFRVs and MFRV bypass valves must be OPERABLE, or either closed and deactivated, or isolated by closed manually actuated valves, whenever there is significant mass and energy in the Primary Coolant System and steam generators.

In MODES 1, 2, and 3, the MFRVs or MFRV bypass valves are required to be OPERABLE, except when both MFRVs and both MFRV bypass valves are either closed and deactivated, or isolated by closed manually actuated valves, in order to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are either closed and deactivated, or isolated by closed manually actuated valves, they are already performing their safety function.

INSERT

However, this failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed. The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a single steam generator blowdown event with continued feedwater, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modification would not provide a cost beneficial improvement to plant safety.

BASES

AND MFRV BYPASS VALVES

BACKGROUND (continued)

A description of the MFIVs is found in the FSAR, Section [10.~~4.7~~] (Ref. 1).

1 2.3

MSLB containment response analysis

an assumption in

RAI 3.7.3-5

Closure 8 2 MITIGATE

APPLICABLE SAFETY ANALYSES

The ~~design basis~~ of the MFIVs is ~~established by~~ the analysis for the ~~large~~ SLB. ~~It is also influenced by the accident analysis for the large FWLB.~~ Closure of the MFIVs and ~~their~~ bypass valves ~~may~~ also be relied on to terminate a ~~steam break for core response~~ analysis and an excess feedwater flow event upon receipt of a MSIS on high steam generator level.

MFRV

assumed in the MSLB core response (DNB) analysis

6 OR MFRV 2

8 M 4

Failure of an MFIV ~~and the~~ bypass valve to close following an SLB~~, FWLB, or excess feedwater flow event~~ can result in additional mass and energy to the steam generators contributing to cooldown. This failure also results in additional mass and energy releases following an SLB ~~or FWLB~~ event.

INSERT

M 8

RAI 3.7.3-2

The MFIVs satisfy Criterion 3 of the ~~NRC Policy Statement~~.

2 AND MFRV BYPASS VALVES 10 CFR 50.36(c)(2)

BOTH BOTH MFRV

LCO

2 8 M

This LCO ensures that ~~the~~ MFIVs and ~~the~~ bypass valves will isolate MFW flow to the steam generators. ~~following an FWLB or SLB, these valves will also isolate the nonsafety related portions from the safety related portions of the system.~~

10 3or MFRV 1

This LCO requires that [~~two~~] MFIVs, [and [~~MFIV~~] bypass valves ~~in each feedwater line~~ be OPERABLE. The MFIVs and ~~the~~ bypass valves are considered OPERABLE when the isolation times are within limits, and are closed on ~~an~~ isolation ~~actuation~~ signal.

4 10 BOTH MFRV

8 M 6

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB ~~or FWLB~~ inside containment. ~~If an MSIS on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.~~

MFRV EITHER

APPLICABILITY

The MFIVs and ~~the~~ bypass valves must be OPERABLE whenever there is significant mass and energy in the ~~Reactor~~ Coolant

4 PRIMARY

(continued)

2

, CLOSED AND DEACTIVATED, OR ISOLATED BY CLOSED MANUALLY ACTUATED VALVES,

8-c

SECTION 3.7

INSERT

However, this failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed. The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a single steam generator blowdown event with continued feedwater, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modifications would not provide a cost beneficial improvement to plant safety.

NRC REQUEST:

3.7.3-6 ITS Bases Page 3.7.3-3

Comment: The MFRVs could clearly be closed in less than 8 hrs. The reason 8 hours is the allotted time is to get the plant to the point where conditions support closing the valves. The STS Bases language that was not adopted explained that. That language or something similar would appear to be appropriate.

Consumers Energy Response:

The Bases discussion for ITS 3.7.3 Required Actions A.1 and A.2 has been revised to include language similar to the ISTS that supports the 8 hours Completion Time for closing inoperable MFRVs or MFRV Bypass valve.

Affected Submittal Pages:

Att 2, ITS B 3.7.3, page B 3.7.3-3
Att 5, NUREG B 3.7.3, page B 3.7-15

BASES

APPLICABILITY (continued)

Once the valves are closed, deactivation can be accomplished by the removal of the motive force (e.g., electrical power, air) to the valve to prevent valve opening. Closing another manual valve in the flow path either remotely (i.e., control room hand switch) or locally by manual operation satisfies isolation requirements.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFRVs and MFRV bypass valves are not required to be OPERABLE.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one MFRV or MFRV bypass valve inoperable, action must be taken to close or isolate the inoperable valve(s) within 8 hours. When these valve(s) are closed or isolated, they are performing their required safety function (e.g., to isolate the line). RAI 3.7.3-6

The 8 hour Completion Time is reasonable to close the MFRV or MFRV bypass ~~valve.~~ valve, which includes performing a controlled plant shutdown to a condition that supports isolation of the affected valve(s)

B.1 and B.2

If the MFRVs or MFRV bypass valves cannot be restored to OPERABLE status, closed, or isolated in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

APPLICABILITY (continued)

System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. 4

10 BOTH MFRV AND BOTH MFRV BYPASS VALVES ARE EITHER, 1

In MODES 1, 2, and 3, the MFIV [or [MFIV] bypass valves] are required to be OPERABLE, except when they are closed and deactivated or isolated by a closed manual valve(s) in order to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and deactivated or isolated by a closed manual valve(s), they are already performing their safety function. 4 LY ACTUATED 4 EITHER 4 LY ACTUATED 4

2 INSERT

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs and the bypass valves are normally closed since MFW is not required. MFRV 2 TO BE OPERABLE

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve. V

A.1 and A.2

With one MFIV or the bypass valve inoperable, action must be taken to close or isolate the inoperable valve(s) within [8 or 72] hours. When these valve(s) are closed or isolated, they are performing their required safety function (e.g., to isolate the line). MFRV 2 1 2

RAI 3.7.3-6

For units with only one MFIV per feedwater line: The [8] hour Completion Time is reasonable to close the MFIV or its bypass valve, which includes performing a controlled unit shutdown to MODE 2. 2 1 MFRV 9

which includes performing a controlled plant shutdown to a condition that supports isolation of the affected valve(s)

The [72] hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves, and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. 4

B.1

If more than one MFIV or [MFIV] bypass valve in the same flow path cannot be restored to OPERABLE status, then there 4

(continued)

9-b

***NRC REQUEST*:**

3.7.4 Atmospheric Dump Valves (ADVs)
3.7.4-1 ITS Bases Page 3.7.4-1

Comment: The Background (as well as the FSAR) state that N2 is not required for operability of the ADVs. What is the technical basis for those statements?

***Consumers Energy Response*:**

The following FSAR excerpt provides the technical basis for the N_2 supply to the ADVs:

"There is also a nitrogen backup system installed to provide backup of the instrument air system to the Main Steam System's Atmospheric Dump Valves (ADVs). The nitrogen backup system provides 90 psig nitrogen to the ADVs provided by the 230 psig bulk nitrogen system. The bulk nitrogen system was utilized as the source of the nitrogen rather than a bottle system due to the large number of bottles that would have been required to provide the four hour coping duration necessary to meet Station Blackout requirements. Use of the bulk nitrogen system allows the ADVs to exceed the Station Blackout coping duration requirements of 10 CFR 50.63 as recommended in Reg Guide 1.155. See Section 8.1.5 for an additional description of Palisades' response to Station Blackout requirements. The sizing of the pressure regulator provides sufficient nitrogen to fully open all four ADVs simultaneously. There is excess capacity for throttling the ADVs for the four hour coping duration. Nitrogen backup to the ADVs is not considered safety related per Reg Guide 1.155 and is not required for the operability of the ADVs. However, the nitrogen backup is required to support post-fire safe shutdown (Appendix R)."

Furthermore, as discussed in JFD #7 for Specification 3.7.4, the power supplies to the instrument air compressors can be connected to the emergency diesel generators in the event of a loss of offsite power to the station. This assures a source of motive air will be available to the ADVs in the event of an accident accompanied by a loss of offsite power.

***Affected Submittal Pages*:**
No page changes.

***NRC REQUEST*:**

3.7.4-2 ITS Bases Pages 3.7.4.1 & 2
JFD #8

Comment: The justification in the JFD for not having a surveillance for the block valve is that no credit is taken for the manual isolation valve in the safety analysis. Following a SGTR, if one ADV is being used to cool the plant and the ADV on the other generator spuriously opens, can the total release be tolerated (assuming the block valve cannot be used to isolate the spuriously open valve)?

***Consumers Energy Response*:**

Yes, the analysis of the radiological consequences for an SGTR event considers the most severe release of secondary as well as primary system activity leaked from the ruptured tube. The SGTR analysis and associated radiological consequences assumes that the faulted S/G is steamed to control level throughout the cooldown. The inventory of fission product activity available for release to the environment is a function of the primary to secondary coolant leakage rate, the assumed fission product concentration, and the mass of the steam discharged to the environment. For an SGTR event concurrent with a loss of offsite power, failure of the ADV on the affected SG to close (or as the result of a spurious reopening) would not result in an increase in the radiological consequences from those previously assumed in the safety analysis.

***Affected Submittal Pages*:**

No page changes.

NRC REQUEST:

3.7.5	Auxiliary Feedwater (AFW) System
3.7.5-1	CTS 3.5.3 ITS 3.7.5 Required Action C.2 DOC L.2 and JFD #1

Comment: (Contractor comment 3.7.5-3, issue #2) The CTS markup indicates that DOC L.2 adds Condition C when CTS 3.5.3 already contains the required Actions C.1 and C.2, as shown on the CTS markup. This apparent contradiction and the CTS markup should be revised to delete the "Adds Condition C".

Consumers Energy Response:

Consumers Energy agrees with the above comment. CTS markup page 3-38a will be revised to delete the notation that Condition C was added.

Affected Submittal Pages:

Att 3, CTS, page 3-38a (ITS 3.7.5, page 2 of 4)

7.5 AUXILIARY FEEDWATER (AFW) SYSTEM

3.5 ~~STEAM AND FEEDWATER SYSTEMS (Cont'd)~~ A.1

~~3.5.2 (Continued)~~

⟨SEE 3.7.6⟩

~~c. The fire water makeup to the Auxiliary Feedwater Pump Suction (P-8A and P-8B) may be inoperable for a period of 7 days provided the pump service water makeup to P-8C, pump P-8C, and its corresponding flow control valves are operable.~~

~~d. The service water makeup to the Auxiliary Feedwater Pump Suction (P-8C) may be inoperable for a period of 7 days provided the fire water makeup to P-8A & P-8B, pumps P-8A and P-8B and their corresponding flow control valves are operable.~~

COND B e. One ~~flow control valve on each~~ train may be inoperable for a period of 72 hours provided the ~~corresponding redundant flow control valve and a pump in the other pipe~~ train ~~are~~ IS operable. L.4

APPL 3.5.3 With the Primary Coolant System at a temperature greater than 300°F and if the system does not satisfy the requirements of Specification 3.5.1 or the conditions of Specification 3.5.2 except as noted in Specification 3.5.4, the reactor shall be placed ~~in hot standby within 6 hours, hot shutdown~~ within ~~the following~~ 6 hours and in ~~cold shutdown~~ within the following 24 hours. M.1

RA C.1

RA C.2 M.3

A.5 MODE 3 A.8 ⟨INSERT⟩ MODE 4 L.2

3.5.4 With all Auxiliary Feedwater Pumps inoperable immediately initiate corrective action to restore at least one Auxiliary Feedwater Pump to OPERABLE status as soon as possible ~~and reduce power within 24 hours to the lowest stable power level consistent with reliable Main Feedwater System operation.~~ L.5

COND D

RA D.1

~~⟨ADD COND C⟩ L.2~~

RAI 3.7.5-1

⟨ADD RA D1 NOTE⟩ L.5

Amendment No. ~~96~~, 161
August 12, 1994

NRC REQUEST:

3.7.5-2 CTS 3.5.4
ITS 3.7.5 Condition D
DOC A.8

Comment: (Contractor comment 3.7.5-8) DOC A.8 provides an explanation that "...or flow paths..." are added which does not match the ITS Condition D markup. This apparent inconsistency should be eliminated or further explained in a revised DOC or CTS/ITS markup, as required.

Consumers Energy Response:

Doc A.8 has been revised to eliminate the conflict.

Affected Submittal Pages:

Att 3, DOC 3.7.5, page 2 of 7

A.6 A Note has been added to SR 3.7.5.3 which states "Not required to be met in MODES 2 or 3 when AFW is in operation." This Note is needed to prevent unnecessary entering of ACTIONS for LCO 3.7.5 during the startup or shutdown of the plant for not being able to meet the SR. Palisades uses the AFW system for steam generator level control during startup and shutdown in MODES 2, 3, and 4. During these operations the flow control valves used are in manual, and will not open automatically when an actuation signal is received, which would fail the SR. This change is administrative because CTS 4.9.b.1 states "each valve to actuates to its correct position (or that the specified flow is established) upon receipt of a simulated auxiliary feedwater pump start signal." During startup or shutdown the valves are providing the proper flow for the existing plant condition. This Note is appropriate because the valves are needed to be throttled in these conditions to prevent overfilling of the steam generators due to low steam flow conditions, also the Note clarifies current licensing basis requirements.

A.7 This change adds the additional "inservice requirements" as described in ASME Code, Section XI to CTS 4.9.a.1 and 2. This change is administrative in that these requirements are performed by current surveillances and also this change only combines the two requirements, Code and TS. This change is consistent with NUREG-1432.

A.8 ~~CTS 3.5.4 requires immediate corrective action to restore AFW in the event that all AFW pumps are inoperable. Proposed ITS Condition D has this same requirement but adds "or flow paths" to the Condition. This change is considered administrative in that the addition clarifies that the same capability is lost when there are no AFW flow paths as with no AFW pumps. This change maintains consistency between NUREG-1432.~~

Insert

RAI 3.7.5-2

INSERT

CTS 3.5.4 provides corrective actions in the event all AFW pumps are inoperable. In this case, the capability to provide the required AFW flow to either steam generator has been lost. Propose ITS 3.7.5 Condition D also provides corrective actions when the capability to provide the required AFW flow to either steam generator has been lost. Condition D is stated as "two AFW trains inoperable with both steam generators having less than 100% of the AFW flow equivalent to a single Operable AFW train available in Mode 1, 2, or 3 or (the) required AFW train inoperable in Mode 4." Since the AFW system inoperability addressed in ITS 3.7.5 Condition D (a loss of AFW function) is equivalent to the condition presented in CTS 3.5.4, this change has been characterized as administrative in nature.

NRC REQUEST:

3.7.5-3 CTS 3.5.4
ITS 3.7.5 Condition D, Note to Required Action D.1
JFD #12

Comment: (Contractor comment 3.7.5-10) JFD #12 appears to be acceptable; however, this change should be submitted as a generic STS traveler for approval by the Owners Group and the NRC.

Consumers Energy Response:

Consumers Energy will pose such a TSTF change at the next CE Owners Group Technical Specifications subcommittee meeting.

Affected Submittal Pages:

No page changes.

NRC REQUEST:

3.7.5-5 ITS LCO 3.7.5

Comment: The way the CTS is constructed, only one AFW pump could be inoperable at a time meaning that Pump C, which has slightly less performance capability, would never be operable by itself. However, now with the switch to an AFW " train" approach, it is possible that Pump C could be operable by itself which appears to constitute a less restrictive change.

Consumers Energy Response:

Neither the CTS, nor the ITS allow continued operations with only one Operable AFW pump. If two AFW Pumps are inoperable in Modes 1, 2, or 3, proposed ITS Condition C requires the plant to be placed in Mode 4 within 30 hours. In Mode 4 only one AFW pump is required to be Operable if the SGs are relied upon for heat removal (this is a new requirement for Palisades). In Mode 4 the maximum average primary coolant temperature is less than 300°F which corresponds to a saturation pressure of 67 psia in the secondary side of the SGs. At this pressure, AFW pump "C" has adequate discharge capacity to fulfill the required heat removal function.

Affected Submittal Pages:

No page changes.

NRC REQUEST:

3.7.6 Condensate Storage Tank (CST)
3.7.6-1 ITS Bases 3.7.6
ITS LCO 3.7.6

Comment: The backup water supplies that are required to be verified in the LCO should be discussed in the Bases. FSAR section 9.7.4, #4 states that the primary water storage tank is a backup, when it is in fact required in the TS.

Consumers Energy Response:

Required Action A.1 of ITS 3.7.6 requires a verification that the backup water supplies are Operable whenever the condensate volume is not within limit. The Bases discussion for Required Action A.1 has been revised to clearly indicate that the backup water supply is from the Fire Water System and Service Water System.

The safety analysis assumes that approximately 100,000 gallons of AFW is required to remove decay heat for 8 hours following a reactor trip. FSAR Section 9.7.4 item 4 states that the condensate storage tank capacity is 125,000 gallons. However, only about 72,000 gallons of the 125,000 gallons are available to supply the suction of the AFW pumps. Therefore, to ensure an adequate supply of condensate is available, LCO 3.7.6 requires that a combined useable volume of ≥ 100,000 gallons be available from both the Condensate Storage Tank and Primary Makeup Storage Tank.

Affected Submittal Pages:

Att 2, ITS B 3.7.6, page B 3.7.6-3
Att 5, NUREG B 3.7.6, page B 3.7-34

BASES

ACTIONS

A.1 and A.2

If the condensate volume is not within the limit, the OPERABILITY of the backup water supplies must be verified by administrative means within 4 hours and once every 12 hours thereafter.

RAI 3.7.6-1

OPERABILITY of the backup feedwater supplies must include verification of the OPERABILITY of flow paths from the backup supplies to the AFW pumps, and availability of the water in the backup supplies. The Condensate Storage and Supply volume must be returned to OPERABLE status within 7 days, as the backup supplies may be performing this function in addition to their normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supplies every 12 hours is adequate to ensure the backup water supplies continue to be available. The 7 day Completion Time is reasonable, based on OPERABLE backup water supplies being available, and the low probability of an event requiring the use of the water from the CST and T-81 occurring during this period.

Fire Water System and SWS

X

X

B.1 and B.2

If the condensate volume cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generator for heat removal, within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

LCO (continued) 8

~~Systems Branch 8-1 (Ref. 4) and exceeds the volume required by the accident analysis.~~

CONDENSATE STORAGE AND SUPPLY COMBINED 5

OPERABILITY of the ~~CST~~ is determined by maintaining ~~the tank~~ level at or above the minimum required ~~level~~.
5 VOLUME

APPLICABILITY In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the ~~CST~~ is required to be OPERABLE. 1
CONDENSATE STORAGE AND SUPPLY 4

In MODES 5 and 6, the ~~CST~~ is not required because the AFW System is not required.

RAI 3.7.6-1

ACTIONS A.1 and A.2 CONDENSATE VOLUME 5

If the ~~CST level~~ is not within the limit, the OPERABILITY of the backup water ~~supply~~ must be verified by administrative means within 4 hours AND ONCE EVERY 12 HOURS THEREAFTER. IES 6

Fire Water System and SWS

OPERABILITY of the backup feedwater ~~supply~~ must include verification of the OPERABILITY of flow paths from the backup ~~supply~~ to the AFW pumps, and availability of the ~~required volume of~~ water in the backup ~~supply~~. The ~~CST~~ ~~level~~ must be returned to OPERABLE status within 7 days, as the backup ~~supply~~ may be performing this function in addition to ~~its~~ normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. The 7 day Completion Time is reasonable, based on ~~an~~ OPERABLE backup water ~~supply~~ being available, and the low probability of an event requiring the use of the water from the CST occurring during this period.

CONDENSATE 5 5 VOLUME 6 IES 2 THEIR 9 <INSERT> 6 IES AND T-81 5

B.1 and B.2 5 CONDENSATE VOLUME OR THE BACKUP SUPPLY

If the ~~CST~~ cannot be restored to OPERABLE status within the associated Completion Time, the ~~unit~~ must be placed in a MODE in which the LCO does not apply. To achieve this status, the ~~unit~~ must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generator for heat removal, within ~~[18]~~ 30 hours. The allowed Completion

4 PLANT 1

(continued)

16-b

NRC REQUEST:

3.7.7 Component Cooling Water (CCW) System
3.7.7-1 CTS 3.4.2 and 3.4.3
ITS 3.7.7 Required Action B.2
DOC M.4

Comment: (Contractor comment 3.7.7-3, issue #2) The markup of CTS 3.3.2 appears incorrect because it shows a total of 60 hours (6+30+24) to place the plant in Cold Shutdown rather than the ITS (and STS) of 36 hours. Explain this difference. Revise the CTS markup.

Consumers Energy Response:

Consumers Energy agrees, the markup of CTS 3.3.2 has been revised. The words "within 24 hours" have been stricken.

Affected Submittal Pages:

Att 3, CTS 3.7.7, page 3-29a (ITS 3.7.7, page 2 of 4)

M.1

3.3 EMERGENCY CORE COOLING SYSTEM (Cont'd) A.1

3.3.2 During power operation the requirements of 3.3.1 may be modified to allow one of the following conditions to be true at any one time. If the system is not restored to meet the requirements of 3.3.1 within the time period specified below, the reactor shall be placed in a hot shutdown condition within 12 hours. 6 If the requirements of 3.3.1 are not met within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours. 30 M.4

RAB.1 M.3 RAB.2 MODE 3 M.3 MODE 5 A.4

RAI 3.7.7-1

SEE 3.5

a. One safety injection tank may be inoperable for a period of no more than one hour.

b. One low-pressure safety injection pump may be inoperable provided the pump is restored to operable status within 24 hours.

c. One high-pressure safety injection pump may be inoperable provided the pump is restored to operable status within 24 hours.

SEE 3.5

d. One shutdown heat exchanger and one component cooling water heat exchanger may be inoperable for a period of no more than 24 hours. M.6

SEE 3.5

e. Any valves, interlocks or piping directly associated with one of the above components and required to function during accident conditions shall be deemed to be part of that component and shall meet the same requirements as listed for that component.

SEE 3.5

f. Any valve, interlock or pipe associated with the safety injection and shutdown cooling system and which is not covered under 3.3.2e above but, which is required to function during accident conditions, may be inoperable for a period of no more than 24 hours.

PAGE 2 OF 4

Amendment No. ~~21~~, ~~51~~, 172
September 26, 1996

NRC REQUEST:

3.7.7-2 ITS 3.7.7 Bases

Comment: It should be in the TS, or at least in the Bases, that the two CCW heat exchangers are both required to be operable to have an operable train of CCW.

Consumers Energy Response:

LCO 3.7.7 requires two Operable trains of CCW. The Bases LCO discussion defines an Operable CCW train as having both CCW heat exchangers Operable. In addition, the Bases Background discussion also states that both CCW heat exchangers are required for an Operable CCW train.

Affected Submittal Pages:

No page changes.

NRC REQUEST:

3.7.7-3 ITS 3.7.7 Bases

Comment: With the failure of either valve CV-0945 or 46, one CCW heat exchanger would be inoperable and there would be no operable CCW train as both heat exchangers are required for an operable train. If that is true, how can the Bases make the statement that the system can sustain an active single failure?

Consumers Energy Response:

If either valve CV-0945 or CV-0946 were closed, one CCW heat exchanger would be isolated rendering both trains of CCW inoperable. During the plant conditions when CCW is required to be Operable, both CV-0945 and CV-0946 are in the full open position to supply cooling water to normal plant heat loads. If a loss of electrical power or control air occurs, both valves fail in the full open position. In the event of a DBA, CV-0945 and CV-0946 receive a confirmatory open signal from the recirculation actuation instrumentation to assure cooling water is available during the recirculation phase of a LOCA. In this configuration, the function of the CCW System is to provide a gradual reduction in the temperature of the recirculated fluid.

The Bases for the CCW System states, in part, "The CCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power." Since CV-0945 and CV-0946 are required to be maintained in the open position, they do not need to reposition in the event of an accident. As such, an active failure of these valves is not assumed.

Affected Submittal Pages:

No page changes.

***NRC REQUEST*:**

3.7.8 Service Water System (SWS)
No Comments.

***NRC REQUEST*:**

3.7.9 Ultimate Heat Sink (UHS)
No Comments.

***NRC REQUEST*:**

3.7.10 Essential Chilled Water (ECW)
No Comments.

***NRC REQUEST*:**

3.7.11 Control Room Emergency Air Cleanup System (CREACS)
No Comments.

NRC REQUEST:

3.7.12 ~~Control Room Emergency Air Temperature Control system (CREATCS)~~
Fuel Handling Area Ventilation System

3.7.12-1 CTS 3.8.1 and 3.8.4
ITS 3.7.12 LCO Statement and related Bases
No DOC and JFD #4, #5, and #8

Comment: (Contractor comment 3.7.12-1, issue #2) ITS 3.7.12 Bases Background, Insert #2, explains that after the fuel accident occurs, then the operator "aligns the fuel handling building exhaust through the emergency filtration arrangement." This appears to be in contradiction with the LCO statement which does not permit any fuel movement until the system is already exhausting through the emergency filtration arrangement. Correct this apparent error or explain the reason for this statement. Also, explain the operator function to "secure" a component. Is this to lock in place, open or close these components?

Consumers Energy Response:

The Bases Background discussion for Specification 3.7.12 has been revised to eliminate the contradictory statement that in the event of a fuel handling or cask drop accident, operators "align the fuel handling building exhaust through the emergency filtration arrangement." In addition, the revised wording no longer includes the word "secured" as it applies to the status or condition of plant equipment.

Upon further review of Specification 3.7.12, several inconsistency were identified and corrected. A brief discussion of these changes is provided as follows:

SPECIFICATION

LCO

The wording of the LCO has been revised to better describe the functional requirement of the specification. That is, the operating fuel handling area exhaust fan must be aligned to the emergency filter bank.

Consumers Energy Response: (continued)

Actions Table
The Actions Table Note has been deleted as discussed in RAI-3.7.12-2. In addition, the Actions have been revised to address the most probable causes for failure to meet the requirements of the LCO. As such, the Actions address the conditions when the Fuel Handling Area Ventilation system is inoperable, not properly aligned, or not in operation. Since the Fuel Handling Area Ventilation system consists of a single train aligned in its accident mitigation mode, the only appropriate Required Actions upon failure to meet the LCO is to immediately suspend all Core Alterations, and suspend movement of fuel assemblies and the fuel cask.

Surveillance Requirements
ITS SR 3.7.12.1 has been deleted. The intent of this SR is to ensure the standby FBAC system functions properly. For plants that rely on automatic actuation signals, or whose Applicability includes Modes 1, 2, 3, and 4, performance of this SR fulfill the intended function. However, for the Palisades plant, the Fuel Handling Area Ventilation system is required to be in operation whenever the plant is in the condition specified in the Applicability. Since SRs are only required to be met during the condition specified in the Applicability, performance of a system functional test would be redundant to the requirements of the LCO. Therefore, specifying this SR in the ITS is not necessary.

ITS SR 3.7.12.4 has been deleted. The Fuel Handling Area Ventilation system does not include an automatic actuation feature. The system is manually configured in the emergency filtration mode prior to entering the conditions specified in the Applicability. As such, is not necessary to verify the system bypass damper can be cycled.

BASES
Background
Additional detail has been provided relative to the configuration of the Fuel Handling Building Ventilation system including system operation during normal plant evolutions and during fuel handling and cask movement activities. In addition, the contradictory statement related to system alignment by plant operators has been eliminated (RAI 3.7.12-1).

Consumers Energy Response: (continued)

Applicable Safety Analyses
Additional details have been provided relative to the licensing basis of the fuel handling accident previously approved by the NRC staff, and for the fuel cask drop accident.

LCO
Clarifying information relative to the requirements of the LCO has been added. This includes an explanation of why only one fuel handling exhaust fan is required.

Applicability
Clarifying information and an explanation of when the LCO in not applicable (consistent with ISTS format) has been included (RAI 3.7.12-5).

Actions
Conforming changes to reflect the revised Specification.

Surveillance Requirements
Conforming changes to reflect the revised Specification.

ATTACHMENT 3 DOCs

CTS Markup
New markups of CTS pages 3-47 and 4-14 have been provided.

Administrative Changes
Added new DOC A.4.

More Restrictive Changes
Deleted DOCs M.1 and M.3.
Added new DOC M.1.
Renumbered DOC M.4 to M.3.

Less Restrictive Changes-Removal of Details to Licensee Controlled Documents
Deleted DOC LA.1.
Renumbered DOC LA.2 to LA.1 with minor wording changes.

Less Restrictive Changes
Deleted DOC L.1
Added new DOC L.1 (RAI 3.7.12-3)

Consumers Energy Response: (continued)

ATTACHMENT 4 NSHC
Deleted NSHC L.1
Added new NSHC L.1 (RAI 3.7.12-3)

ATTACHMENT 5 NUREG MARKUP
New markups provided.

ATTACHMENT 6 JFDs
Deleted JFD 6, and JFDs 8 through 11.
Renumbered JFD 7 to JFD 6 with minor wording changes.
Added new JFDs 7 through 13 (RAI 3.7.12-4 for JFD 9).

Affected Submittal Pages:

In lieu of detailed markups and to facilitate NRC staff review, all pages related to our initial submittal of Specification 3.7.12 have been superseded by this change. The revised pages are provided in Enclosure 3.

NRC REQUEST:

3.7.12-2 CTS 3.8.4
ITS 3.7.12 Applicability
DOC L.1 and JFD #9

Comment: (Contractor comment 3.7.12-2) The addition of the Action Note is acceptable because the licensee has stated that the Fuel Handling Area Ventilation System does not filter any fission product removal associated with ECCS leakage following an accident. In Modes 1, 2, 3, and 4, the System is independent of reactor operation and is not required to be Operable (that is reflected in the deletion of the Bracketed modes of the ITS markup for Applicability). DOC L.1 justifies the new actions Note based upon how the operator cannot "cease fuel movement" and the need to enter LCO 3.0.3. This appears to be a violation of TS requirements and appears to contradict DOC LA.2. The DOC for this CTS change should be revised. The contents of JFD #9 should be placed in the Bases Applicability discussion, to clearly explain how the Operability of the Fuel Handling Area Ventilation is independent of reactor operations.

Consumers Energy Response:

Consumers Energy agrees. The proposed Actions Table Note and associated justification have been deleted. The Bases Applicability discussion has been revised to clearly state that the Fuel Handling Area Ventilation system is required whenever the potential exists for an accident which could result in damage to irradiated fuel assemblies.

Affected Submittal Pages:

See RAI 3.7.12-1

NRC REQUEST:

3.7.12-3 CTS 4.2 Table 4.2.3, Item 2.c
ITS SR 3.7.12.1
DOC LA.1

Comment: (Contractor comment 3.7.12-3) DOC LA.1 states these SR details are moved to the Bases; when in fact, these details are not retained in the ITS SR 3.7.12 Bases. Revise the CTS/ITS markup as applicable to comply with the technical justification as provided in this DOC.

Consumers Energy Response:

Based on the additional restriction of the ITS which requires the Fuel Handling Area Ventilation System to be in the emergency filtration arrangement whenever the plant is in the specified Mode of Applicability, the requirements of CTS 4.2, Table 4.2.3, Item 2.c is no longer necessary. A new DOC (DOC L.1) has been provided to justify this change. As such, the details presently contained in the CTS are no longer required in the Bases for Specification 3.7.12.

Affected Submittal Pages:

See RAI 3.7.12-1

NRC REQUEST:

3.7.12-4 No CTS requirement
STS SR 3.7.14.3 and Bases
No JFD

Comment: (Contractor comment 3.7.12-5) It is acknowledged that this is not an automatically initiated system; however, the NEI 96-06 guidelines require that all deviations from the STS be justified with a JFD. There is no JFD provided for this STS requirement that is not retained.

Consumers Energy Response:

Consumers Energy agrees. JFD 9 has been provided to describe this deviation from the ISTS.

Affected Submittal Pages:

See RAI 3.7.12-1

NRC REQUEST:

3.7.12-5 ITS LCO 3.7.12

Comment: Where does the 90 days of the Applicability come from?

Consumers Energy Response:

The 90 day Applicability is associated with the fuel cask drop analysis. That analysis has shown when irradiated fuel has decayed for 90 days or greater, the dose rates at the site boundary are well within the guideline of 10 CFR 100 for all analyzed cask drop events without crediting filtration by the Fuel Handling Ventilation system. The Bases discussion for the Applicability and Applicable Safety Analyses have been revised to include this information.

Affected Submittal Pages:

See RAI 3.7.12-1

NRC REQUEST:

3.7.13	ECCS Pump Room Exhaust Air Cleanup System (PREACS)
3.7.13-1	New LCO from CTS Table 3.17.3, Item 4
	ITS 3.7.13 LCO Statement and related Bases
	DOC M.1; JFD #4 and #6

Comment: (Contractor comment 3.7.13-1, issue #2) The ITS markup of the Bases is missing inserts #1, #2, #3, and #4, as identified on page B 3.7-65. Provide the missing documents.

Consumers Energy Response:

The missing page has been provided as part of this response.

Affected Submittal Pages:

Att 5, NUREG, page B 3.7-65 insert

No Changes
Page was missing

RAL 3.7.13-1

SECTION 3.7

INSERT 1

.....isolate the safeguards rooms by closing the inlet and exhaust plenum dampers on the initiation of a high radiation alarm from their respective airborne particulate monitor. This isolation lowers the offsite dose to well within 10 CFR 100 (Ref. 1) limits if a leak should occur. Typically, high radiation would only be expected due to excessive leakage during the recirculation phase of operation following a loss of coolant accident (LOCA).

INSERT 2

.....supply plenum damper, an exhaust plenum damper, a radiation monitor, and associated piping, valves, and duct work.

INSERT 3

.....which is addressed in LCO 3.3.10, "Engineered Safeguards Room Ventilation (ESRV) Instrumentation"

INSERT 4

.....shut, isolating the affected safeguards room(s) from the rest of the auxiliary building ventilation system lowering the leakage to the environment from the auxiliary building.

NRC REQUEST:

3.7.13-2 New LCO from CTS Table 3.17.3, Item 4
ITS 3.7.13 Actions Note
DOC M.1 and JFD #2

Comment: (Contractor comment 3.7.13-2) No specific technical justification is provided to explain the rationale for developing this LCO as "Separate Condition entry" rather than as a two train system as the STS is developed. "Separate Condition entry" is normally used in the STS for individual inoperable components rather than trains. Also, "Separate Condition entry" is used where the number of inoperabilities are more than two. Therefore, this does not appear to be an appropriate usage of the "Separate Condition entry." The resolution will also depend upon the configuration and contents of each ESRV train noted above in Comment #3.7.13-1.

Consumers Energy Response:

Consumers Energy agrees with the above comment. The Action Note specifying that separate condition entry is allowed for each train has been deleted.

Affected Submittal Pages:

Att 1, ITS 3.7.13, page 3.7.13-1
Att 2, ITS B 3.7.13, page B 3.7.13-2
ATT 5, NUREG 3.7.13, page 3.7-29
ATT 5, NUREG B 3.7.13, page B 3.7-67
Att 5, NUREG B 3.7.13, page B 3.7-67 insert

3.7 PLANT SYSTEMS

3.7.13 Engineered Safeguards Room Ventilation (ESRV) Dampers

LCO 3.7.13 Two ESRV Damper trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

RAI 3.7.13-2

ACTIONS

~~-----NOTE-----
Separate Condition entry is allowed for each train.~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ESRV Damper trains inoperable.	A.1 Initiate action to isolate associated ESRV Damper train(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.13.1 Verify each ESRV Damper train closes on an actual or simulated actuation signal.	31 days

BASES

LCO

Two ESRV Damper trains are required to be OPERABLE to ensure that each engineered safeguards room isolates upon receipt of its respective high radiation alarm. Total system failure could result in the atmospheric release from the engineered safeguards rooms exceeding the required limits in the event of a Design Basis Accident (DBA).

An ESRV Damper train is considered OPERABLE when its associated radiation monitor, instrumentation, ductwork, valves, and dampers are OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ESR-Damper trains are required to be OPERABLE consistent with the OPERABILITY requirements of the Emergency Core Cooling System (ECCS).

In MODES 5 and 6, the ESRV Damper trains are not required to be OPERABLE, since the ECCS is not required to be OPERABLE.

ACTIONS

RAI 3.7.13-2

~~A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The condition of this Specification may be entered independently for each train. The Completion Times of each inoperable train will be tracked separately for each train, starting from the time the condition is entered.~~

A.1

Condition A addresses the failure of one or both ESRV Damper trains. Operation may continue as long as action is immediately initiated to isolate the affected engineered safeguards room. With the inlet and exhaust dampers closed, or if the inlet and outlet ventilation plenums are adequately sealed, the engineered safeguards room is isolated and the intended safety function is achieved, since the potential pathway for radioactivity to escape to the environment from the engineered safeguards room has been minimized.

The Completion Time for this Required Action is commensurate with the importance of maintaining the engineered safeguards room atmosphere isolated from the outside environment when the ECCS pumps are circulating primary coolant after an accident.

3.7 PLANT SYSTEMS

4 3.7.13 Engineered Safeguards Room Ventilation (ESRV) Dampers ~~Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)~~

New

LCO 3.7.13 Two ECCS PREACS trains shall be OPERABLE.

RAI 3.7.13-2

APPLICABILITY: MODES 1, 2, 3, and 4.

2 ACTIONS ~~NOTE Separate Condition entry is allowed for each train~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ECCS PREACS trains inoperable. (2) New	A.1 ~~Restore ECCS PREACS train to OPERABLE status.~~ Initiate action to isolate associated ESRV Damper train. (4)	~~7 days~~ Immediately
~~B. Required Action and associated Completion Time not met.~~ (6)	~~B.1 Be in MODE 3.~~	~~6 hours~~
	~~AND~~	
	~~B.2 Be in MODE 5.~~	~~36 hours~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
~~SR 3.7.13.1~~ (6)	~~Operate each ECCS PREACS train for [≥ 10 continuous hours with the heater operating or (for systems without heaters) ≥ 15 minutes].~~	~~31 days~~

(continued)

BASES

LCO (continued)

~~ECCS PREACS is considered OPERABLE when the individual components necessary to maintain the ECCS Pump Room filtration are OPERABLE in both trains.~~

(4) An ECCS PREACS train is considered OPERABLE when its associated ~~:~~ RADIATION MONITOR, INSTRUMENTATION, DUCTWORK, VALVES, AND DAMPERS ARE OPERABLE

(6)

~~a. Fan is OPERABLE;~~

~~b. HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions; and~~

~~c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.~~

(2) APPLICABILITY

In MODES 1, 2, 3, and 4, the ECCS PREACS ~~is~~ TRAINS ARE required to be OPERABLE consistent with the OPERABILITY requirements of the EMERGENCY CORE COOLING SYSTEM (ECCS).

In MODES 5 and 6, the ECCS PREACS ~~is~~ TRAINS ARE not required to be OPERABLE, since the ECCS is not required to be OPERABLE.

ACTIONS (2) ~~<INSERT 1>~~

A.1

<INSERT 2>

RAI 3.7.13-2

(6)

~~With one ECCS PREACS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ECCS PREACS function.~~

~~The 7 day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72 hour Completion Time) and this system is not a direct support system for the ECCS. The 7 day Completion Time is reasonable, based on the low probability of a DBA occurring during this time period, and the consideration that the remaining train can provide the required capability.~~

(continued)

30-d

RAI 3.7.13-2

SECTION 3.7

~~INSERT 1~~

~~A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Condition of this Specification may be entered independently for each train. The Completion Times of each inoperable train will be tracked separately for each train, starting from the time the Condition is entered.~~

INSERT ~~2~~ 1

Condition A addresses the failure of one or both ESRV Damper train(s). Operation may continue as long as action is immediately initiated to isolate the affected engineered safeguards room. With the inlet and exhaust dampers closed or if the inlet and outlet ventilation plenums are adequately sealed, the engineered safeguards room is isolated and the intended safety function is achieved, since the potential pathway for radioactivity to escape to the environment from the engineered safeguards room has been minimized.

The Completion Time for this Required Action is commensurate with the importance of maintaining the engineered safeguards room atmosphere isolated from the outside environment when the ECCS pumps are circulating primary coolant after an accident.

NRC REQUEST:

3.7.14 Fuel Building Air Cleanup System (FBACS)
3.7.14-1 ITS 3.7.14

Comment: Level is greater than or equal to 674 ft relative to what? (above MSL)?

Consumers Energy Response:

In general, reference to various plant elevations throughout the CTS, ITS, FSAR, and other plant documents is relative to "mean sea level" and, as such, is not explicitly stated. Since the level of the Great Lakes is currently reported using International Great Lake Datum, discussions pertaining to the level of Lake Michigan and to external flooding hazards will specify "mean sea level" as appropriate to clearly indicate the correct reference point (i.e., MSL or IGLD).

Affected Submittal Pages:

No page changes.

***NRC REQUEST*:**

3.7.15 Penetration Room Exhaust Air Cleanup System (PREACS)
No comments

***NRC REQUEST*:**

3.7.16 Fuel Storage Pool Water Level
3.7.16-1 New LCO from CTS 5.4.2.c, d, and i; and Table 5.4-1
ITS 3.7.16 LCO statement, SR 3.7.16.1, and Bases
JFD #4

Comment: (Contractor comment 3.7.16-1) JFD #4 contains no specific technical justification for not retaining the requirements that spent fuel storage is in accordance with Specification 4.3.1.1. The Bases discussion of LCO and SR 3.7.16.1 state these requirements are met which is in contradiction with the ITS LCO proposed. Provide explanation and technical justification that resolves this apparent inconsistency.

***Consumers Energy Response*:**

A new JFD (JFD #7) has been provided to explain why proposed SR 3.7.16.1 does not ensure compliance with Specification 4.3.1.1. As such, reference to Specification 4.3.11 in SR 3.7.16.1 can be deleted. Conforming changes have also been made to the Bases to eliminate inconsistency with the actual surveillance requirement.

***Affected Submittal Pages*:**

Att 2, ITS B 3.7.16, page B 3.7.16-2
Att 5, NUREG 3.7.18, page 3.7-39
Att 5, NUREG B 3.7.18, page B 3.7-90
Att 6, JFD 3.7.18, page 1 of 1

BASES

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Table 3.7.16-1, in the accompanying LCO, ensures that the k_{eff} of the spent fuel pool will always remain < 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool according to Table 3.7.16-1, in the accompanying LCO. Fuel assemblies not meeting the criteria of Table 3.7.16-1 shall be stored in accordance with Specification 4.3.1.1.

RAI 3.7.16-3

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region II of either the spent fuel pool ~~and~~ or the north tilt pit.

X

ACTIONS

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

A.1

When the configuration of fuel assemblies stored in Region II the spent fuel pool is not in accordance with Table 3.7.16-1, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Table 3.7.16-1.

SURVEILLANCE REQUIREMENTS

RAI 3.7.16-1

SR 3.7.16.1

Prior to placing the fuel assembly in a Region II storage location,

This SR verifies by administrative ~~means~~ that the initial enrichment and burnup of the fuel assembly is in accordance with Table 3.7.16-1 in the accompanying LCO. ~~For fuel assemblies in the unacceptable range of Table 3.7.16-1, performance of this SR will ensure compliance with Specification 4.3.1.1.~~

3.7.18 = 3.7.16 4

3.7 PLANT SYSTEMS

3.7.~~18~~ 16 Spent Fuel Assembly Storage

CTS 5.4.2.1 2 3 4

LCO 3.7.~~18~~ 16 The combination of initial [Table] enrichment and burnup of each spent fuel assembly stored in [Region ~~2~~ II] shall be within the [requirements] ~~acceptable [burnup domain] of Figure~~ [Table] 3.7.~~18~~ 16-1. ~~[or in accordance with Specification 4.3.1.1]~~

New

APPLICABILITY: 1 2 4 Whenever any fuel assembly is stored in [Region ~~2~~ II] of [either] the spent fuel storage pool [or the north tilt pit].

RAI 3.7.16-3

ACTIONS 2

New

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met. 1 2	A.1 --------NOTE-------- LCO 3.0.3 is not applicable. ----------------------- Initiate action to move the noncomplying fuel assembly from [Region ~~2~~ II].	Immediately

SURVEILLANCE REQUIREMENTS

CTS 5.4.2d

SURVEILLANCE	FREQUENCY
SR 3.7.~~18~~ 16.1 3 4 7 6 1 Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with ~~Figure~~ Table 3.7.~~18~~ 16-1 ~~or Specification 4.3.1.1~~.	Prior to storing the fuel assembly in [Region ~~2~~ II]

RAI 3.7.16-1

32-b

BASES (continued)

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in [Region 2] of the spent fuel pool.

either II or the and the north tilt pit

ACTIONS

RAI 3.7.16-3

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in [Region 2] the spent fuel pool is not in accordance with Figure [3.7.18-1], immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure [3.7.18-1].

II Table Table

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.18.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure [3.7.18-1] in the accompanying LCO. For fuel assemblies in the unacceptable range of Figure 3.7.18-1, performance of this SR will ensure compliance with Specification 4.3.1.1.

Table 16

REFERENCES

None.

prior to placing the fuel assembly in a Region II storage location.

RAI 3.7.16-1

< INSERT >

32-C

Change	Discussion

Note: This attachment provides a brief discussion of the deviations from NUREG-1432 that were made to support the development of the Palisades Nuclear Plant ITS. The Change Numbers correspond to the respective deviation shown on the "NUREG MARKUPS." The first five justifications were used generically throughout the markup of the NUREG. Not all generic justifications are used in each specification.

1. The brackets have been removed and the proper plant specific information or value has been provided.

2. Deviations have been made for clarity, grammatical preference, or to establish consistency within the Improved Technical Specifications. These deviations are editorial in nature and do not involve technical changes or changes of intent.

3. The requirement/statement has been deleted since it is not applicable to this facility. The following requirements have been renumbered, where applicable, to reflect this deletion.

4. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the facility specific nomenclature, number, reference, system description, or analysis description.

5. This change reflects the current licensing basis/technical specification.

6. The storage of failed fuel is accomplished by the use of canisters that fit in the same storage racks as the fuel assemblies themselves. Therefore, the storage pool does not have any specifically designed rack(s) for failed fuel. The reference to a specific number of storage locations for failed fuel is deleted.

RAI
3.7.16-1
X

7. INSERT

INSERT

ISTS 3.7.18 applies to plants which restrict the storage of fuel assemblies in high density storage locations based on meeting an acceptable combination of initial enrichment and discharge burnup. For fuel assemblies which do not meet the initial enrichment and discharge burnup requirements, the assemblies may be stored in compliance with other NRC approved methods or configurations as stipulated in ISTS 4.3.1.1. ISTS SR 3.7.18.1 requires an administrative verification of the initial enrichment and discharge burnup of a fuel assembly prior to storing any assembly in a Region 2 location. For the Palisades Plant, storage of fuel assemblies in high density racks (Region II) is only permitted for fuel assemblies which meet the initial enrichment and discharge burnup requirements. Alternate storage methods or configurations (e.g., checkerboading) in Region II has not been approved by the NRC. Therefore, reference to storage of fuel assemblies in accordance with Specification 4.3.1.1 in the LCO, SR, and SR Bases has been deleted. Assurance that fuel assembly enrichments do not exceed the limits of Region I locations (ITS 4.3.1.1) is controlled administratively in the design of new cores and the procurement of new fuel.

***NRC REQUEST*:**

3.7.16-2 CTS 5.4.2.c and d Bases for ITS 3.7.16
No DOC

Comment: (Contractor comment 3.7.16-5) The movement of these CTS requirements to a location under licensee control must be justified with a DOC as required by NEI 96-06. Provide the necessary technical justification in a "LA" DOC and revise the CTS markup as required.

***Consumers Energy Response*:**

CTS page 5-4a has been provided only to show that a new specification (ITS 3.7.16) has been added. As denoted on this page, the requirements of CTS 5.4.2c and CTS 5.4.2d are addressed in proposed Specification 4.3. The addition of Specification 3.7.16 is justified in DOC M.1.

***Affected Submittal Pages*:**

No page changes.

NRC REQUEST:

3.7.16-3 ITS 3.7.16 Applicability

Comment: The Applicability would be much clearer if it was written as Region II, of either the SFP or the north tilt pit. The present version could be read as Region II of the SFP or anywhere in the north tilt pit.

Consumers Energy Response:

Consumers Energy agrees with the above comment. The Applicability has been revised as suggested.

Affected Submittal Pages:

Att 1, ITS 3.7.16, page 3.7.16-1
Att 2, ITS B 3.7.16, page B 3.7.16-2
Att 5, NUREG 3.7.18, page 3.7-39
Att 5, NUREG B 3.7.18, page B 3.7-90

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Assembly Storage

RAI 3.7.16-3

LCO 3.7.16 The combination of initial enrichment and burnup of each spent fuel assembly stored in Region II shall be within the requirements of Table 3.7.16-1.

APPLICABILITY: Whenever any fuel assembly is stored in Region II of either the spent fuel pool ~~and~~ or the north tilt pit.

X

ACTIONS

-----------------------------------NOTE-----------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to move the noncomplying fuel assembly from Region II.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Table 3.7.16-1.	Prior to storing the fuel assembly in Region II

BASES

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Table 3.7.16-1, in the accompanying LCO, ensures that the k_{eff} of the spent fuel pool will always remain < 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool according to Table 3.7.16-1, in the accompanying LCO. Fuel assemblies not meeting the criteria of Table 3.7.16-1 shall be stored in accordance with Specification 4.3.1.1.

RAI 3.7.16-3

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region II of either the spent fuel pool ~~and~~ or the north tilt pit.

X

ACTIONS

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

A.1

When the configuration of fuel assemblies stored in Region II the spent fuel pool is not in accordance with Table 3.7.16-1, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Table 3.7.16-1.

SURVEILLANCE REQUIREMENTS

RAI 3.7.16-1

SR 3.7.16.1

This SR verifies by administrative ~~means~~ that the initial enrichment and burnup of the fuel assembly is in accordance with Table 3.7.16-1 in the accompanying LCO prior to placing the fuel assembly in a Region II storage location. ~~For fuel assemblies in the unacceptable range of Table 3.7.16-1, performance of this SR will ensure compliance with Specification 4.3.1.1.~~

3.7.18 = 3.7.16 4

3.7 PLANT SYSTEMS

3.7.18 16 Spent Fuel Assembly Storage

CTS 5.4.2.1 2 3

RAI 3.7.16-1

LCO 3.7.18 16 4 7 — The combination of initial enrichment and burnup of each spent fuel assembly stored in [Region 2] II shall be within the requirements 5 acceptable [burnup domain] of [Figure] Table 3.7.18 16-1 [or in accordance with Specification 4.3.1.1].

New 1 2 either

APPLICABILITY: 4 — Whenever any fuel assembly is stored in [Region 2] II of either the spent fuel storage pool or the 670 north tilt pit.

RAI 3.7.16-3

ACTIONS 2

New

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met. 1 2	A.1 --------NOTE-------- LCO 3.0.3 is not applicable. ------------------------ Initiate action to move the noncomplying fuel assembly from [Region 2] II.	Immediately

SURVEILLANCE REQUIREMENTS

CTS 5.4.2.d

SURVEILLANCE	FREQUENCY
SR 3.7.18 16.1 3 4 6 1 7 — Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with [Figure] Table 3.7.18 16-1 [or Specification 4.3.1.1].	Prior to storing the fuel assembly in [Region 2] II

RAI 3.7.16-1

-34-C

BASES (continued)

APPLICABILITY	This LCO applies either whenever any fuel assembly is stored in [Region 2] of the spent fuel pool. II 2 or the ~~and~~ the north tilt pit 4
ACTIONS	A.1 RAI 3.7.16-3 X Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. When the configuration of fuel assemblies stored in [Region 2] the spent fuel pool is not in accordance with ~~Figure~~ Table [3.7.18-1], immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with ~~Figure~~ Table [3.7.18-1]. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.
SURVEILLANCE REQUIREMENTS	SR 3.7.18.1 This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with ~~Figure~~ Table [3.7.18-1] in the accompanying LCO. ~~For fuel assemblies in the unacceptable range of Figure 3.7.18-1, performance of this SR will ensure compliance with Specification 4.3.1.1.~~
REFERENCES	None. prior to placing the fuel assembly in a Region II storage location. X

RAI 3.7.16-1

< INSERT >

34-d

***NRC REQUEST*:**

3.7.17 Fuel Storage Pool Boron Concentration
3.7.17-1 CTS 4.2, Table 4.2.1, Item #7
ITS SR 3.7.17.1
DOC L.1

Comment: (Contractor comment 3.7.17-3) The removal of this CTS requirement appears acceptable; however, the DOC L.1 explains this CTS change but does not provide a specific technical justification for why this CTS requirement can be deleted. Provide this missing justification in a revision to the DOC.

***Consumers Energy Response*:**

DOC L.1 has been revised to provide additional justification for the deletion of CTS 4.2, Table 4.2.1, Item #7.

***Affected Submittal Pages*:**

Att 3, DOC 3.7.17, page 2 of 2
Att 4, NSHC 3.7.17, page 1 of 2

A.5 CTS 3.1.5c requires that with specific activity of the secondary coolant >0.1 μCi/gram DOSE EQUIVALENT I-131, the plant must be placed in COLD SHUTDOWN. In proposed ITS the term is replaced with MODE 5 (see DOC A.4). In proposed ITS 3.7.17 Applicability, the Specification is applicable in MODES 1, 2, 3, and 4. Placing the plant in COLD SHUTDOWN in CTS and having the Applicability in MODES 1, 2, 3, and 4 in proposed ITS is basically the same. This change is considered to be an administrative change since the effect on operations is similar. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE (M)

M.1 CTS 4.2 Table 4.2.1, item 7a, requires the specific activity of the secondary coolant system to be determined once per 31 days whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit, and once per 6 months whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. Proposed ITS SR 3.7.17.1 will require the specific activity to be determined once per 31 days. The proposed ITS SR will not contain the allowance to extend the SR interval to 6 months whenever the gross activity determination indicates iodine concentration below 10% of the allowable limit. This change does not adversely affect safety because the 31 day interval ensures that the specific activity is checked frequently enough to establish a trend to identify secondary activity problems in a timely manner. Deleting an allowance to extend an SR interval constitutes a more restrictive change. This change is consistent with NUREG-1432.

LESS RESTRICTIVE CHANGES - REMOVAL OF DETAILS TO LICENSEE CONTROLLED DOCUMENTS (LA)

There were no "Removal of Details" associated with this specification.

LESS RESTRICTIVE CHANGES (L)

RAI 3.7.17-1

L.1 ~~CTS 4.2, Table 4.2.7 requires a sample of secondary coolant be analyzed for gross radioactivity 3 times every 7 days with a maximum of 72 hours between samples. This requirement has been deleted. The CTS contains no LCO, limiting value, or Required Actions associated with this requirement in CTS, only that sampling is required. This change is considered Less Restrictive because this sampling requirement is deleted. This change is consistent with NUREG-1432.~~

INSERT

CTS 4.2, Table 4.2.1 requires a sample of secondary coolant to be analyzed for gross radioactivity 3 times every 7 days with a maximum of 72 hours between samples. The CTS contains no LCO, limiting value, or Required Actions for secondary coolant gross radioactivity, only that sampling is required. The intent of this surveillance is to monitor the iodine concentration in the secondary coolant in order to determine the frequency at which an isotopic analysis for Dose Equivalent I-131 concentration in the secondary coolant is performed. The CTS requires an isotopic analysis for Dose equivalent I-131 of the secondary coolant once per 31 days whenever the gross activity indicates iodine concentrations greater than 10% of the allowable limit or, once per 6 months whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. However as discussed in DOC M.1 for this specification, the extended surveillance interval of 6 months for the determination of Dose Equivalent I-131 in the secondary coolant has been proposed for deletion and that future testing be performed every 31 days. Thus, the need to perform sampling of the secondary coolant for gross radioactivity is no longer necessary and has been delete in the ITS. This change is acceptable since gross radioactivity in the secondary coolant is not evaluated for radiological consequences in any of the accidents assumed in the FSAR, and the concentration of the Dose Equivalent I-131 in the secondary coolant will continue to be determined at an appropriate frequency. In addition, radiation monitoring instrumentation, controlled in accordance with the Offsite Dose Calculation Manual (e.g., SG blowdown monitors and condenser off gas monitor), is available to monitor increases in the radioactivity levels in the secondary coolant. This change is consistent with NUREG-1432.

LESS RESTRICTIVE CHANGE L.1

New →
RAI
3.7.17-1

CTS 4.2, Table 4.2.7 requires a sample of secondary coolant be analyzed for gross radioactivity 3 times every 7 days with a maximum of 72 hours between samples. This requirement has been deleted. The CTS contains no LCO, limiting value, or Required Actions associated with this requirement in CTS, only that sampling is required. This change is considered Less Restrictive because this sampling requirement is deleted. This change is consistent with NUREG-1432.

1. **Does the change involve a significant increase in the probability or consequence of an accident previously evaluated?**

   Analyzed events are assumed to be initiated by the failure of plant structures, systems, or components. The proposed change deletes the sample requirement for gross radioactivity of the secondary coolant. This sample does not have a detrimental impact on the integrity of any plant structure, system, or component. Deletion of this sample requirement will not alter the operation of any plant equipment, or otherwise increase its failure probability. As such, the probability of occurrence for a previously analyzed accident is not significantly increased.

   The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event. Gross radioactivity of the secondary coolant is not an initial condition input assumed for any analyzed event. The amount of Dose Equivalent I-131 in the secondary coolant is the assumed parameter. The limit requirement for Dose Equivalent I-131 remains unchanged and the sampling requirement has become more restrictive (see M.1). The deletion of the gross radioactivity sampling requirement does not affect the assumptions of an analyzed event. This change does not affect the performance of any credited equipment since the sample requirement is for an unassumed parameter. As a result, no analysis assumptions are violated. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

35-c

INSERT

CTS 4.2, Table 4.2.1 requires a sample of secondary coolant to be analyzed for gross radioactivity 3 times every 7 days with a maximum of 72 hours between samples. The CTS contains no LCO, limiting value, or Required Actions for secondary coolant gross radioactivity, only that sampling is required. The intent of this surveillance is to monitor the iodine concentration in the secondary coolant in order to determine the frequency at which an isotopic analysis for Dose Equivalent I-131 concentration in the secondary coolant is performed. The CTS requires an isotopic analysis for Dose equivalent I-131 of the secondary coolant once per 31 days whenever the gross activity indicates iodine concentrations greater than 10% of the allowable limit or, once per 6 months whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. However as discussed in DOC M.1 for this specification, the extended surveillance interval of 6 months for the determination of Dose Equivalent I-131 in the secondary coolant has been proposed for deletion and that future testing be performed every 31 days. Thus, the need to perform sampling of the secondary coolant for gross radioactivity is no longer necessary and has been delete in the ITS. This change is acceptable since gross radioactivity in the secondary coolant is not evaluated for radiological consequences in any of the accidents assumed in the FSAR, and the concentration of the Dose Equivalent I-131 in the secondary coolant will continue to be determined at an appropriate frequency. In addition, radiation monitoring instrumentation, controlled in accordance with the Offsite Dose Calculation Manual (e.g., SG blowdown monitors and condenser off gas monitor), is available to monitor increases in the radioactivity levels in the secondary coolant. This change is consistent with NUREG-1432.

ENCLOSURE 2

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO JANUARY 26, 1999
REQUEST FOR ADDITIONAL INFORMATION

EDITORIAL CHANGES

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. UHS inoperable.	A.1 Be in MODE 3. <u>AND</u> A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify water level of UHS is ≥ ~~571.0~~ 568.25 ft above mean sea level.	24 hours
SR 3.7.9.2	Verify water temperature of UHS is ≤ 81.5°F.	24 hours

X
Tech Change

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Primary Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction through a common suction line from the Condensate Storage Tank (CST) (LCO 3.7.6, "Condensate Storage and Supply") and pump to the steam generator secondary side via two separate and independent flow paths to a common AFW supply header for each steam generator. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the Main Steam Safety Valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") or Atmospheric Dump Valves (ADVs) (LCO 3.7.4, "Atmospheric Dump Valves (ADVs)"). If the main condenser is available, steam may be released via the turbine bypass valve.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into two trains. One train (A/B) consists of a motor driven pump (P-8A) and the turbine driven pump (P-8B) in parallel, the discharges join together to form a common discharge. The A/B train common discharge separates to form two flow paths, which feed each steam generator via each steam generators AFW penetration. The second motor driven pump (P-8C) feeds both steam generators through separate flow paths via each steam generator AFW penetration and forms the other train (C). The two trains join together at each AFW penetration to form a common supply to the steam generators. Each AFW pump is capable of providing 100% of the required capacity to the steam generators as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Ed X

Each motor driven AFW pump is powered from an independent Class 1E power supply, and feeds both steam generators.

BASES

BACKGROUND (continued)

The steam turbine driven AFW pump receives steam from either main steam header upstream of the Main Steam Isolation Valve (MSIV). Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump. The steam supply from steam generator E-50A receives an open signal from the Auxiliary Feedwater Actuation Signal (AFAS) instrumentation. The steam supply from steam generator E-50B does not. This steam source is a manual backup. The turbine driven AFW pump feeds both steam generators through the same flow paths as motor driven AFW pump P-8A.

One pump at full flow is sufficient to remove decay heat and cool the plant to Shutdown Cooling (SDC) System entry conditions.

The AFW System supplies feedwater to the steam generators during normal plant startup, shutdown, and hot standby conditions.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs, with exception of AFW pump P-8C. If AFW pump P-8C is used, operator action may be required to either trip two of four Primary Coolant Pumps (PCPs), start an additional AFW pump, or reduce steam generator pressure. This will allow the required flowrates to the steam generators that are assumed in the safety analyses. Subsequently, the AFW System supplies sufficient water to cool the plant to SDC entry conditions, and steam is released through the ADVs, or the turbine bypass valve if the condenser is available.

The AFW System actuates automatically on low steam generator level by an AFAS as described in LCO 3.3.3, "Engineered Safety Feature (ESF) Instrumentation" and 3.3.4, "ESF Logic." The AFAS initiates signals for starting the AFW pumps and repositioning the valves to initiate AFW flow to the steam generators. The actual pump starts are on an "as required" basis. P-8A is started initially, if the pump fails to start, or if the required flow is not established in a specified period of time, P-8C is started. If P-8A and P-8C do not start, or if required flow is not established in a specified period of time, then P-8B is started.

The AFW System is discussed in the FSAR, Section 9.7 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with a loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat, by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest MSSV set pressure plus 3% with exception of AFW pump P-8C. If AFW pump P-8C is used, operator action maybe required to either trip two of the four PCPs, start an additional AFW pump or reduce steam generator pressure. This will allow the required flowrate to the steam generators that are assumed in the safety analyses.

The limiting Design Basis Accident for the AFW System is a loss of normal feedwater.

In addition, the minimum available AFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident.

The AFW System design is such that it can perform its function following loss of normal feedwater, combined with a loss of offsite power with one AFW pump injecting AFW to one steam generator.

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

LCO

This LCO requires that two AFW trains be OPERABLE to ensure that the AFW System will perform the design safety function to mitigate the consequences of accidents that could result in overpressurization of the primary coolant pressure boundary. Three independent AFW pumps, in two diverse trains, ensure availability of residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two pumps from independent emergency buses. The third AFW pump is powered by a diverse means, a steam driven turbine supplied with steam from a source not isolated by the closure of the MSIVs.

BASES

APPLICABLE SAFETY ANALYSES

The Condensate Storage and Supply provides condensate to remove decay heat and to cool down the plant following all events in the accident analysis, discussed in the FSAR, Chapters 5 and 14. For anticipated operational occurrences and accidents which do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 30 minutes at MODE 3, steaming through the MSSVs followed by a cooldown to Shutdown Cooling (SDC) entry conditions at the design cooldown rate.

The Condensate Storage and Supply satisfies Criterion 3 of 10 CFR 50.36(c)(2).

LCO

To satisfy accident analysis assumptions, the CST and T-81 must contain sufficient cooling water to remove decay heat for 8 hours following a reactor trip from 102% RTP. This amount of time allows for cool down of the PCS to SDC entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this the CST and T-81 must retain sufficient water to ensure adequate net positive suction head for the AFW pumps, and makeup for steaming required to remove decay heat.

The combined CST and T-81 level required is a usable volume of at least 100,000 gallons, which is based on holding the plant in MODE 3 for 4 hours, followed by a cooldown to SDC entry conditions at approximately 75°F per hour. This basis was ~~is~~ established by the Systematic Evaluation Program.

OPERABILITY of the Condensate Storage and Supply System is determined by maintaining the combined tank levels at or above the minimum required volume.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the Condensate Storage and Supply is required to be OPERABLE.

In MODES 5 and 6, the Condensate Storage and Supply is not required because the AFW System is not required.

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System (SWS), and thus to the environment.

Ed

The CCW System consists of three pumps connected in parallel to common suction and discharge headers. The discharge header splits into two parallel heat exchangers and then combines again into a common distribution header ~~to~~ which supplies various heat loads. A common surge tank provides the necessary net positive suction head for the CCW pumps and a surge volume for the system. A train of CCW is considered to ~~shall~~ be that equipment electrically connected to a common safety bus necessary to transfer heat acquired from the various heat loads to the SWS. There are two CCW trains, each associated with a Safeguards Electrical Distribution Train which are described in Specification 3.8.9, "Distribution Systems - Operating." The CCW train associated with the Left Safeguards Electrical Distribution Train consists of two CCW pumps (P-52A, P-52C), both CCW heat exchangers (E-54A, E-54B), the CCW surge tank (T-3), associated piping, valves, and controls for the equipment to perform their safety function. The CCW train associated with the Right Safeguards Electrical Distribution Train consists of one CCW pump (P-52B), both CCW heat exchangers (E-54A, E-54B), the CCW surge tank (T-3), associated piping, valves, and controls for the equipment to perform their safety function. The pumps and valves are automatically started upon receipt of a safety injection actuation signal and all essential valves are aligned to their post accident positions. CCW valve repositioning also occurs following a Recirculation Actuation Signal (RAS) which aligns associated valves to provide full cooling to the CCW heat exchangers during the recirculation phase following a design basis Loss of Coolant Accident (LOCA).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 (continued)

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.8.2

This SR verifies proper automatic operation of the SWS valves on an actual or simulated actuation signal. Specific signals (e.g., safety injection) are tested under Section 3.3, "Instrumentation." If the isolation valve for the noncritical service water header (CV-1359) or for containment air cooler VHX-4 isolation (CV-0869) fail to close, then both trains of SWS are considered inoperable due to the diversion of cooling water flow. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This SR is modified by a Note which states this SR is not required to be met in MODE 4. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

ed X

SR 3.7.8.3

The SR verifies proper automatic operation of the SWS pumps on an actual or simulated actuation signal in the "with standby power available" mode which tests the starting of the pumps by the SIS-X relays. The starting of the pumps by the sequencer is performed in Section 3.8, "Electrical Power Systems." This SR is modified by a Note which states this SR is not required to be met in MODE 4. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for process and operating heat from safety related components during a Design Basis Accident (DBA) or transient, as well as during normal operation. This is done utilizing the Service Water System (SWS).

The UHS has been defined as Lake Michigan. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

The basic performance requirements are that an adequate Net Positive Suction Head (NPSH) to the SWS pumps be available, and that the design basis temperatures of safety related equipment not be exceeded.

Additional information on the design and operation of the system along with a list of components served can be found in FSAR, Section 9.1 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the plant is cooled down and placed on shutdown cooling. Maximum post accident heat load occurs between 20 to 40 minutes after a design basis Loss of Coolant Accident (LOCA). Near this time, the plant switches from injection to recirculation, and the containment cooling systems are required to remove the core decay heat.

Tech Changed

INSERT → ~~The operating limits are~~ based on conservative heat transfer analyses for the worst case LOCA. FSAR, Section 14.18 (Ref. 2) and Design Basis Document (DBD) 1.02 (Ref. 3) provide~~s~~ the details of the analysis which forms the basis for these ~~the~~ operating limits. The assumptions include: worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst case single active failure. X X

The UHS satisfies Criterion 3 of 10 CFR 50.36(c)(2).

INSERT

The minimum water level of the UHS is based on the NPSH requirements for the SWS pumps. The NPSH calculation assumes a minimum water level of 4 feet above the bottom of the pump suction bell which corresponds to an elevation of 557.25 ft. Violation of the SWS pump submergence requirement should never become a factor unless the Lake Michigan water level falls below the top of the sluice gate opening which is at elevation 568.25 ft. Early warning of a falling intake water level is provided by the intake structure level alarm. The nominal lake level is approximately 580 ft mean sea level. The minimum water temperature of the UHS is...

BASES

LCO

The UHS is required to be OPERABLE. The UHS is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate without the loss of NPSH, and without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the UHS temperature should not exceed 81.5°F and the level should not fall below ~~571.0~~ ft above mean sea level during normal plant operation. 568.25

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

A.1 and A.2

If the UHS is inoperable, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies adequate cooling can be maintained. The level specified also ensures sufficient NPSH is available for operating the SWS pumps. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ ~~571.0~~ ft above mean sea level as measured within the boundaries of the intake structure. 568.25

Tech Change

X

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Pool (SFP) Water Level

BASES

BACKGROUND

The minimum water level in the SFP meets the assumptions of iodine decontamination factors following a fuel handling or cask drop accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the SFP design is given in the FSAR, Section 9.11 (Ref. 1), and the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section 9.4 (Ref. 2). The assumptions of fuel handling and fuel cask drop accidents are given in the FSAR, Section 14.19 and 14.11 (Refs. 3 and 4), respectively.

APPLICABLE SAFETY ANALYSES

The minimum water level in the SFP meets the assumptions of fuel handling or fuel cask drop accident analyses described in References 3 and 4 and are consistent with the assumptions of Regulatory Guide 1.25 (Ref. 5). The resultant 2 hour thyroid dose to a person at the exclusion area boundary is well within the 10 CFR 100 (Ref. 6) limits. Ed

assumed

~~According to~~ Reference 5, there is 23 ft of water between the top of the damaged fuel assembly and the fuel pool surface for a fuel handling or fuel cask drop accident. This LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single assembly, dropped and lying horizontally on top of the spent fuel racks, there may be < 23 ft of water above the top of the assembly and the surface, by the width of the assembly. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rods fail from a hypothetical maximum drop. X

The SFP water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2).

BASES

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool until a complete spent fuel pool verification of the stored assemblies has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

<u>A.1, A.2.1, and A.2.2</u>

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to within limit. Alternately, beginning a verification of the SFP fuel locations to ensure proper locations of the fuel can be performed.

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or used (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store 892 irradiated fuel assemblies, which includes storage for failed fuel canisters. The spent fuel storage racks are grouped into two regions, Region I and Region II per Figure 3.7.16-1. The racks are designed as a ~~Class~~ Seismic Category I structure able to withstand seismic events. Region I contains racks in the spent fuel pool having a 10.25 inch center-to-center spacing and a single rack in the north tilt pit having a 11.25 inch by 10.69 inch center-to-center spacing. Region II contains racks in both the spent fuel pool and the north tilt pit having a 9.17 inch center-to-center spacing. Because of the smaller spacing and poison concentration, Region II racks have more limitations for fuel storage than Region I racks. Further information on these limitations can be found in Section 4.0, "Design Features." These limitations (e.g., enrichment, burnup) are sufficient to maintain a k_{eff} of $\leq$ 0.95 for spent fuel of original enrichment of up to 4.40%. ~~However, as higher initial enrichment fuel assemblies are stored in the spent fuel pool, they must be stored in a checkerboard pattern taking into account fuel burnup to maintain a k_{eff} of 0.95 or less.~~

Cd X

Ed

APPLICABLE SAFETY ANALYSES

The spent fuel storage facility is designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans.

The spent fuel assembly storage satisfies Criterion 2 of 10 CFR 50.36(c)(2).

B 3.7 PLANT SYSTEMS

B 3.7.17 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Primary Coolant System (PCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives, and thus is indication of current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak of primary coolant at the limit of 1.0 μCi/gm as assumed in the safety analyses with exception of the control rod ejection analysis which assumes 0.6 gpm. LCO 3.4.13, "PCS Operational LEAKAGE," is more restrictive in that the limit for a primary to secondary tube leak is 0.3 gpm. The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and primary coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

Operating a plant at the allowable limits ~~could~~ would result in a 2 hour Exclusion Area Boundary (EAB) exposure well within the 10 CFR 100 (Ref. 1) limits.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the PCS and steam generators are at low pressure or depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the PCS, and contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Ed
X
X

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR ensures that the secondary specific activity is within the limits of the accident analysis. A gamma isotope analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in primary coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

4.2 ~~EQUIPMENT SAMPLING AND TESTS~~

~~Table 4.2.3~~

~~VENTILATION SYSTEM TESTS~~ A.1

<SEE 3.7.12>

The Control Room Ventilation and Isolation System ~~and the Fuel Storage Area HEPA Charcoal Exhaust System~~ shall be demonstrated to be OPERABLE by the following tests:

<SEE 3.7.12>

3.7.10.2 1. Performing required Control Room Ventilation ~~and Fuel Storage Area filter testing~~ in accordance with the Ventilation Filter Testing Program.

A.8 ~~2. At least once per refueling cycle by:~~ ACTUAL OR SIMULATED ACTUATION SIGNAL A.7

SR 3.7.10.3 a. Verifying that on a ~~containment high pressure and high-radiation test signal~~, the Control Room Ventilation system automatically switches into the emergency mode of operation with flow through the HEPA filter and charcoal adsorber bank.

ADJACENT AREA A.6 0.125 INCHES A.6

SR 3.7.10.4 b. Verifying that ~~the Control Room Ventilation~~ system maintains the Control Room at a positive pressure ≥ ~~1/8 inch~~ WG relative to the ~~outside atmosphere~~ during system emergency mode operation. AT A FLOW RATE ≥ 3040 AND ≤ 3520 cfm M.2

<SEE 3.7.12> ~~c. Verifying that the Fuel Pool Ventilation System is OPERABLE by initiating flow through the HEPA filter and charcoal adsorbers from the control room.~~

<ADD SR 3.7.10.1> M.1 ed

<ADD SR 3.7.10.3 NotE> (M) L.1 Y

ADMINISTRATIVE CHANGES (A)

A.1 All reformatting and renumbering are in accordance with NUREG-1432. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1432. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or implied) to the TS. Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1432. Since the design is already approved by the NRC, adding more details does not result in a technical change.

A.2 CTS 3.4.2 and 3.4.3 require that if a component(s) listed in Specification 3.4.1 is inoperable for more than the time specified, the plant must be placed in HOT SHUTDOWN. In proposed ITS 3.7.7 Required Action B.1, the CTS term is replaced with MODE 3. This is considered to be an administrative change since the effect on operations is similar. This change is consistent with NUREG-1432. Ed X

A.3 CTS 3.4.4 specifies that valves, interlocks and piping that are directly associated with the "above" (CTS 3.4.1) components shall meet the same requirements as listed for that component. CTS 3.4.5 specifies that valves, interlocks and piping which is associated with the containment cooling system and not covered by CTS 3.4.4 may be inoperable for no more than 24 hours if it is required to function during an accident. These requirements are addressed by the definition of OPERABILITY which requires that all associated equipment be OPERABLE. In the proposed ITS, all equipment in a particular train which is required to function during an accident must be OPERABLE and all equipment in the train will have the same Completion Time. This is an administrative change since the requirement remains that all equipment in a train of containment cooling must be OPERABLE. This change is consistent with NUREG-1432.

A.4 CTS 3.3.2 and 3.4.3 require that with the Required Action and associated Completion Time not met the plant must be placed in COLD SHUTDOWN. In proposed ITS 3.7.7 Required Action B.2, the CTS term is replaced with MODE 5. This is considered to be an administrative change since the effect on operations is similar. This change is consistent with NUREG-1432.

A.5 CTS 3.4.3 states "....Continued power operation with one component out of service shall be as specified in Section 3.4.2, with the permissible period in inoperability starting at the time that the first of the two components became inoperable." This explanatory information on the usage rules of technical specifications is addressed in the proposed ITS Section 1.3, "Completion Times," and does not need to be addressed in the Actions of proposed ITS 3.7.8. This is considered to be an administrative change since the requirements on complying with the completion times is addressed in the proposed ITS. This change is consistent with NUREG-1432.

A.6 The Note added to proposed SR 3.7.7.1 to aid the operator in the prevention of entering an inappropriate LCO. The Note reminds the operator that loss of CCW flow to a component may render that component inoperable but does not affect the OPERABILITY of the CCW System. This change is considered administrative that this is a clarifier to the operator to prevent confusion. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE (M)

M.1 CTS 3.3.1, 3.3.2, 3.4.1, and 3.4.2 establish the Applicability for the various components which comprise the CCW by stating that "the reactor shall not be made critical.... unless all of the following conditions are met." The Applicability of the CCW in proposed ITS 3.7.7 is MODES 1, 2, 3, and 4. As such, the requirements associated with CTS 3.3.1, 3.3.2, 3.4.1, and 3.4.2 have been revised to be more restrictive by requiring the CCW to also be OPERABLE during the additional MODES 3 and 4. SRs 3.7.7.2 and 3.7.7.3 are modified by a Note which states that these SRs are not required to be met in MODE 4. This is due to the instrumentation providing the signals are not required in MODE 4. This change keeps consistency with ITS 3.3.3, "ESF Instrumentation," and current licensing basis. This change is an additional restriction on plant operations and is consistent with NUREG-1432.

LESS RESTRICTIVE CHANGES (L) Ed

L.1 CTS 4.2, Table 4.2.3, item ~~1~~ 2.a requires a verification that the Control Room Ventilation system automatically switches into the emergency mode of operation on a "containment high pressure and high radiation test signal." The Applicability of this requirement is "above COLD SHUTDOWN, during REFUELING OPERATIONS, during movement of irradiated fuel assemblies, and during movement of a fuel cask in or over the Spent Fuel Pool." Proposed SR 3.7.10.3 requires a verification that each CRV Filtration train actuates on an actual or simulated actuation signal. The requirement and Applicability of CTS 4.2, Table 4.2.3, item ~~1~~ 2.a is similar to the requirement and Applicability of SR 3.7.10.3. However, SR 3.7.10.3 is further modified by a Note which states that the SR is "not required to be met during movement of irradiate fuel assemblies in the SFP, or during movement of a fuel cask in or over the SFP." The purpose of this Note is to exclude the requirement of the SR during those plant evolutions in which no instrumentation is available to actuate the CRV System. The CRV System is designed to automatically switch to the emergency mode of operation on a "containment high pressure or containment high radiation signal." The instruments used to initiate these actuation signals are not capable of detecting an increase in radiation levels in the fuel handling building, and as such, can not provide automatic actuation of the CRV System in the event of a fuel handling accident or cask drop accident in the SFP. Therefore, the addition of the Note in SR 3.7.10.3 establishes consistency with the design of the CRV System and the requirement of the SR. During movement of irradiate fuel assemblies in the SFP, or during movement of a fuel cask in or over the SFP, manual operator action is necessary to initiate the emergency filtration mode of the CRV System. X Ed

ENCLOSURE 3

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO JANUARY 26, 1999
REQUEST FOR ADDITIONAL INFORMATION

REVISED PAGES FOR SECTION 3.7

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO JANUARY 26, 1999 REQUEST FOR ADDITIONAL INFORMATION
REVISED PAGES FOR SECTION 3.7, PLANT SYSTEMS

Page Change Instructions

Revise the Palisades submittal for conversion to Improved Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by date and contain vertical lines in the margin indicating the areas of change.

REMOVE PAGES	INSERT PAGES	REV DATE	NRC COMMENT#
ATTACHMENT 1 TO ITS CONVERSION SUBMITTAL			
ITS 3.7.9-1	ITS 3.7.9-1	03/15/99	Tech change
ITS 3.7.12-1	ITS 3.7.12-1	03/15/99	RAI 3.7.12-1
ITS 3.7.12-2	ITS 3.7.12-2	03/15/99	RAI 3.7.12-1
ITS 3.7.12-3	------------	03/15/99	RAI 3.7.12-1
ITS 3.7.13-1	ITS 3.7.13-1	03/15/99	RAI 3.7.13-2
ITS 3.7.16-1	ITS 3.7.16-1	03/15/99	RAI 3.7.16-3
ATTACHMENT 2 TO ITS CONVERSION SUBMITTAL			
ITS B 3.7.2-2	ITS B 3.7.2-2	03/15/99	RAI 3.7.2-2
ITS B 3.7.2-3	ITS B 3.7.2-3	03/15/99	RAI 3.7.2-2
ITS B 3.7.2-4	ITS B 3.7.2-4	03/15/99	RAI 3.7.2-2
ITS B 3.7.2-5	ITS B 3.7.2-5	03/15/99	RAI 3.7.2-2
ITS B 3.7.3-1	ITS B 3.7.3-1	03/15/99	RAI 3.7.3-1 RAI 3.7.3-2
ITS B 3.7.3-2	ITS B 3.7.3-2	03/15/99	RAI 3.7.3-2 RAI 3.7.3-5
ITS B 3.7.3-3	ITS B 3.7.3-3	03/15/99	RAI 3.7.3-6
ITS B 3.7.3-4	ITS B 3.7.3-4	03/15/99	RAI 3.7.3-4
ITS B 3.7.3-5	ITS B 3.7.3-5	03/15/99	RAI 3.7.3-4
ITS B 3.7.5-1	ITS B 3.7.5-1	03/15/99	editorial
ITS B 3.7.5-2	ITS B 3.7.5-2	03/15/99	editorial
ITS B 3.7.5-3	ITS B 3.7.5-3	03/15/99	editorial
ITS B 3.7.6-2	ITS B 3.7.6-2	03/15/99	editorial
ITS B 3.7.6-3	ITS B 3.7.6-3	03/15/99	RAI 3.7.6-1
ITS B 3.7.7-1	ITS B 3.7.7-1	03/15/99	editorial
ITS B 3.7.8-5	ITS B 3.7.8-5	03/15/99	editorial
ITS B 3.7.9-1	ITS B 3.7.9-1	03/15/99	Tech change
ITS B 3.7.9-2	ITS B 3.7.9-2	03/15/99	Tech change
ITS B 3.7.9-3	ITS B 3.7.9-3	03/15/99	Tech change
ITS B 3.7.12-1	ITS B 3.7.12-1	03/15/99	RAI 3.7.12-1
ITS B 3.7.12-2	ITS B 3.7.12-2	03/15/99	RAI 3.7.12-1
ITS B 3.7.12-3	ITS B 3.7.12-3	03/15/99	RAI 3.7.12-1
ITS B 3.7.12-4	ITS B 3.7.12-4	03/15/99	RAI 3.7.12-1
ITS B 3.7.12-5	ITS B 3.7.12-5	03/15/99	RAI 3.7.12-1
ITS B 3.7.12-6	ITS B 3.7.12-6	03/15/99	RAI 3.7.12-1
--------------	ITS B 3.7.12-7	03/15/99	RAI 3.7.12-1

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REVISED PAGES FOR SECTION 3.7, PLANT SYSTEMS

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ATTACHMENT 2 TO ITS CONVERSION SUBMITTAL (continued)			
ITS B 3.7.13-2	ITS B 3.7.13-2	03/15/99	RAI 3.7.13-2
ITS B 3.7.14-1	ITS B 3.7.14-1	03/15/99	editorial
ITS B 3.7.15-2	ITS B 3.7.15-2	03/15/99	editorial
ITS B 3.7.16-1	ITS B 3.7.16-1	03/15/99	editorial
ITS B 3.7.16-2	ITS B 3.7.16-2	03/15/99	RAI 3.7.16-1 RAI 3.7.16-3
ITS B 3.7.16-3	--------------	03/15/99	RAI 3.7.16-3
ITS B 3.7.17-1	ITS B 3.7.17-1	03/15/99	editorial
ITS B 3.7.17-3	ITS B 3.7.17-3	03/15/99	editorial
ATTACHMENT 3 TO ITS CONVERSION SUBMITTAL			
CTS 3.7.5, pg 3-38a	CTS 3.7.5, pg 3-38a	03/15/99	RAI 3.7.5-1
CTS 3.7.7, pg 3-29a	CTS 3.7.7, pg 3-29a	03/15/99	RAI 3.7.7-1
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CTS 3.7.12, pg 3-46	CTS 3.7.12, pg 3-46	03/15/99	RAI 3.7.12-1
CTS 3.7.12, pg 4-14	CTS 3.7.12, pg 4-14	03/15/99	RAI 3.7.12-1
DOC 3.7.5, pg 2 of 7	DOC 3.7.5, pg 2 of 7	03/15/99	RAI 3.7.5-2
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DOC 3.7.12, pg 1 of 4	DOC 3.7.12, pg 1 of 3	03/15/99	RAI 3.7.12-1
DOC 3.7.12, pg 2 of 4	DOC 3.7.12, pg 2 of 3	03/15/99	RAI 3.7.12-1
DOC 3.7.12, pg 3 of 4	DOC 3.7.12, pg 3 of 3	03/15/99	RAI 3.7.12-1
DOC 3.7.12, pg 4 of 4	---------------------	03/15/99	RAI 3.7.12-1
DOC 3.7.17, pg 1 of 2	DOC 3.7.17, pg 1 of 3	03/15/99	RAI 3.7.17-1
DOC 3.7.17, pg 2 of 2	DOC 3.7.17, pg 2 of 3	03/15/99	RAI 3.7.17-1
---------------------	DOC 3.7.17, pg 3 of 3	03/15/99	RAI 3.7.17-1
ATTACHMENT 4 TO ITS CONVERSION SUBMITTAL			
NSHC 3.7.12, pg 1 of 2	NSHC 3.7.12, pg 1 of 2	03/15/99	RAI 3.7.12-1
NSHC 3.7.12, pg 2 of 2	NSHC 3.7.12, pg 2 of 2	03/15/99	RAI 3.7.12-1
NSHC 3.7.17, pg 1 of 2	NSHC 3.7.17, pg 1 of 2	03/15/99	RAI 3.7.17-1
NSHC 3.7.17, pg 2 of 2	NSHC 3.7.17, pg 2 of 2	03/15/99	RAI 3.7.17-1
ATTACHMENT 5 TO ITS CONVERSION SUBMITTAL			
NUREG 3.7-21	NUREG 3.7-21	03/15/99	Tech change
NUREG 3.7-29	NUREG 3.7-29	03/15/99	RAI 3.7.13-2
NUREG 3.7-31	NUREG 3.7-31	03/15/99	RAI 3.7.12-1
NUREG 3.7-31 insert	------------	03/15/99	RAI 3.7.12-1
NUREG 3.7-31 inserts	NUREG 3.7-31 inserts	03/15/99	RAI 3.7.12-1
NUREG 3.7-32	NUREG 3.7-32	03/15/99	RAI 3.7.12-1
NUREG 3.7-33	NUREG 3.7-33	03/15/99	RAI 3.7.12-1
NUREG 3.7-39	NUREG 3.7-39	03/15/99	RAI 3.7.16-1 RAI 3.7.16-3

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ATTACHMENT 5 TO ITS CONVERSION SUBMITTAL			
NUREG B 3.7-7	NUREG B 3.7-7	03/15/99	RAI 3.7.2-2
NUREG B 3.7-7 insert	NUREG B 3.7-7 insert	03/15/99	RAI 3.7.2-2
NUREG B 3.7-8	NUREG B 3.7-8	03/15/99	editorial
-------------	NUREG B 3.7-8 insert	03/15/99	editorial
NUREG B 3.7-13 insert	NUREG B 3.7-13 insert	03/15/99	RAI 3.7.3-1 RAI 3.7.3-2
NUREG B 3.7-14	NUREG B 3.7-14	03/15/99	RAI 3.7.3-2 RAI 3.7.3-5
NUREG B 3.7-14 insert	NUREG B 3.7-14 insert	03/15/99	RAI 3.7.3-2
NUREG B 3.7-15	NUREG B 3.7-15	03/15/99	RAI 3.7.3-6
NUREG B 3.7-17	NUREG B 3.7-17	03/15/99	RAI 3.7.3-4
NUREG B 3.7-34	NUREG B 3.7-34	03/15/99	RAI 3.7.6-1
NUREG B 3.7-36 insert	NUREG B 3.7-36 insert	03/15/99	RAI 3.7.6-1
NUREG B 3.7-44 insert	NUREG B 3.7-44 insert	03/15/99	RAI 3.7.6-1
NUREG B 3.7-47	NUREG B 3.7-47	03/15/99	Tech change
--------------	NUREG B 3.7-47 insert	03/15/99	Tech change
NUREG B 3.7-49	NUREG B 3.7-49	03/15/99	Tech change
--------------	NUREG B 3.7-65 insert	03/15/99	RAI 3.7.13-1
NUREG B 3.7-67	NUREG B 3.7-67	03/15/99	RAI 3.7.13-2
NUREG B 3.7-67 insert	NUREG B 3.7-67 insert	03/15/99	RAI 3.7.13-2
NUREG B 3.7-71	NUREG B 3.7-71	03/15/99	RAI 3.7.12-1
NUREG B 3.7-71 insert	NUREG B 3.7-71 insert	03/15/99	RAI 3.7.12-1
NUREG B 3.7-72	NUREG B 3.7-72	03/15/99	RAI 3.7.12-1
NUREG B 3.7-72 in. (2pgs)	NUREG B 3.7-72 in. (3pgs)	03/15/99	RAI 3.7.12-1
NUREG B 3.7-73	NUREG B 3.7-73	03/15/99	RAI 3.7.12-1
NUREG B 3.7-73 insert	NUREG B 3.7-73 insert	03/15/99	RAI 3.7.12-1
NUREG B 3.7-74	NUREG B 3.7-74	03/15/99	RAI 3.7.12-1
NUREG B 3.7-74 insert	--------------	03/15/99	RAI 3.7.12-1
NUREG B 3.7-75	NUREG B 3.7-75	03/15/99	RAI 3.7.12-1
NUREG B 3.7-75 insert	NUREG B 3.7-75 insert	03/15/99	RAI 3.7.12-1
NUREG B 3.7-76	NUREG B 3.7-76	03/15/99	RAI 3.7.12-1
NUREG B 3.7-89	NUREG B 3.7-89	03/15/99	editorial
NUREG B 3.7-90	NUREG B 3.7-90	03/15/99	RAI 3.7.16-1 RAI 3.7.16-3
ATTACHMENT 6 TO ITS CONVERSION SUBMITTAL			
JFD 3.7.14, pg 1 of 2	JFD 3.7.14, pg 1 of 4	03/15/99	RAI 3.7.12-1
JFD 3.7.14, pg 2 of 2	JFD 3.7.14, pg 2 of 4	03/15/99	RAI 3.7.12-1
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JFD 3.7.18, pg 1 of 2	JFD 3.7.18, pg 1 of 2	03/15/99	RAI 3.7.16-1
JFD 3.7.18, pg 2 of 2	JFD 3.7.18, pg 2 of 2	03/15/99	RAI 3.7.16-1

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. UHS inoperable.	A.1 Be in MODE 3.	6 hours
	AND	
	A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify water level of UHS is ≥ 568.25 ft above mean sea level.	24 hours
SR 3.7.9.2	Verify water temperature of UHS is ≤ 81.5°F.	24 hours

3.7 PLANT SYSTEMS

3.7.12 Fuel Handling Area Ventilation System

LCO 3.7.12 The Fuel Handling Area Ventilation System shall be OPERABLE with one fuel handling area exhaust fan aligned to the emergency filter bank and in operation.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel handling building when irradiated fuel assemblies with < 30 days decay time are in the fuel handling building,

During movement of a fuel cask in or over the SFP when irradiated fuel assemblies with < 90 days decay time are in the fuel handling building,

During CORE ALTERATIONS when irradiated fuel assemblies with < 30 days decay time are in the containment with the equipment hatch open,

During movement of irradiated fuel assemblies in the containment when irradiated fuel assemblies with < 30 days decay time are in the containment with the equipment hatch open.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel Handling Area Ventilation System not aligned or in operation. __OR__ Fuel Handling Area Ventilation System inoperable.	A.1 Suspend movement of fuel assemblies. __AND__ A.2 Suspend CORE ALTERATIONS. __AND__ A.3 Suspend movement of a fuel cask in or over the SFP.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Perform required Fuel Handling Area Ventilation System filter testing in accordance with the Ventilation Filter Testing Program.	In accordance with the Ventilation Filter Testing Program
SR 3.7.12.2	Verify the flow rate of the Fuel Handling Area Ventilation System, when aligned to the emergency filter bank, is $\geq$ 5840 cfm and $\leq$ 8760 cfm.	18 months

3.7 PLANT SYSTEMS

3.7.13 Engineered Safeguards Room Ventilation (ESRV) Dampers

LCO 3.7.13 Two ESRV Damper trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ESRV Damper trains inoperable.	A.1 Initiate action to isolate associated ESRV Damper train(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.13.1 Verify each ESRV Damper train closes on an actual or simulated actuation signal.	31 days

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Assembly Storage

LCO 3.7.16 The combination of initial enrichment and burnup of each spent fuel assembly stored in Region II shall be within the requirements of Table 3.7.16-1.

APPLICABILITY: Whenever any fuel assembly is stored in Region II of either the spent fuel pool or the north tilt pit.

ACTIONS

------------------------------------NOTE------------------------------------

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to move the noncomplying fuel assembly from Region II.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Table 3.7.16-1.	Prior to storing the fuel assembly in Region II

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the Main Steam Line Break (MSLB) inside containment, as discussed in the FSAR, Section 14.18 (Ref. 2). It is also influenced by the accident analysis of the MSLB events presented in the FSAR, Section 14.14 (Ref. 3). The MSIVs are swing disc check valves. The inherent characteristic of this type of valve allows for reverse flow through the valve on a differential pressure even if the valve is closed. In the event of an MSLB, if the MSIV associated with the unaffected steam generator fails to close, both steam generators may blowdown. This failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed. The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a double steam generator blowdown event, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modifications would not provide a cost beneficial improvement to plant safety.

There are three different limiting MSLB cases that have been evaluated, one for fuel integrity and two for containment analysis (one for containment temperature and one for containment pressure). The limiting case for containment temperature is the hot full power MSLB inside containment following a turbine trip. At hot full power, the stored energy in the primary coolant is maximized.

The limiting case for the containment analysis for containment pressure and fuel integrity is the hot zero power MSLB inside containment. At zero power, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Reverse flow due to the open MSIV bypass valves, contributes to the total release of the additional mass and energy. With the most reactive control rod assumed stuck in the fully withdrawn position, there is an increased possibility that the core will return to power. The core is ultimately shut down by a combination of doppler feedback, steam generator dryout, and borated water injection delivered by the Emergency Core Cooling System.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The accident analysis compares several different MSLB events against different acceptance criteria. The MSLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The MSLB inside containment at hot full power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following a turbine trip.

With offsite power available, the primary coolant pumps continue to circulate coolant through the steam generators, maximizing the Primary Coolant System (PCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Safety Injection (HPSI) pumps, is delayed.

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. An MSLB inside containment. For this accident scenario, steam is discharged into containment from both steam generators until closure of the MSIV in the intact steam generator occurs. After MSIV closure, steam is discharged into containment only from the affected steam generator.

b. A break outside of containment and upstream from the MSIVs. This scenario is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled PCS cooldown and positive reactivity addition. Closure of the MSIVs limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs. This type of break will be isolated by the closure of the MSIVs. Events such as increased steam flow through the turbine or the turbine bypass valve will also terminate on closure of the MSIVs.

d. A steam generator tube rupture. For this scenario, closure of the MSIVs isolates the affected steam generator from the intact steam generator and minimizes radiological releases.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2).

BASES

LCO

This LCO requires that the MSIV in each of the two steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100.11 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when both MSIVs are closed and deactivated when there is significant mass and energy in the PCS and steam generators. When the MSIVs are closed, they are already performing their safety function. Deactivation can be accomplished by the removal of the motive force (e.g., air) to the valve to prevent valve opening.

In MODE 4, the steam generator energy is low; therefore, the MSIVs are not required to be OPERABLE.

In MODES 5 and 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

<u>A.1</u>

With one MSIV inoperable in MODE 1, time is allowed to restore the component to OPERABLE status. Some repairs can be made to the MSIV with the plant hot. The 8 hour Completion Time is reasonable, considering the probability of an accident occurring during the time period that would require closure of the MSIVs.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment.

BASES

ACTIONS (continued)

B.1

If the MSIV cannot be restored to OPERABLE status within 8 hours, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2 in an orderly manner and without challenging plant systems.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The 8 hour Completion Time is consistent with that allowed in Condition A.

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, MSIV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position.

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status, or closed, within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from MODE 2 in an orderly manner and without challenging plant systems.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves

BASES

BACKGROUND

The MFRVs and MFRV bypass valves in conjunction with feed pump speed, control Main Feedwater (MFW) flow to the steam generators for level control during normal plant operation. The valves also isolate MFW flow to the secondary side of the steam generators following a High Energy Line Break (HELB). Closure of the MFRVs and MFRV bypass valves terminates flow to both steam generators. Closure of the MFRV and MFRV bypass valve effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for Main Steam Line Breaks (MSLBs) inside containment, and reducing the cooldown effects.

The MFRVs and MFRV bypass valves isolate MFW in the event of a secondary side pipe rupture inside containment to limit the quantity of high energy fluid that enters containment through the break. Controlled addition of Auxiliary Feedwater (AFW) is provided by a separate piping system.

One MFRV and one MFRV bypass valve are located on each MFW line outside containment. The piping volume from the valves to the steam generator must be accounted for in calculating mass and energy releases following an MSLB.

The MFRVs and MFRV bypass valves close on receipt of a isolation signal generated by either; steam generator low pressure from its respective steam generator, or containment high pressure. These isolation signals also actuate the Main Steam Isolation Valves (MSIVs) to close. The MFRVs and MFRV bypass valves may also be actuated manually. The MFRVs and MFRV Bypass valves are non-safety grade valves located on non-safety grade piping that fail "as-is" on a loss of air. If required, MFW isolation can be accomplished using manually operated valves upstream or downstream of the MFRVs and MFRV Bypass valves. In addition, each MRFV is equipped with a handwheel that can be used to isolate this MFW flowpath.

A description of the MFRVs and MFRV bypass valves is found in the FSAR, Section 10.2.3 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

Closure of the MFRVs is an assumption in the MSLB containment response analysis. Closure of the MFRVs and MFRV bypass valves is also assumed in the MSLB core response (DNB) analysis.

Failure of an MFRV or MFRV bypass valve to close following a MSLB can result in additional mass and energy to the steam generators contributing to cooldown. This failure also results in additional mass and energy releases following an MSLB event. However, this failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed. The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a single steam generator blowdown event with continued feedwater, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modifications would not provide a cost beneficial improvement to plant safety.

The MFRVs and MFRV bypass valves satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

This LCO ensures that the MFRVs and MFRV bypass valves will isolate MFW flow to the steam generators following an MSLB. This LCO requires that both MFRVs and both MFRV bypass valves be OPERABLE. The MFRVs and MFRV bypass valves are considered OPERABLE when the isolation times are within limits, and are closed on an isolation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an MSLB inside containment.

BASES

APPLICABILITY

All MFRVs and MFRV bypass valves must be OPERABLE, or either closed and deactivated, or isolated by closed manually actuated valves, whenever there is significant mass and energy in the Primary Coolant System and steam generators.

In MODES 1, 2, and 3, the MFRVs or MFRV bypass valves are required to be OPERABLE, except when both MFRVs and both MFRV bypass valves are either closed and deactivated, or isolated by closed manually actuated valves, in order to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are either closed and deactivated, or isolated by closed manually actuated valves, they are already performing their safety function.

Once the valves are closed, deactivation can be accomplished by the removal of the motive force (e.g., electrical power, air) to the valve to prevent valve opening. Closing another manual valve in the flow path either remotely (i.e., control room hand switch) or locally by manual operation satisfies isolation requirements.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFRVs and MFRV bypass valves are not required to be OPERABLE.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one MFRV or MFRV bypass valve inoperable, action must be taken to close or isolate the inoperable valve(s) within 8 hours. When these valve(s) are closed or isolated, they are performing their required safety function (e.g., to isolate the line).

The 8 hour Completion Time is reasonable to close the MFRV or MFRV bypass valve, which includes performing a controlled plant shutdown to condition that supports isolation of the affected valve(s).

BASES

ACTIONS (continued)

B.1 and B.2

If the MFRVs or MFRV bypass valves cannot be restored to OPERABLE status, closed, or isolated in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies the closure time for each MFRV and MFRV bypass valve is ≤ 22.0 seconds on an actual or simulated actuation signal. Specific signals (e.g., steam generator low pressure and containment high pressure) are tested under Section 3.3, "Instrumentation." The MFRV and MFRV bypass valves closure times are bounding values assumed in the MSLB containment response and core response (DNB) analyses (Refs. 3 and 4). This SR is normally performed upon returning the plant to operation following a refueling outage. The MFRVs and MFRV bypass valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the plant generating power. As these valves are not stroke tested at power, they are exempt from the ASME Code, Section XI (Ref. 2) requirements during operation in MODES 1 and 2.

The Frequency is 18 months. The 18 month Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency.

BASES

REFERENCES

1. FSAR, Section 10.2.3
2. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400
3. FSAR, Section 14.18.2
4. FSAR, Section 14.14

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Primary Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction through a common suction line from the Condensate Storage Tank (CST) (LCO 3.7.6, "Condensate Storage and Supply") and pump to the steam generator secondary side via two separate and independent flow paths to a common AFW supply header for each steam generator. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the Main Steam Safety Valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") or Atmospheric Dump Valves (ADVs) (LCO 3.7.4, "Atmospheric Dump Valves (ADVs)"). If the main condenser is available, steam may be released via the turbine bypass valve.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into two trains. One train (A/B) consists of a motor driven pump (P-8A) and the turbine driven pump (P-8B) in parallel, the discharges join together to form a common discharge. The A/B train common discharge separates to form two flow paths, which feed each steam generator via each steam generator's AFW penetration. The second motor driven pump (P-8C) feeds both steam generators through separate flow paths via each steam generator AFW penetration and forms the other train (C). The two trains join together at each AFW penetration to form a common supply to the steam generators. Each AFW pump is capable of providing 100% of the required capacity to the steam generators as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system.

Each motor driven AFW pump is powered from an independent Class 1E power supply, and feeds both steam generators.

BASES

BACKGROUND (continued)

The steam turbine driven AFW pump receives steam from either main steam header upstream of the Main Steam Isolation Valve (MSIV). Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump. The steam supply from steam generator E-50A receives an open signal from the Auxiliary Feedwater Actuation Signal (AFAS) instrumentation. The steam supply from steam generator E-50B does not. This steam source is a manual backup. The turbine driven AFW pump feeds both steam generators through the same flow paths as motor driven AFW pump P-8A.

One pump at full flow is sufficient to remove decay heat and cool the plant to Shutdown Cooling (SDC) System entry conditions.

The AFW System supplies feedwater to the steam generators during normal plant startup, shutdown, and hot standby conditions.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs, with exception of AFW pump P-8C. If AFW pump P-8C is used, operator action may be required to either trip two of four Primary Coolant Pumps (PCPs), start an additional AFW pump, or reduce steam generator pressure. This will allow the required flowrates to the steam generators that are assumed in the safety analyses. Subsequently, the AFW System supplies sufficient water to cool the plant to SDC entry conditions, and steam is released through the ADVs, or the turbine bypass valve if the condenser is available.

The AFW System actuates automatically on low steam generator level by an AFAS as described in LCO 3.3.3, "Engineered Safety Feature (ESF) Instrumentation" and 3.3.4, "ESF Logic." The AFAS initiates signals for starting the AFW pumps and repositioning the valves to initiate AFW flow to the steam generators. The actual pump starts are on an "as required" basis. P-8A is started initially, if the pump fails to start, or if the required flow is not established in a specified period of time, P-8C is started. If P-8A and P-8C do not start, or if required flow is not established in a specified period of time, then P-8B is started.

The AFW System is discussed in the FSAR, Section 9.7 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with a loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat, by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest MSSV set pressure plus 3% with the exception of AFW pump P-8C. If AFW pump P-8C is used, operator action may be required to either trip two of the four PCPs, start an additional AFW pump or reduce steam generator pressure. This will allow the required flowrate to the steam generators that are assumed in the safety analyses.

The limiting Design Basis Accident for the AFW System is a loss of normal feedwater.

In addition, the minimum available AFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident.

The AFW System design is such that it can perform its function following loss of normal feedwater combined with a loss of offsite power with one AFW pump injecting AFW to one steam generator.

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

LCO

This LCO requires that two AFW trains be OPERABLE to ensure that the AFW System will perform the design safety function to mitigate the consequences of accidents that could result in overpressurization of the primary coolant pressure boundary. Three independent AFW pumps, in two diverse trains, ensure availability of residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two pumps from independent emergency buses. The third AFW pump is powered by a diverse means, a steam driven turbine supplied with steam from a source not isolated by the closure of the MSIVs.

BASES

APPLICABLE SAFETY ANALYSES	The Condensate Storage and Supply provides condensate to remove decay heat and to cool down the plant following all events in the accident analysis, discussed in the FSAR, Chapters 5 and 14. For anticipated operational occurrences and accidents which do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 30 minutes at MODE 3, steaming through the MSSVs followed by a cooldown to Shutdown Cooling (SDC) entry conditions at the design cooldown rate. The Condensate Storage and Supply satisfies Criterion 3 of 10 CFR 50.36(c)(2).
LCO	To satisfy accident analysis assumptions, the CST and T-81 must contain sufficient cooling water to remove decay heat for 8 hours following a reactor trip from 102% RTP. This amount of time allows for cool down of the PCS to SDC entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this the CST and T-81 must retain sufficient water to ensure adequate net positive suction head for the AFW pumps, and makeup for steaming required to remove decay heat. The combined CST and T-81 level required is a usable volume of at least 100,000 gallons, which is based on holding the plant in MODE 3 for 4 hours, followed by a cooldown to SDC entry conditions at approximately 75°F per hour. This basis was established by the Systematic Evaluation Program. OPERABILITY of the Condensate Storage and Supply System is determined by maintaining the combined tank levels at or above the minimum required volume.
APPLICABILITY	In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the Condensate Storage and Supply is required to be OPERABLE. In MODES 5 and 6, the Condensate Storage and Supply is not required because the AFW System is not required.

BASES

ACTIONS

A.1 and A.2

If the condensate volume is not within the limit, the OPERABILITY of the backup water supplies must be verified by administrative means within 4 hours and once every 12 hours thereafter.

OPERABILITY of the backup feedwater supplies must include verification of the OPERABILITY of flow paths from the Fire Water System and SWS to the AFW pumps, and availability of the water in the backup supplies. The Condensate Storage and Supply volume must be returned to OPERABLE status within 7 days, as the backup supplies may be performing this function in addition to their normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the Fire Water System and SWS. Additionally, verifying the backup water supplies every 12 hours is adequate to ensure the backup water supplies continue to be available. The 7 day Completion Time is reasonable, based on OPERABLE backup water supplies being available, and the low probability of an event requiring the use of the water from the CST and T-81 occurring during this period.

B.1 and B.2

If the condensate volume cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generator for heat removal, within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System (SWS), and thus to the environment.

The CCW System consists of three pumps connected in parallel to common suction and discharge headers. The discharge header splits into two parallel heat exchangers and then combines again into a common distribution header which supplies various heat loads. A common surge tank provides the necessary net positive suction head for the CCW pumps and a surge volume for the system. A train of CCW is considered to be that equipment electrically connected to a common safety bus necessary to transfer heat acquired from the various heat loads to the SWS. There are two CCW trains, each associated with a Safeguards Electrical Distribution Train which are described in Specification 3.8.9, "Distribution Systems - Operating." The CCW train associated with the Left Safeguards Electrical Distribution Train consists of two CCW pumps (P-52A, P-52C), both CCW heat exchangers (E-54A, E-54B), the CCW surge tank (T-3), associated piping, valves, and controls for the equipment to perform their safety function. The CCW train associated with the Right Safeguards Electrical Distribution Train consists of one CCW pump (P-52B), both CCW heat exchangers (E-54A, E-54B), the CCW surge tank (T-3), associated piping, valves, and controls for the equipment to perform their safety function. The pumps and valves are automatically started upon receipt of a safety injection actuation signal and all essential valves are aligned to their post accident positions. CCW valve repositioning also occurs following a Recirculation Actuation Signal (RAS) which aligns associated valves to provide full cooling to the CCW heat exchangers during the recirculation phase following a design basis Loss of Coolant Accident (LOCA).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 (continued)

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.8.2

This SR verifies proper automatic operation of the SWS valves on an actual or simulated actuation signal. Specific signals (e.g., safety injection) are tested under Section 3.3, "Instrumentation." If the isolation valve for the noncritical service water header (CV-1359) or for containment air cooler VHX-4 (CV-0869) fail to close, then both trains of SWS are considered inoperable due to the diversion of cooling water flow. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This SR is modified by a Note which states this SR is not required to be met in MODE 4. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.3

The SR verifies proper automatic operation of the SWS pumps on an actual or simulated actuation signal in the "with standby power available" mode which tests the starting of the pumps by the SIS-X relays. The starting of the pumps by the sequencer is performed in Section 3.8, "Electrical Power Systems." This SR is modified by a Note which states this SR is not required to be met in MODE 4. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for process and operating heat from safety related components during a Design Basis Accident (DBA) or transient, as well as during normal operation. This is done utilizing the Service Water System (SWS).

The UHS has been defined as Lake Michigan. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

The basic performance requirements are that an adequate Net Positive Suction Head (NPSH) to the SWS pumps be available, and that the design basis temperatures of safety related equipment not be exceeded.

Additional information on the design and operation of the system along with a list of components served can be found in FSAR, Section 9.1 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the plant is cooled down and placed on shutdown cooling. Maximum post accident heat load occurs between 20 to 40 minutes after a design basis Loss of Coolant Accident (LOCA). Near this time, the plant switches from injection to recirculation, and the containment cooling systems are required to remove the core decay heat.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The minimum water level of the UHS is based on the NPSH requirements for the SWS pumps. The NPSH calculation assumes a minimum water level of 4 feet above the bottom of the pump suction bell which corresponds to an elevation of 557.25 ft. Violation of the SWS pump submergence requirement should never become a factor unless the Lake Michigan water level falls below the top of the sluice gate opening which is at elevation 568.25 ft. Early warning of a falling intake water level is provided by the intake structure level alarm. The nominal lake level is approximately 580 ft mean sea level. The minimum water temperature of the UHS is based on conservative heat transfer analyses for the worst case LOCA. FSAR, Section 14.18 (Ref. 2) and Design Basis Document (DBD) 1.02 (Ref. 3) provide the details of the analysis which forms the basis for these operating limits. The assumptions include: worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst case single active failure.

The UHS satisfies Criterion 3 of 10 CFR 50.36(c)(2).

LCO

The UHS is required to be OPERABLE. The UHS is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate without the loss of NPSH, and without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the UHS temperature should not exceed 81.5°F and the level should not fall below 568.25 ft above mean sea level during normal plant operation.

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

BASES

ACTIONS

A.1 and A.2

If the UHS is inoperable, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies adequate cooling can be maintained. The level specified also ensures sufficient NPSH is available for operating the SWS pumps. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ 568.25 ft above mean sea level as measured within the boundaries of the intake structure.

SR 3.7.9.2

This SR verifies that the SWS is available to provide adequate cooling for the maximum accident or normal design heat loads following a DBA. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the water temperature from the UHS is ≤ 81.5°F.

REFERENCES

1. FSAR, Section 9.1
2. FSAR, Section 14.18
3. Design Basis Document (DBD) 1.02, "Service Water System"

NRC REQUEST:

3.7.13-2 New LCO from CTS Table 3.17.3, Item 4
ITS 3.7.13 Actions Note
DOC M.1 and JFD #2

Comment: (Contractor comment 3.7.13-2) No specific technical justification is provided to explain the rationale for developing this LCO as "Separate Condition entry" rather than as a two train system as the STS is developed. "Separate Condition entry" is normally used in the STS for individual inoperable components rather than trains. Also, "Separate Condition entry" is used where the number of inoperabilities are more than two. Therefore, this does not appear to be an appropriate usage of the "Separate Condition entry." The resolution will also depend upon the configuration and contents of each ESRV train noted above in Comment #3.7.13-1.

Consumers Energy Response:

Consumers Energy agrees with the above comment. The Action Note specifying that separate condition entry is allowed for each train has been deleted.

Affected Submittal Pages:

Att 1, ITS 3.7.13, page 3.7.13-1
Att 2, ITS B 3.7.13, page B 3.7.13-2
ATT 5, NUREG 3.7.13, page 3.7-29
ATT 5, NUREG B 3.7.13, page B 3.7-67
Att 5, NUREG B 3.7.13, page B 3.7-67 insert

3.7 PLANT SYSTEMS

3.7.13 Engineered Safeguards Room Ventilation (ESRV) Dampers

LCO 3.7.13 Two ESRV Damper trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

RAI 3.7.13-2

ACTIONS

~~------------------------------------NOTE------------------------------------~~
~~Separate Condition entry is allowed for each train.~~
~~--~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ESRV Damper trains inoperable.	A.1 Initiate action to isolate associated ESRV Damper train(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.13.1 Verify each ESRV Damper train closes on an actual or simulated actuation signal.	31 days

BASES

LCO

Two ESRV Damper trains are required to be OPERABLE to ensure that each engineered safeguards room isolates upon receipt of its respective high radiation alarm. Total system failure could result in the atmospheric release from the engineered safeguards rooms exceeding the required limits in the event of a Design Basis Accident (DBA).

An ESRV Damper train is considered OPERABLE when its associated radiation monitor, instrumentation, ductwork, valves, and dampers are OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ESR-Damper trains are required to be OPERABLE consistent with the OPERABILITY requirements of the Emergency Core Cooling System (ECCS).

In MODES 5 and 6, the ESRV Damper trains are not required to be OPERABLE, since the ECCS is not required to be OPERABLE.

ACTIONS

RAI 3.7.13-2

~~A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The condition of this Specification may be entered independently for each train. The Completion Times of each inoperable train will be tracked separately for each train, starting from the time the condition is entered.~~

A.1

Condition A addresses the failure of one or both ESRV Damper trains. Operation may continue as long as action is immediately initiated to isolate the affected engineered safeguards room. With the inlet and exhaust dampers closed, or if the inlet and outlet ventilation plenums are adequately sealed, the engineered safeguards room is isolated and the intended safety function is achieved, since the potential pathway for radioactivity to escape to the environment from the engineered safeguards room has been minimized.

The Completion Time for this Required Action is commensurate with the importance of maintaining the engineered safeguards room atmosphere isolated from the outside environment when the ECCS pumps are circulating primary coolant after an accident.

4 ECCS PRCAC = ESRV Damper

3.7 PLANT SYSTEMS

4 3.7.13 Engineered Safeguards Room Ventilation (ESRV) Dampers ~~Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)~~

New

LCO 3.7.13 Two ECCS PREACS trains shall be OPERABLE.

RAI 3.7.13-2

APPLICABILITY: MODES 1, 2, 3, and 4.

~~2~~

ACTIONS ~~---- NOTE ---- Separate condition entry is allowed for each train~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
2 A. One or more ECCS PREACS trains inoperable.	A.1 ~~Restore ECCS PREACS train to OPERABLE status~~ 4 Initiate action to isolate associated ESRV Damper train.	~~7 days~~ Immediately
6 ~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 5.~~	~~6 hours~~ ~~36 hours~~

New

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
6 ~~SR 3.7.13.1 Operate each ECCS PREACS train for [≥ 10 continuous hours with the heater operating or (for systems without heaters) ≥ 15 minutes].~~	~~31 days~~

(continued)

BASES

LCO (continued)

~~ECCS PREACS is considered OPERABLE when the individual components necessary to maintain the ECCS Pump Room filtration are OPERABLE in both trains.~~

(4) An ECCS PREACS train is considered OPERABLE when its associated~~:~~ RADIATION MONITOR, INSTRUMENTATION, DUCTWORK, VALVES, AND DAMPERS ARE OPERABLE

~~a. Fan is OPERABLE;~~

(6)

~~b. HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions; and~~

~~c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.~~

(2) APPLICABILITY

In MODES 1, 2, 3, and 4, the ECCS PREACS ~~is~~ TRAINS ARE required to be OPERABLE consistent with the OPERABILITY requirements of the ~~ECCS~~ EMERGENCY CORE COOLING SYSTEM.

In MODES 5 and 6, the ECCS PREACS ~~is~~ TRAINS ARE not required to be OPERABLE, since the ECCS is not required to be OPERABLE.

ACTIONS (2)

~~<INSERT 1>~~

A.1

<INSERT 2> 1

RAI 3.7.13-2

~~With one ECCS PREACS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ECCS PREACS function.~~

(6)

~~The 7 day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72 hour Completion Time) and this system is not a direct support system for the ECCS. The 7 day Completion Time is reasonable, based on the low probability of a DBA occurring during this time period, and the consideration that the remaining train can provide the required capability.~~

(continued)

30-d

RAI 3.7.13-2

SECTION 3.7

INSERT 1

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Condition of this Specification may be entered independently for each train. The Completion Times of each inoperable train will be tracked separately for each train, starting from the time the Condition is entered.

INSERT ~~2~~ 1

Condition A addresses the failure of one or both ESRV Damper train(s). Operation may continue as long as action is immediately initiated to isolate the affected engineered safeguards room. With the inlet and exhaust dampers closed or if the inlet and outlet ventilation plenums are adequately sealed, the engineered safeguards room is isolated and the intended safety function is achieved, since the potential pathway for radioactivity to escape to the environment from the engineered safeguards room has been minimized.

The Completion Time for this Required Action is commensurate with the importance of maintaining the engineered safeguards room atmosphere isolated from the outside environment when the ECCS pumps are circulating primary coolant after an accident.

***NRC REQUEST*:**

3.7.14 Fuel Building Air Cleanup System (FBACS)
3.7.14-1 ITS 3.7.14

Comment: Level is greater than or equal to 674 ft relative to what? (above MSL)?

***Consumers Energy Response*:**

In general, reference to various plant elevations throughout the CTS, ITS, FSAR, and other plant documents is relative to "mean sea level" and, as such, is not explicitly stated. Since the level of the Great Lakes is currently reported using International Great Lake Datum, discussions pertaining to the level of Lake Michigan and to external flooding hazards will specify "mean sea level" as appropriate to clearly indicate the correct reference point (i.e., MSL or IGLD).

***Affected Submittal Pages*:**

No page changes.

NRC REQUEST:

3.7.15 Penetration Room Exhaust Air Cleanup System (PREACS)
No comments

NRC REQUEST:

3.7.16 Fuel Storage Pool Water Level
3.7.16-1 New LCO from CTS 5.4.2.c, d, and i; and Table 5.4-1
ITS 3.7.16 LCO statement, SR 3.7.16.1, and Bases
JFD #4

Comment: (Contractor comment 3.7.16-1) JFD #4 contains no specific technical justification for not retaining the requirements that spent fuel storage is in accordance with Specification 4.3.1.1. The Bases discussion of LCO and SR 3.7.16.1 state these requirements are met which is in contradiction with the ITS LCO proposed. Provide explanation and technical justification that resolves this apparent inconsistency.

Consumers Energy Response:

A new JFD (JFD #7) has been provided to explain why proposed SR 3.7.16.1 does not ensure compliance with Specification 4.3.1.1. As such, reference to Specification 4.3.11 in SR 3.7.16.1 can be deleted. Conforming changes have also been made to the Bases to eliminate inconsistency with the actual surveillance requirement.

Affected Submittal Pages:

Att 2, ITS B 3.7.16, page B 3.7.16-2
Att 5, NUREG 3.7.18, page 3.7-39
Att 5, NUREG B 3.7.18, page B 3.7-90
Att 6, JFD 3.7.18, page 1 of 1

BASES

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Table 3.7.16-1, in the accompanying LCO, ensures that the k_{eff} of the spent fuel pool will always remain < 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool according to Table 3.7.16-1, in the accompanying LCO. Fuel assemblies not meeting the criteria of Table 3.7.16-1 shall be stored in accordance with Specification 4.3.1.1.

RAI 3.7.16-3

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region II of either the spent fuel pool ~~and~~ or the north tilt pit. X

ACTIONS

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

A.1

When the configuration of fuel assemblies stored in Region II the spent fuel pool is not in accordance with Table 3.7.16-1, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Table 3.7.16-1.

SURVEILLANCE REQUIREMENTS

RAI 3.7.16-1

SR 3.7.16.1

Prior to placing the fuel assembly in a Region II storage location,

This SR verifies by administrative ~~means~~ that the initial enrichment and burnup of the fuel assembly ~~is~~ in accordance with Table 3.7.16-1 in the accompanying LCO. ~~For fuel assemblies in the unacceptable range of Table 3.7.16-1, performance of this SR will ensure compliance with Specification 4.3.1.1.~~

32-a

3.7.18 = 3.7.16 (4)

(3)

3.7 PLANT SYSTEMS

3.7.~~18~~ 16 Spent Fuel Assembly Storage

CTS 5.4.2.1 (2) (3) (4)

LCO 3.7.~~18~~ 16 The combination of initial enrichment and burnup of each spent fuel assembly stored in [Region ~~2~~ II] shall be within the ~~acceptable [burnup domain]~~ requirements of ~~Figure~~ Table 3.7.~~18~~ 16-1. ~~[or in accordance with Specification 4.3.1.1]~~

New (1) (2) (4)

APPLICABILITY: Whenever any fuel assembly is stored in [Region ~~2~~ II] of either the spent fuel storage pool or the north tilt pit.

RAI 3.7.16-3

ACTIONS (2)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 --------NOTE-------- LCO 3.0.3 is not applicable. ------------------------ Initiate action to move the noncomplying fuel assembly from ~~[~~Region ~~2]~~ II.	Immediately

New (1) (2)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.~~18~~ 16.1 Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with ~~Figure~~ Table 3.7.~~18~~ 16-1 ~~or Specification 4.3.1.1~~.	Prior to storing the fuel assembly in ~~[~~Region ~~2]~~ II

CTS 5.4.2d (3) (4) (7) (2) (1)

RAI 3.7.16-1

32-b

BASES (continued)

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in either Region 2 II of the spent fuel pool. and the north tilt pit or the north tilt pit

ACTIONS

A.1

RAI 3.7 16-3

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in Region 2 II the spent fuel pool is not in accordance with Figure Table 3.7.18-1, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure Table 3.7.18-1.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.18.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure Table 3.7.18-1 in the accompanying LCO prior to placing the fuel assembly in a Region II storage location. ~~For fuel assemblies in the unacceptable range of Figure 3.7.18-1, performance of this SR will ensure compliance with Specification 4.3.1.1.~~

REFERENCES

None.

RAI 3.7.16-1

< INSERT >

32-C

Change **Discussion**

Note: This attachment provides a brief discussion of the deviations from NUREG-1432 that were made to support the development of the Palisades Nuclear Plant ITS. The Change Numbers correspond to the respective deviation shown on the "NUREG MARKUPS." The first five justifications were used generically throughout the markup of the NUREG. Not all generic justifications are used in each specification.

1. The brackets have been removed and the proper plant specific information or value has been provided.

2. Deviations have been made for clarity, grammatical preference, or to establish consistency within the Improved Technical Specifications. These deviations are editorial in nature and do not involve technical changes or changes of intent.

3. The requirement/statement has been deleted since it is not applicable to this facility. The following requirements have been renumbered, where applicable, to reflect this deletion.

4. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the facility specific nomenclature, number, reference, system description, or analysis description.

5. This change reflects the current licensing basis/technical specification.

6. The storage of failed fuel is accomplished by the use of canisters that fit in the same storage racks as the fuel assemblies themselves. Therefore, the storage pool does not have any specifically designed rack(s) for failed fuel. The reference to a specific number of storage locations for failed fuel is deleted.

RAI 3.7.16-1

X

7. INSERT

INSERT

ISTS 3.7.18 applies to plants which restrict the storage of fuel assemblies in high density storage locations based on meeting an acceptable combination of initial enrichment and discharge burnup. For fuel assemblies which do not meet the initial enrichment and discharge burnup requirements, the assemblies may be stored in compliance with other NRC approved methods or configurations as stipulated in ISTS 4.3.1.1. ISTS SR 3.7.18.1 requires an administrative verification of the initial enrichment and discharge burnup of a fuel assembly prior to storing any assembly in a Region 2 location. For the Palisades Plant, storage of fuel assemblies in high density racks (Region II) is only permitted for fuel assemblies which meet the initial enrichment and discharge burnup requirements. Alternate storage methods or configurations (e.g., checkerboading) in Region II has not been approved by the NRC. Therefore, reference to storage of fuel assemblies in accordance with Specification 4.3.1.1 in the LCO, SR, and SR Bases has been deleted. Assurance that fuel assembly enrichments do not exceed the limits of Region I locations (ITS 4.3.1.1) is controlled administratively in the design of new cores and the procurement of new fuel.

NRC REQUEST:

3.7.16-2 CTS 5.4.2.c and d Bases for ITS 3.7.16
No DOC

Comment: (Contractor comment 3.7.16-5) The movement of these CTS requirements to a location under licensee control must be justified with a DOC as required by NEI 96-06. Provide the necessary technical justification in a "LA" DOC and revise the CTS markup as required.

Consumers Energy Response:

CTS page 5-4a has been provided only to show that a new specification (ITS 3.7.16) has been added. As denoted on this page, the requirements of CTS 5.4.2c and CTS 5.4.2d are addressed in proposed Specification 4.3. The addition of Specification 3.7.16 is justified in DOC M.1.

Affected Submittal Pages:

No page changes.

NRC REQUEST:

3.7.16-3 ITS 3.7.16 Applicability

Comment: The Applicability would be much clearer if it was written as Region II, of either the SFP or the north tilt pit. The present version could be read as Region II of the SFP or anywhere in the north tilt pit.

Consumers Energy Response:

Consumers Energy agrees with the above comment. The Applicability has been revised as suggested.

Affected Submittal Pages:

Att 1, ITS 3.7.16, page 3.7.16-1
Att 2, ITS B 3.7.16, page B 3.7.16-2
Att 5, NUREG 3.7.18, page 3.7-39
Att 5, NUREG B 3.7.18, page B 3.7-90

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Assembly Storage

RAI 3.7.16-3

LCO 3.7.16 The combination of initial enrichment and burnup of each spent fuel assembly stored in Region II shall be within the requirements of Table 3.7.16-1.

APPLICABILITY: Whenever any fuel assembly is stored in Region II of either the spent fuel pool ~~and~~ or the north tilt pit.

X

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to move the noncomplying fuel assembly from Region II.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.16.1 Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Table 3.7.16-1.	Prior to storing the fuel assembly in Region II

BASES

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Table 3.7.16-1, in the accompanying LCO, ensures that the k_{eff} of the spent fuel pool will always remain < 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool according to Table 3.7.16-1, in the accompanying LCO. Fuel assemblies not meeting the criteria of Table 3.7.16-1 shall be stored in accordance with Specification 4.3.1.1.

RAI 3.7.16-3

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region II of either the spent fuel pool ~~and~~ or the north tilt pit. X

ACTIONS

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

A.1

When the configuration of fuel assemblies stored in Region II the spent fuel pool is not in accordance with Table 3.7.16-1, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Table 3.7.16-1.

SURVEILLANCE REQUIREMENTS

RAI 3.7.16-1

SR 3.7.16.1

Prior to placing the fuel assembly in a Region II storage location,

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Table 3.7.16-1 in the accompanying LCO. ~~For fuel assemblies in the unacceptable range of Table 3.7.16-1, performance of this SR will ensure compliance with Specification 4.3.1.1.~~

3.7.18 = 3.7.16 (4)

(3)

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Assembly Storage

RAI 3.7.16-1

CTS 5.4.2.1 (2) (3) (4) (7)

LCO 3.7.16 The combination of initial enrichment and burnup of each spent fuel assembly stored in [Region II] shall be within the requirements of Table ~~acceptable [burnup domain] of Figure~~ 3.7.16-1. ~~[or in accordance with Specification 4.3.1.1]~~

(1) (2) either

New

APPLICABILITY: Whenever any fuel assembly is stored in [Region II] of the spent fuel storage pool or the north tilt pit.

(4) RAI 3.7.16-3

ACTIONS (2)

New

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 --------NOTE-------- LCO 3.0.3 is not applicable. ---------------------------- Initiate action to move the noncomplying fuel assembly from [Region II].	Immediately

(1) (2)

SURVEILLANCE REQUIREMENTS

CTS 5.4.2d (3) (4) (2) (1) (7)

SURVEILLANCE	FREQUENCY
SR 3.7.16.1 Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Table ~~Figure~~ 3.7.16-1 ~~or Specification 4.3.1.1~~.	Prior to storing the fuel assembly in [Region II]

RAI 3.7.16-1

-34-c

BASES (continued)

APPLICABILITY

This LCO applies either whenever any fuel assembly is stored in [Region 2] of the spent fuel pool] or the ~~and~~ the north tilt pit.

II

ACTIONS

A.1

RAI 3.7.16-3

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in [Region 2] the spent fuel pool is not in accordance with ~~Figure~~ Table [3.7.18-1], immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with ~~Figure~~ Table [3.7.18-1].

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.18.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with ~~Figure~~ Table [3.7.18-1] in the accompanying LCO. ~~For fuel assemblies in the unacceptable range of Figure 3.7.18-1, performance of this SR will ensure compliance with Specification 4.3.1.1.~~

prior to placing the fuel assembly in a Region II storage location.

REFERENCES

None.

RAI 3.7.16-1

<INSERT>

34-d

NRC REQUEST:

3.7.17	Fuel Storage Pool Boron Concentration
3.7.17-1	CTS 4.2, Table 4.2.1, Item #7 ITS SR 3.7.17.1 DOC L.1

Comment: (Contractor comment 3.7.17-3) The removal of this CTS requirement appears acceptable; however, the DOC L.1 explains this CTS change but does not provide a specific technical justification for why this CTS requirement can be deleted. Provide this missing justification in a revision to the DOC.

Consumers Energy Response:

DOC L.1 has been revised to provide additional justification for the deletion of CTS 4.2, Table 4.2.1, Item #7.

Affected Submittal Pages:

Att 3, DOC 3.7.17, page 2 of 2
Att 4, NSHC 3.7.17, page 1 of 2

A.5 CTS 3.1.5c requires that with specific activity of the secondary coolant >0.1 μCi/gram DOSE EQUIVALENT I-131, the plant must be placed in COLD SHUTDOWN. In proposed ITS the term is replaced with MODE 5 (see DOC A.4). In proposed ITS 3.7.17 Applicability, the Specification is applicable in MODES 1, 2, 3, and 4. Placing the plant in COLD SHUTDOWN in CTS and having the Applicability in MODES 1, 2, 3, and 4 in proposed ITS is basically the same. This change is considered to be an administrative change since the effect on operations is similar. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE (M)

M.1 CTS 4.2 Table 4.2.1, item 7a, requires the specific activity of the secondary coolant system to be determined once per 31 days whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit, and once per 6 months whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. Proposed ITS SR 3.7.17.1 will require the specific activity to be determined once per 31 days. The proposed ITS SR will not contain the allowance to extend the SR interval to 6 months whenever the gross activity determination indicates iodine concentration below 10% of the allowable limit. This change does not adversely affect safety because the 31 day interval ensures that the specific activity is checked frequently enough to establish a trend to identify secondary activity problems in a timely manner. Deleting an allowance to extend an SR interval constitutes a more restrictive change. This change is consistent with NUREG-1432.

LESS RESTRICTIVE CHANGES - REMOVAL OF DETAILS TO LICENSEE CONTROLLED DOCUMENTS (LA)

There were no "Removal of Details" associated with this specification.

LESS RESTRICTIVE CHANGES (L)

RAI 3.7.17-1

L.1 ~~CTS 4.2, Table 4.2.7 requires a sample of secondary coolant be analyzed for gross radioactivity 3 times every 7 days with a maximum of 72 hours between samples. This requirement has been deleted. The CTS contains no LCO, limiting value, or Required Actions associated with this requirement in CTS, only that sampling is required. This change is considered Less Restrictive because this sampling requirement is deleted. This change is consistent with NUREG-1432.~~

INSERT

CTS 4.2, Table 4.2.1 requires a sample of secondary coolant to be analyzed for gross radioactivity 3 times every 7 days with a maximum of 72 hours between samples. The CTS contains no LCO, limiting value, or Required Actions for secondary coolant gross radioactivity, only that sampling is required. The intent of this surveillance is to monitor the iodine concentration in the secondary coolant in order to determine the frequency at which an isotopic analysis for Dose Equivalent I-131 concentration in the secondary coolant is performed. The CTS requires an isotopic analysis for Dose equivalent I-131 of the secondary coolant once per 31 days whenever the gross activity indicates iodine concentrations greater than 10% of the allowable limit or, once per 6 months whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. However as discussed in DOC M.1 for this specification, the extended surveillance interval of 6 months for the determination of Dose Equivalent I-131 in the secondary coolant has been proposed for deletion and that future testing be performed every 31 days. Thus, the need to perform sampling of the secondary coolant for gross radioactivity is no longer necessary and has been delete in the ITS. This change is acceptable since gross radioactivity in the secondary coolant is not evaluated for radiological consequences in any of the accidents assumed in the FSAR, and the concentration of the Dose Equivalent I-131 in the secondary coolant will continue to be determined at an appropriate frequency. In addition, radiation monitoring instrumentation, controlled in accordance with the Offsite Dose Calculation Manual (e.g., SG blowdown monitors and condenser off gas monitor), is available to monitor increases in the radioactivity levels in the secondary coolant. This change is consistent with NUREG-1432.

LESS RESTRICTIVE CHANGE L.1

New → RAI 3.7.17-1

CTS 4.2, Table 4.2.7 requires a sample of secondary coolant be analyzed for gross radioactivity 3 times every 7 days with a maximum of 72 hours between samples. This requirement has been deleted. The CTS contains no LCO, limiting value, or Required Actions associated with this requirement in CTS, only that sampling is required. This change is considered Less Restrictive because this sampling requirement is deleted. This change is consistent with NUREG-1432.

1. **Does the change involve a significant increase in the probability or consequence of an accident previously evaluated?**

Analyzed events are assumed to be initiated by the failure of plant structures, systems, or components. The proposed change deletes the sample requirement for gross radioactivity of the secondary coolant. This sample does not have a detrimental impact on the integrity of any plant structure, system, or component. Deletion of this sample requirement will not alter the operation of any plant equipment, or otherwise increase its failure probability. As such, the probability of occurrence for a previously analyzed accident is not significantly increased.

The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event. Gross radioactivity of the secondary coolant is not an initial condition input assumed for any analyzed event. The amount of Dose Equivalent I-131 in the secondary coolant is the assumed parameter. The limit requirement for Dose Equivalent I-131 remains unchanged and the sampling requirement has become more restrictive (see M.1). The deletion of the gross radioactivity sampling requirement does not affect the assumptions of an analyzed event. This change does not affect the performance of any credited equipment since the sample requirement is for an unassumed parameter. As a result, no analysis assumptions are violated. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

INSERT

CTS 4.2, Table 4.2.1 requires a sample of secondary coolant to be analyzed for gross radioactivity 3 times every 7 days with a maximum of 72 hours between samples. The CTS contains no LCO, limiting value, or Required Actions for secondary coolant gross radioactivity, only that sampling is required. The intent of this surveillance is to monitor the iodine concentration in the secondary coolant in order to determine the frequency at which an isotopic analysis for Dose Equivalent I-131 concentration in the secondary coolant is performed. The CTS requires an isotopic analysis for Dose equivalent I-131 of the secondary coolant once per 31 days whenever the gross activity indicates iodine concentrations greater than 10% of the allowable limit or, once per 6 months whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. However as discussed in DOC M.1 for this specification, the extended surveillance interval of 6 months for the determination of Dose Equivalent I-131 in the secondary coolant has been proposed for deletion and that future testing be performed every 31 days. Thus, the need to perform sampling of the secondary coolant for gross radioactivity is no longer necessary and has been delete in the ITS. This change is acceptable since gross radioactivity in the secondary coolant is not evaluated for radiological consequences in any of the accidents assumed in the FSAR, and the concentration of the Dose Equivalent I-131 in the secondary coolant will continue to be determined at an appropriate frequency. In addition, radiation monitoring instrumentation, controlled in accordance with the Offsite Dose Calculation Manual (e.g., SG blowdown monitors and condenser off gas monitor), is available to monitor increases in the radioactivity levels in the secondary coolant. This change is consistent with NUREG-1432.

ENCLOSURE 2

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO JANUARY 26, 1999
REQUEST FOR ADDITIONAL INFORMATION

EDITORIAL CHANGES

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. UHS inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify water level of UHS is ≥ ~~571.0~~ 568.25 ft above mean sea level.	24 hours
SR 3.7.9.2	Verify water temperature of UHS is ≤ 81.5°F.	24 hours

X
TECH CHANGE

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Primary Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction through a common suction line from the Condensate Storage Tank (CST) (LCO 3.7.6, "Condensate Storage and Supply") and pump to the steam generator secondary side via two separate and independent flow paths to a common AFW supply header for each steam generator. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the Main Steam Safety Valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") or Atmospheric Dump Valves (ADVs) (LCO 3.7.4, "Atmospheric Dump Valves (ADVs)"). If the main condenser is available, steam may be released via the turbine bypass valve.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into two trains. One train (A/B) consists of a motor driven pump (P-8A) and the turbine driven pump (P-8B) in parallel, the discharges join together to form a common discharge. The A/B train common discharge separates to form two flow paths, which feed each steam generator via each steam generator's AFW penetration. The second motor driven pump (P-8C) feeds both steam generators through separate flow paths via each steam generator AFW penetration and forms the other train (C). The two trains join together at each AFW penetration to form a common supply to the steam generators. Each AFW pump is capable of providing 100% of the required capacity to the steam generators as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Ed X

Each motor driven AFW pump is powered from an independent Class 1E power supply, and feeds both steam generators.

BASES

BACKGROUND (continued)

The steam turbine driven AFW pump receives steam from either main steam header upstream of the Main Steam Isolation Valve (MSIV). Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump. The steam supply from steam generator E-50A receives an open signal from the Auxiliary Feedwater Actuation Signal (AFAS) instrumentation. The steam supply from steam generator E-50B does not. This steam source is a manual backup. The turbine driven AFW pump feeds both steam generators through the same flow paths as motor driven AFW pump P-8A. ed X

One pump at full flow is sufficient to remove decay heat and cool the plant to Shutdown Cooling (SDC) System entry conditions.

The AFW System supplies feedwater to the steam generators during normal plant startup, shutdown, and hot standby conditions.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs, with exception of AFW pump P-8C. If AFW pump P-8C is used, operator action may be required to either trip two of four Primary Coolant Pumps (PCPs), start an additional AFW pump, or reduce steam generator pressure. This will allow the required flowrates to the steam generators that are assumed in the safety analyses. Subsequently, the AFW System supplies sufficient water to cool the plant to SDC entry conditions, and steam is released through the ADVs, or the turbine bypass valve if the condenser is available. ed X

The AFW System actuates automatically on low steam generator level by an AFAS as described in LCO 3.3.3, "Engineered Safety Feature (ESF) Instrumentation" and 3.3.4, "ESF Logic." The AFAS initiates signals for starting the AFW pumps and repositioning the valves to initiate AFW flow to the steam generators. The actual pump starts are on an "as required" basis. P-8A is started initially, if the pump fails to start, or if the required flow is not established in a specified period of time, P-8C is started. If P-8A and P-8C do not start, or if required flow is not established in a specified period of time, then P-8B is started. ed X

The AFW System is discussed in the FSAR, Section 9.7 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with a loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat, by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest MSSV set pressure plus 3% with the exception of AFW pump P-8C. If AFW pump P-8C is used, operator action maybe required to either trip two of the four PCPs, start an additional AFW pump or reduce steam generator pressure. This will allow the required flowrate to the steam generators that are assumed in the safety analyses.

The limiting Design Basis Accident for the AFW System is a loss of normal feedwater.

In addition, the minimum available AFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident.

The AFW System design is such that it can perform its function following loss of normal feedwater, combined with a loss of offsite power with one AFW pump injecting AFW to one steam generator.

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

LCO

This LCO requires that two AFW trains be OPERABLE to ensure that the AFW System will perform the design safety function to mitigate the consequences of accidents that could result in overpressurization of the primary coolant pressure boundary. Three independent AFW pumps, in two diverse trains, ensure availability of residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two pumps from independent emergency buses. The third AFW pump is powered by a diverse means, a steam driven turbine supplied with steam from a source not isolated by the closure of the MSIVs.

BASES

APPLICABLE SAFETY ANALYSES

The Condensate Storage and Supply provides condensate to remove decay heat and to cool down the plant following all events in the accident analysis, discussed in the FSAR, Chapters 5 and 14. For anticipated operational occurrences and accidents which do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 30 minutes at MODE 3, steaming through the MSSVs followed by a cooldown to Shutdown Cooling (SDC) entry conditions at the design cooldown rate.

The Condensate Storage and Supply satisfies Criterion 3 of 10 CFR 50.36(c)(2).

LCO

To satisfy accident analysis assumptions, the CST and T-81 must contain sufficient cooling water to remove decay heat for 8 hours following a reactor trip from 102% RTP. This amount of time allows for cool down of the PCS to SDC entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this the CST and T-81 must retain sufficient water to ensure adequate net positive suction head for the AFW pumps, and makeup for steaming required to remove decay heat.

The combined CST and T-81 level required is a usable volume of at least 100,000 gallons, which is based on holding the plant in MODE 3 for 4 hours, followed by a cooldown to SDC entry conditions at approximately 75°F per hour. This basis was ~~is~~ established by the Systematic Evaluation Program.

OPERABILITY of the Condensate Storage and Supply System is determined by maintaining the combined tank levels at or above the minimum required volume.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the Condensate Storage and Supply is required to be OPERABLE.

In MODES 5 and 6, the Condensate Storage and Supply is not required because the AFW System is not required.

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System (SWS), and thus to the environment.

Ed

The CCW System consists of three pumps connected in parallel to common suction and discharge headers. The discharge header splits into two parallel heat exchangers and then combines again into a common distribution header ~~to~~ which supplies various heat loads. A common surge tank provides the necessary net positive suction head for the CCW pumps and a surge volume for the system. A train of CCW ~~shall~~ is considered to be that equipment electrically connected to a common safety bus necessary to transfer heat acquired from the various heat loads to the SWS. There are two CCW trains, each associated with a Safeguards Electrical Distribution Train which are described in Specification 3.8.9, "Distribution Systems - Operating." The CCW train associated with the Left Safeguards Electrical Distribution Train consists of two CCW pumps (P-52A, P-52C), both CCW heat exchangers (E-54A, E-54B), the CCW surge tank (T-3), associated piping, valves, and controls for the equipment to perform their safety function. The CCW train associated with the Right Safeguards Electrical Distribution Train consists of one CCW pump (P-52B), both CCW heat exchangers (E-54A, E-54B), the CCW surge tank (T-3), associated piping, valves, and controls for the equipment to perform their safety function. The pumps and valves are automatically started upon receipt of a safety injection actuation signal and all essential valves are aligned to their post accident positions. CCW valve repositioning also occurs following a Recirculation Actuation Signal (RAS) which aligns associated valves to provide full cooling to the CCW heat exchangers during the recirculation phase following a design basis Loss of Coolant Accident (LOCA).

X

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 (continued)

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.8.2

This SR verifies proper automatic operation of the SWS valves on an actual or simulated actuation signal. Specific signals (e.g., safety injection) are tested under Section 3.3, "Instrumentation." If the isolation valve for the noncritical service water header (CV-1359) or for containment air cooler VHX-4 ~~isolation~~ (CV-0869) fail to close, then both trains of SWS are considered inoperable due to the diversion of cooling water flow. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This SR is modified by a Note which states this SR is not required to be met in MODE 4. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

ed X

SR 3.7.8.3

The SR verifies proper automatic operation of the SWS pumps on an actual or simulated actuation signal in the "with standby power available" mode which tests the starting of the pumps by the SIS-X relays. The starting of the pumps by the sequencer is performed in Section 3.8, "Electrical Power Systems." This SR is modified by a Note which states this SR is not required to be met in MODE 4. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for process and operating heat from safety related components during a Design Basis Accident (DBA) or transient, as well as during normal operation. This is done utilizing the Service Water System (SWS).

The UHS has been defined as Lake Michigan. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

The basic performance requirements are that an adequate Net Positive Suction Head (NPSH) to the SWS pumps be available, and that the design basis temperatures of safety related equipment not be exceeded.

Additional information on the design and operation of the system along with a list of components served can be found in FSAR, Section 9.1 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the plant is cooled down and placed on shutdown cooling. Maximum post accident heat load occurs between 20 to 40 minutes after a design basis Loss of Coolant Accident (LOCA). Near this time, the plant switches from injection to recirculation, and the containment cooling systems are required to remove the core decay heat.

INSERT → ~~The operating limits are~~ based on conservative heat transfer analyses for the worst case LOCA. FSAR, Section 14.18 (Ref. 2) and Design Basis Document (DBD) 7 1.02 (Ref. 3) provides the details of the analysis which forms the basis for these ~~the~~ operating limits. The assumptions include: worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst case single active failure.

Tech Changed
X
X

The UHS satisfies Criterion 3 of 10 CFR 50.36(c)(2).

INSERT

The minimum water level of the UHS is based on the NPSH requirements for the SWS pumps. The NPSH calculation assumes a minimum water level of 4 feet above the bottom of the pump suction bell which corresponds to an elevation of 557.25 ft. Violation of the SWS pump submergence requirement should never become a factor unless the Lake Michigan water level falls below the top of the sluice gate opening which is at elevation 568.25 ft. Early warning of a falling intake water level is provided by the intake structure level alarm. The nominal lake level is approximately 580 ft mean sea level. The minimum water temperature of the UHS is...

BASES

LCO

The UHS is required to be OPERABLE. The UHS is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate without the loss of NPSH, and without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the UHS temperature should not exceed 81.5°F and the level should not fall below ~~571.0~~ 568.25 ft above mean sea level during normal plant operation.

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

A.1 and A.2

If the UHS is inoperable, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies adequate cooling can be maintained. The level specified also ensures sufficient NPSH is available for operating the SWS pumps. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ ~~571.0~~ 568.25 ft above mean sea level as measured within the boundaries of the intake structure.

Tech change

X

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Pool (SFP) Water Level

BASES

BACKGROUND

The minimum water level in the SFP meets the assumptions of iodine decontamination factors following a fuel handling or cask drop accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the SFP design is given in the FSAR, Section 9.11 (Ref. 1), and the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section 9.4 (Ref. 2). The assumptions of fuel handling and fuel cask drop accidents are given in the FSAR, Section 14.19 and 14.11 (Refs. 3 and 4), respectively.

APPLICABLE SAFETY ANALYSES

The minimum water level in the SFP meets the assumptions of fuel handling or fuel cask drop accident analyses described in References 3 and 4 and are consistent with the assumptions of Regulatory Guide 1.25 (Ref. 5). The resultant 2 hour thyroid dose to a person at the exclusion area boundary is well within the 10 CFR 100 (Ref. 6) limits. Ed

~~According to~~ Reference 5 assumed there is 23 ft of water between the top of the damaged fuel assembly and the fuel pool surface for a fuel handling or fuel cask drop accident. This LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single assembly, dropped and lying horizontally on top of the spent fuel racks, there may be < 23 ft of water above the top of the assembly and the surface, by the width of the assembly. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rods fail from a hypothetical maximum drop. X

The SFP water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2).

BASES

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool until a complete spent fuel pool verification of the stored assemblies has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

<u>A.1, A.2.1, and A.2.2</u>

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to within limit. Alternately, beginning a verification of the SFP fuel locations to ensure proper locations of the fuel can be performed.

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or used (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store 892 irradiated fuel assemblies, which includes storage for failed fuel canisters. The spent fuel storage racks are grouped into two regions, Region I and Region II per Figure 3.7.16-1. The racks are designed as a ~~Class~~ Seismic Category I structure able to withstand seismic events. Region I contains racks in the spent fuel pool having a 10.25 inch center-to-center spacing and a single rack in the north tilt pit having a 11.25 inch by 10.69 inch center-to-center spacing. Region II contains racks in both the spent fuel pool and the north tilt pit having a 9.17 inch center-to-center spacing. Because of the smaller spacing and poison concentration, Region II racks have more limitations for fuel storage than Region I racks. Further information on these limitations can be found in Section 4.0, "Design Features." These limitations (e.g., enrichment, burnup) are sufficient to maintain a k_{eff} of $\leq$ 0.95 for spent fuel of original enrichment of up to 4.40%. ~~However, as higher initial enrichment fuel assemblies are stored in the spent fuel pool, they must be stored in a checkerboard pattern taking into account fuel burnup to maintain a k_{eff} of 0.95 or less.~~

Cd
X
ed

APPLICABLE SAFETY ANALYSES

The spent fuel storage facility is designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans.

The spent fuel assembly storage satisfies Criterion 2 of 10 CFR 50.36(c)(2).

B 3.7 PLANT SYSTEMS

B 3.7.17 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Primary Coolant System (PCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives, and thus is indication of current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak of primary coolant at the limit of 1.0 μCi/gm as assumed in the safety analyses with exception of the control rod ejection analysis which assumes 0.6 gpm. LCO 3.4.13, "PCS Operational LEAKAGE," is more restrictive in that the limit for a primary to secondary tube leak is 0.3 gpm. The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and primary coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

Operating a plant at the allowable limits ~~c~~would result in a 2 hour Exclusion Area Boundary (EAB) exposure well within the 10 CFR 100 (Ref. 1) limits.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the PCS and steam generators are at low pressure or depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

Ed

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant is an indication of a problem in the PCS and contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR ensures that the secondary specific activity is within the limits of the accident analysis. A gamma isotope analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in primary coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

4.2 ~~EQUIPMENT SAMPLING AND TESTS~~

~~Table 4.2.3~~

~~VENTILATION SYSTEM TESTS~~

A.1

The Control Room Ventilation and Isolation System ~~and the Fuel Storage Area HEPA Charcoal Exhaust System~~ <SEE 3.7.12> shall be demonstrated to be OPERABLE by the following tests:

3.7.10.2 1. Performing required Control Room Ventilation ~~and Fuel Storage Area filter testing~~ <SEE 3.7.12> in accordance with the Ventilation Filter Testing Program.

A.8 ~~2. At least once per refueling cycle by:~~

SR 3.7.10.3 a. Verifying that on a ~~containment high-pressure and high-radiation test signal~~ ACTUAL OR SIMULATED ACTUATION SIGNAL A.7, the Control Room Ventilation system automatically switches into the emergency mode of operation with flow through the HEPA filter and charcoal adsorber bank.

SR 3.7.10.4 b. Verifying that ~~the Control Room Ventilation~~ ADJACENT AREA A.6 system maintains the Control Room at a positive pressure ≥ ~~1/8 inch~~ 0.125 INCHES A.6 WG relative to the ~~outside atmosphere~~ during system emergency mode operation. AT A FLOW RATE ≥ 3040 AND ≤ 3520 cfm M.2

<SEE 3.7.12> ~~c. Verifying that the Fuel Pool Ventilation System is OPERABLE by initiating flow through the HEPA filter and charcoal adsorbers from the control room.~~

<ADD SR 3.7.10.1> M.1

<ADD SR 3.7.10.3 NOTE> M L.1

ADMINISTRATIVE CHANGES (A)

A.1 All reformatting and renumbering are in accordance with NUREG-1432. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1432. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or implied) to the TS. Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1432. Since the design is already approved by the NRC, adding more details does not result in a technical change.

A.2 CTS 3.4.2 and 3.4.3 require that if a component(s) listed in Specification 3.4.1 is inoperable for more than the time specified, the plant must be placed in HOT SHUTDOWN. In proposed ITS 3.7.~~8~~ 7 Required Action B.1, the CTS term is replaced with MODE 3. This is considered to be an administrative change since the effect on operations is similar. This change is consistent with NUREG-1432. ed X

A.3 CTS 3.4.4 specifies that valves, interlocks and piping that are directly associated with the "above" (CTS 3.4.1) components shall meet the same requirements as listed for that component. CTS 3.4.5 specifies that valves, interlocks and piping which is associated with the containment cooling system and not covered by CTS 3.4.4 may be inoperable for no more than 24 hours if it is required to function during an accident. These requirements are addressed by the definition of OPERABILITY which requires that all associated equipment be OPERABLE. In the proposed ITS, all equipment in a particular train which is required to function during an accident must be OPERABLE and all equipment in the train will have the same Completion Time. This is an administrative change since the requirement remains that all equipment in a train of containment cooling must be OPERABLE. This change is consistent with NUREG-1432.

, 3.4.2,

 Ed

A.4 CTS 3.3.2 and 3.4.3 require that with the Required Action and associated Completion Time not met the plant must be placed in COLD SHUTDOWN. In proposed ITS 3.7.7 Required Action B.2, the CTS term is replaced with MODE 5. This is considered to be an administrative change since the effect on operations is similar. This change is consistent with NUREG-1432. X

A.5 CTS 3.4.3 states "....Continued power operation with one component out of service shall be as specified in Section 3.4.2, with the permissible period in inoperability starting at the time that the first of the two components became inoperable." This explanatory information on the usage rules of technical specifications is addressed in the proposed ITS Section 1.3, "Completion Times," and does not need to be addressed in the Actions of proposed ITS 3.7.~~8~~7. This is considered to be an administrative change since the requirements on complying with the completion times is addressed in the proposed ITS. This change is consistent with NUREG-1432. Ed X

A.6 The Note added to proposed SR 3.7.7.1 to aid the operator in the prevention of entering an inappropriate LCO. The Note reminds the operator that loss of CCW flow to a component may render that component inoperable but does not affect the OPERABILITY of the CCW System. This change is considered administrative that this is a clarifier to the operator to prevent confusion. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE (M)

M.1 CTS 3.3.1, 3.3.2, 3.4.1, and 3.4.2 establish the Applicability for the various components which comprise the CCW by stating that "the reactor shall not be made critical.... unless all of the following conditions are met." The Applicability of the CCW in proposed ITS 3.7.7 is MODES 1, 2, 3, and 4. As such, the requirements associated with CTS 3.3.1, 3.3.2, 3.4.1, and 3.4.2 have been revised to be more restrictive by requiring the CCW to also be OPERABLE during the additional MODES 3 and 4. SRs 3.7.7.2 and 3.7.7.3 are modified by a Note which states that these SRs are not required to be met in MODE 4. This is due to the instrumentation providing the signals are not required in MODE 4. This change keeps consistency with ITS 3.3.3, "ESF Instrumentation," and current licensing basis. This change is an additional restriction on plant operations and is consistent with NUREG-1432.

LESS RESTRICTIVE CHANGES (L) Ed

L.1 CTS 4.2, Table 4.2.3, item ~~1~~ 2.a requires a verification that the Control Room Ventilation system automatically switches into the emergency mode of operation on a "containment high pressure and high radiation test signal." The Applicability of this requirement is "above COLD SHUTDOWN, during REFUELING OPERATIONS, during movement of irradiated fuel assemblies, and during movement of a fuel cask in or over the Spent Fuel Pool." Proposed SR 3.7.10.3 requires a verification that each CRV Filtration train actuates on an actual or simulated actuation signal. The requirement and Applicability of CTS 4.2, Table 4.2.3, item ~~1~~ 2.a is similar to the requirement and Applicability of SR 3.7.10.3. However, SR 3.7.10.3 is further modified by a Note which states that the SR is "not required to be met during movement of irradiate fuel assemblies in the SFP, or during movement of a fuel cask in or over the SFP." The purpose of this Note is to exclude the requirement of the SR during those plant evolutions in which no instrumentation is available to actuate the CRV System. The CRV System is designed to automatically switch to the emergency mode of operation on a "containment high pressure or containment high radiation signal." The instruments used to initiate these actuation signals are not capable of detecting an increase in radiation levels in the fuel handling building, and as such, can not provide automatic actuation of the CRV System in the event of a fuel handling accident or cask drop accident in the SFP. Therefore, the addition of the Note in SR 3.7.10.3 establishes consistency with the design of the CRV System and the requirement of the SR. During movement of irradiate fuel assemblies in the SFP, or during movement of a fuel cask in or over the SFP, manual operator action is necessary to initiate the emergency filtration mode of the CRV System. X Ed

ENCLOSURE 3

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO JANUARY 26, 1999
REQUEST FOR ADDITIONAL INFORMATION

REVISED PAGES FOR SECTION 3.7

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO JANUARY 26, 1999 REQUEST FOR ADDITIONAL INFORMATION
REVISED PAGES FOR SECTION 3.7, PLANT SYSTEMS

Page Change Instructions

Revise the Palisades submittal for conversion to Improved Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by date and contain vertical lines in the margin indicating the areas of change.

REMOVE PAGES	INSERT PAGES	REV DATE	NRC COMMENT#
ATTACHMENT 1 TO ITS CONVERSION SUBMITTAL			
ITS 3.7.9-1	ITS 3.7.9-1	03/15/99	Tech change
ITS 3.7.12-1	ITS 3.7.12-1	03/15/99	RAI 3.7.12-1
ITS 3.7.12-2	ITS 3.7.12-2	03/15/99	RAI 3.7.12-1
ITS 3.7.12-3	------------	03/15/99	RAI 3.7.12-1
ITS 3.7.13-1	ITS 3.7.13-1	03/15/99	RAI 3.7.13-2
ITS 3.7.16-1	ITS 3.7.16-1	03/15/99	RAI 3.7.16-3
ATTACHMENT 2 TO ITS CONVERSION SUBMITTAL			
ITS B 3.7.2-2	ITS B 3.7.2-2	03/15/99	RAI 3.7.2-2
ITS B 3.7.2-3	ITS B 3.7.2-3	03/15/99	RAI 3.7.2-2
ITS B 3.7.2-4	ITS B 3.7.2-4	03/15/99	RAI 3.7.2-2
ITS B 3.7.2-5	ITS B 3.7.2-5	03/15/99	RAI 3.7.2-2
ITS B 3.7.3-1	ITS B 3.7.3-1	03/15/99	RAI 3.7.3-1 RAI 3.7.3-2
ITS B 3.7.3-2	ITS B 3.7.3-2	03/15/99	RAI 3.7.3-2 RAI 3.7.3-5
ITS B 3.7.3-3	ITS B 3.7.3-3	03/15/99	RAI 3.7.3-6
ITS B 3.7.3-4	ITS B 3.7.3-4	03/15/99	RAI 3.7.3-4
ITS B 3.7.3-5	ITS B 3.7.3-5	03/15/99	RAI 3.7.3-4
ITS B 3.7.5-1	ITS B 3.7.5-1	03/15/99	editorial
ITS B 3.7.5-2	ITS B 3.7.5-2	03/15/99	editorial
ITS B 3.7.5-3	ITS B 3.7.5-3	03/15/99	editorial
ITS B 3.7.6-2	ITS B 3.7.6-2	03/15/99	editorial
ITS B 3.7.6-3	ITS B 3.7.6-3	03/15/99	RAI 3.7.6-1
ITS B 3.7.7-1	ITS B 3.7.7-1	03/15/99	editorial
ITS B 3.7.8-5	ITS B 3.7.8-5	03/15/99	editorial
ITS B 3.7.9-1	ITS B 3.7.9-1	03/15/99	Tech change
ITS B 3.7.9-2	ITS B 3.7.9-2	03/15/99	Tech change
ITS B 3.7.9-3	ITS B 3.7.9-3	03/15/99	Tech change
ITS B 3.7.12-1	ITS B 3.7.12-1	03/15/99	RAI 3.7.12-1
ITS B 3.7.12-2	ITS B 3.7.12-2	03/15/99	RAI 3.7.12-1
ITS B 3.7.12-3	ITS B 3.7.12-3	03/15/99	RAI 3.7.12-1
ITS B 3.7.12-4	ITS B 3.7.12-4	03/15/99	RAI 3.7.12-1
ITS B 3.7.12-5	ITS B 3.7.12-5	03/15/99	RAI 3.7.12-1
ITS B 3.7.12-6	ITS B 3.7.12-6	03/15/99	RAI 3.7.12-1
--------------	ITS B 3.7.12-7	03/15/99	RAI 3.7.12-1

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO JANUARY 26, 1999 REQUEST FOR ADDITIONAL INFORMATION
REVISED PAGES FOR SECTION 3.7, PLANT SYSTEMS

REMOVE PAGES	INSERT PAGES	REV DATE	NRC COMMENT#
ATTACHMENT 2 TO ITS CONVERSION SUBMITTAL (continued)			
ITS B 3.7.13-2	ITS B 3.7.13-2	03/15/99	RAI 3.7.13-2
ITS B 3.7.14-1	ITS B 3.7.14-1	03/15/99	editorial
ITS B 3.7.15-2	ITS B 3.7.15-2	03/15/99	editorial
ITS B 3.7.16-1	ITS B 3.7.16-1	03/15/99	editorial
ITS B 3.7.16-2	ITS B 3.7.16-2	03/15/99	RAI 3.7.16-1 RAI 3.7.16-3
ITS B 3.7.16-3	--------------	03/15/99	RAI 3.7.16-3
ITS B 3.7.17-1	ITS B 3.7.17-1	03/15/99	editorial
ITS B 3.7.17-3	ITS B 3.7.17-3	03/15/99	editorial
ATTACHMENT 3 TO ITS CONVERSION SUBMITTAL			
CTS 3.7.5, pg 3-38a	CTS 3.7.5, pg 3-38a	03/15/99	RAI 3.7.5-1
CTS 3.7.7, pg 3-29a	CTS 3.7.7, pg 3-29a	03/15/99	RAI 3.7.7-1
CTS 3.7.10, pg 4-14	CTS 3.7.10, pg 4-14	03/15/99	editorial
CTS 3.7.12, pg 3-47	CTS 3.7.12, pg 3-47	03/15/99	RAI 3.7.12-1
CTS 3.7.12, pg 3-46	CTS 3.7.12, pg 3-46	03/15/99	RAI 3.7.12-1
CTS 3.7.12, pg 4-14	CTS 3.7.12, pg 4-14	03/15/99	RAI 3.7.12-1
DOC 3.7.5, pg 2 of 7	DOC 3.7.5, pg 2 of 7	03/15/99	RAI 3.7.5-2
DOC 3.7.7, pg 1 of 5	DOC 3.7.7, pg 1 of 5	03/15/99	editorial
DOC 3.7.7, pg 2 of 5	DOC 3.7.7, pg 2 of 5	03/15/99	editorial
DOC 3.7.10, pg 4 of 4	DOC 3.7.10, pg 4 of 4	03/15/99	editorial
DOC 3.7.12, pg 1 of 4	DOC 3.7.12, pg 1 of 3	03/15/99	RAI 3.7.12-1
DOC 3.7.12, pg 2 of 4	DOC 3.7.12, pg 2 of 3	03/15/99	RAI 3.7.12-1
DOC 3.7.12, pg 3 of 4	DOC 3.7.12, pg 3 of 3	03/15/99	RAI 3.7.12-1
DOC 3.7.12, pg 4 of 4	---------------------	03/15/99	RAI 3.7.12-1
DOC 3.7.17, pg 1 of 2	DOC 3.7.17, pg 1 of 3	03/15/99	RAI 3.7.17-1
DOC 3.7.17, pg 2 of 2	DOC 3.7.17, pg 2 of 3	03/15/99	RAI 3.7.17-1
---------------------	DOC 3.7.17, pg 3 of 3	03/15/99	RAI 3.7.17-1
ATTACHMENT 4 TO ITS CONVERSION SUBMITTAL			
NSHC 3.7.12, pg 1 of 2	NSHC 3.7.12, pg 1 of 2	03/15/99	RAI 3.7.12-1
NSHC 3.7.12, pg 2 of 2	NSHC 3.7.12, pg 2 of 2	03/15/99	RAI 3.7.12-1
NSHC 3.7.17, pg 1 of 2	NSHC 3.7.17, pg 1 of 2	03/15/99	RAI 3.7.17-1
NSHC 3.7.17, pg 2 of 2	NSHC 3.7.17, pg 2 of 2	03/15/99	RAI 3.7.17-1
ATTACHMENT 5 TO ITS CONVERSION SUBMITTAL			
NUREG 3.7-21	NUREG 3.7-21	03/15/99	Tech change
NUREG 3.7-29	NUREG 3.7-29	03/15/99	RAI 3.7.13-2
NUREG 3.7-31	NUREG 3.7-31	03/15/99	RAI 3.7.12-1
NUREG 3.7-31 insert	------------	03/15/99	RAI 3.7.12-1
NUREG 3.7-31 inserts	NUREG 3.7-31 inserts	03/15/99	RAI 3.7.12-1
NUREG 3.7-32	NUREG 3.7-32	03/15/99	RAI 3.7.12-1
NUREG 3.7-33	NUREG 3.7-33	03/15/99	RAI 3.7.12-1
NUREG 3.7-39	NUREG 3.7-39	03/15/99	RAI 3.7.16-1 RAI 3.7.16-3

CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS
RESPONSE TO JANUARY 26, 1999 REQUEST FOR ADDITIONAL INFORMATION
REVISED PAGES FOR SECTION 3.7, PLANT SYSTEMS

REMOVE PAGES	INSERT PAGES	REV DATE	NRC COMMENT#
ATTACHMENT 5 TO ITS CONVERSION SUBMITTAL			
NUREG B 3.7-7	NUREG B 3.7-7	03/15/99	RAI 3.7.2-2
NUREG B 3.7-7 insert	NUREG B 3.7-7 insert	03/15/99	RAI 3.7.2-2
NUREG B 3.7-8	NUREG B 3.7-8	03/15/99	editorial
-------------	NUREG B 3.7-8 insert	03/15/99	editorial
NUREG B 3.7-13 insert	NUREG B 3.7-13 insert	03/15/99	RAI 3.7.3-1 RAI 3.7.3-2
NUREG B 3.7-14	NUREG B 3.7-14	03/15/99	RAI 3.7.3-2 RAI 3.7.3-5
NUREG B 3.7-14 insert	NUREG B 3.7-14 insert	03/15/99	RAI 3.7.3-2
NUREG B 3.7-15	NUREG B 3.7-15	03/15/99	RAI 3.7.3-6
NUREG B 3.7-17	NUREG B 3.7-17	03/15/99	RAI 3.7.3-4
NUREG B 3.7-34	NUREG B 3.7-34	03/15/99	RAI 3.7.6-1
NUREG B 3.7-36 insert	NUREG B 3.7-36 insert	03/15/99	RAI 3.7.6-1
NUREG B 3.7-44 insert	NUREG B 3.7-44 insert	03/15/99	RAI 3.7.6-1
NUREG B 3.7-47	NUREG B 3.7-47	03/15/99	Tech change
--------------	NUREG B 3.7-47 insert	03/15/99	Tech change
NUREG B 3.7-49	NUREG B 3.7-49	03/15/99	Tech change
--------------	NUREG B 3.7-65 insert	03/15/99	RAI 3.7.13-1
NUREG B 3.7-67	NUREG B 3.7-67	03/15/99	RAI 3.7.13-2
NUREG B 3.7-67 insert	NUREG B 3.7-67 insert	03/15/99	RAI 3.7.13-2
NUREG B 3.7-71	NUREG B 3.7-71	03/15/99	RAI 3.7.12-1
NUREG B 3.7-71 insert	NUREG B 3.7-71 insert	03/15/99	RAI 3.7.12-1
NUREG B 3.7-72	NUREG B 3.7-72	03/15/99	RAI 3.7.12-1
NUREG B 3.7-72 in. (2pgs)	NUREG B 3.7-72 in. (3pgs)	03/15/99	RAI 3.7.12-1
NUREG B 3.7-73	NUREG B 3.7-73	03/15/99	RAI 3.7.12-1
NUREG B 3.7-73 insert	NUREG B 3.7-73 insert	03/15/99	RAI 3.7.12-1
NUREG B 3.7-74	NUREG B 3.7-74	03/15/99	RAI 3.7.12-1
NUREG B 3.7-74 insert	--------------	03/15/99	RAI 3.7.12-1
NUREG B 3.7-75	NUREG B 3.7-75	03/15/99	RAI 3.7.12-1
NUREG B 3.7-75 insert	NUREG B 3.7-75 insert	03/15/99	RAI 3.7.12-1
NUREG B 3.7-76	NUREG B 3.7-76	03/15/99	RAI 3.7.12-1
NUREG B 3.7-89	NUREG B 3.7-89	03/15/99	editorial
NUREG B 3.7-90	NUREG B 3.7-90	03/15/99	RAI 3.7.16-1 RAI 3.7.16-3
ATTACHMENT 6 TO ITS CONVERSION SUBMITTAL			
JFD 3.7.14, pg 1 of 2	JFD 3.7.14, pg 1 of 4	03/15/99	RAI 3.7.12-1
JFD 3.7.14, pg 2 of 2	JFD 3.7.14, pg 2 of 4	03/15/99	RAI 3.7.12-1
---------------------	JFD 3.7.14, pg 3 of 4	03/15/99	RAI 3.7.12-1
---------------------	JFD 3.7.14, pg 4 of 4	03/15/99	RAI 3.7.12-1
JFD 3.7.18, pg 1 of 2	JFD 3.7.18, pg 1 of 2	03/15/99	RAI 3.7.16-1
JFD 3.7.18, pg 2 of 2	JFD 3.7.18, pg 2 of 2	03/15/99	RAI 3.7.16-1

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. UHS inoperable.	A.1 Be in MODE 3. <u>AND</u> A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify water level of UHS is ≥ 568.25 ft above mean sea level.	24 hours
SR 3.7.9.2	Verify water temperature of UHS is ≤ 81.5°F.	24 hours

3.7 PLANT SYSTEMS

3.7.12 Fuel Handling Area Ventilation System

LCO 3.7.12 The Fuel Handling Area Ventilation System shall be OPERABLE with one fuel handling area exhaust fan aligned to the emergency filter bank and in operation.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel handling building when irradiated fuel assemblies with < 30 days decay time are in the fuel handling building,

During movement of a fuel cask in or over the SFP when irradiated fuel assemblies with < 90 days decay time are in the fuel handling building,

During CORE ALTERATIONS when irradiated fuel assemblies with < 30 days decay time are in the containment with the equipment hatch open,

During movement of irradiated fuel assemblies in the containment when irradiated fuel assemblies with < 30 days decay time are in the containment with the equipment hatch open.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel Handling Area Ventilation System not aligned or in operation. OR Fuel Handling Area Ventilation System inoperable.	A.1 Suspend movement of fuel assemblies. AND A.2 Suspend CORE ALTERATIONS. AND A.3 Suspend movement of a fuel cask in or over the SFP.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Perform required Fuel Handling Area Ventilation System filter testing in accordance with the Ventilation Filter Testing Program.	In accordance with the Ventilation Filter Testing Program
SR 3.7.12.2	Verify the flow rate of the Fuel Handling Area Ventilation System, when aligned to the emergency filter bank, is ≥ 5840 cfm and ≤ 8760 cfm.	18 months

3.7 PLANT SYSTEMS

3.7.13 Engineered Safeguards Room Ventilation (ESRV) Dampers

LCO 3.7.13 Two ESRV Damper trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ESRV Damper trains inoperable.	A.1 Initiate action to isolate associated ESRV Damper train(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.13.1 Verify each ESRV Damper train closes on an actual or simulated actuation signal.	31 days

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Assembly Storage

LCO 3.7.16 The combination of initial enrichment and burnup of each spent fuel assembly stored in Region II shall be within the requirements of Table 3.7.16-1.

APPLICABILITY: Whenever any fuel assembly is stored in Region II of either the spent fuel pool or the north tilt pit.

ACTIONS

-----------------------------------NOTE-----------------------------------

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to move the noncomplying fuel assembly from Region II.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.16.1 Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Table 3.7.16-1.	Prior to storing the fuel assembly in Region II

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the Main Steam Line Break (MSLB) inside containment, as discussed in the FSAR, Section 14.18 (Ref. 2). It is also influenced by the accident analysis of the MSLB events presented in the FSAR, Section 14.14 (Ref. 3). The MSIVs are swing disc check valves. The inherent characteristic of this type of valve allows for reverse flow through the valve on a differential pressure even if the valve is closed. In the event of an MSLB, if the MSIV associated with the unaffected steam generator fails to close, both steam generators may blowdown. This failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed. The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a double steam generator blowdown event, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modifications would not provide a cost beneficial improvement to plant safety.

There are three different limiting MSLB cases that have been evaluated, one for fuel integrity and two for containment analysis (one for containment temperature and one for containment pressure). The limiting case for containment temperature is the hot full power MSLB inside containment following a turbine trip. At hot full power, the stored energy in the primary coolant is maximized.

The limiting case for the containment analysis for containment pressure and fuel integrity is the hot zero power MSLB inside containment. At zero power, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Reverse flow due to the open MSIV bypass valves, contributes to the total release of the additional mass and energy. With the most reactive control rod assumed stuck in the fully withdrawn position, there is an increased possibility that the core will return to power. The core is ultimately shut down by a combination of doppler feedback, steam generator dryout, and borated water injection delivered by the Emergency Core Cooling System.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The accident analysis compares several different MSLB events against different acceptance criteria. The MSLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The MSLB inside containment at hot full power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following a turbine trip.

With offsite power available, the primary coolant pumps continue to circulate coolant through the steam generators, maximizing the Primary Coolant System (PCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Safety Injection (HPSI) pumps, is delayed.

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. An MSLB inside containment. For this accident scenario, steam is discharged into containment from both steam generators until closure of the MSIV in the intact steam generator occurs. After MSIV closure, steam is discharged into containment only from the affected steam generator.

b. A break outside of containment and upstream from the MSIVs. This scenario is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled PCS cooldown and positive reactivity addition. Closure of the MSIVs limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs. This type of break will be isolated by the closure of the MSIVs. Events such as increased steam flow through the turbine or the turbine bypass valve will also terminate on closure of the MSIVs.

d. A steam generator tube rupture. For this scenario, closure of the MSIVs isolates the affected steam generator from the intact steam generator and minimizes radiological releases.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2).

BASES

LCO

This LCO requires that the MSIV in each of the two steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100.11 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when both MSIVs are closed and deactivated when there is significant mass and energy in the PCS and steam generators. When the MSIVs are closed, they are already performing their safety function. Deactivation can be accomplished by the removal of the motive force (e.g., air) to the valve to prevent valve opening.

In MODE 4, the steam generator energy is low; therefore, the MSIVs are not required to be OPERABLE.

In MODES 5 and 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, time is allowed to restore the component to OPERABLE status. Some repairs can be made to the MSIV with the plant hot. The 8 hour Completion Time is reasonable, considering the probability of an accident occurring during the time period that would require closure of the MSIVs.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment.

BASES

ACTIONS (continued)

B.1

If the MSIV cannot be restored to OPERABLE status within 8 hours, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2 in an orderly manner and without challenging plant systems.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The 8 hour Completion Time is consistent with that allowed in Condition A.

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, MSIV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position.

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status, or closed, within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from MODE 2 in an orderly manner and without challenging plant systems.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves

BASES

BACKGROUND

The MFRVs and MFRV bypass valves in conjunction with feed pump speed, control Main Feedwater (MFW) flow to the steam generators for level control during normal plant operation. The valves also isolate MFW flow to the secondary side of the steam generators following a High Energy Line Break (HELB). Closure of the MFRVs and MFRV bypass valves terminates flow to both steam generators. Closure of the MFRV and MFRV bypass valve effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for Main Steam Line Breaks (MSLBs) inside containment, and reducing the cooldown effects.

The MFRVs and MFRV bypass valves isolate MFW in the event of a secondary side pipe rupture inside containment to limit the quantity of high energy fluid that enters containment through the break. Controlled addition of Auxiliary Feedwater (AFW) is provided by a separate piping system.

One MFRV and one MFRV bypass valve are located on each MFW line outside containment. The piping volume from the valves to the steam generator must be accounted for in calculating mass and energy releases following an MSLB.

The MFRVs and MFRV bypass valves close on receipt of a isolation signal generated by either; steam generator low pressure from its respective steam generator, or containment high pressure. These isolation signals also actuate the Main Steam Isolation Valves (MSIVs) to close. The MFRVs and MFRV bypass valves may also be actuated manually. The MFRVs and MFRV Bypass valves are non-safety grade valves located on non-safety grade piping that fail "as-is" on a loss of air. If required, MFW isolation can be accomplished using manually operated valves upstream or downstream of the MFRVs and MFRV Bypass valves. In addition, each MRFV is equipped with a handwheel that can be used to isolate this MFW flowpath.

A description of the MFRVs and MFRV bypass valves is found in the FSAR, Section 10.2.3 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

Closure of the MFRVs is an assumption in the MSLB containment response analysis. Closure of the MFRVs and MFRV bypass valves is also assumed in the MSLB core response (DNB) analysis.

Failure of an MFRV or MFRV bypass valve to close following a MSLB can result in additional mass and energy to the steam generators contributing to cooldown. This failure also results in additional mass and energy releases following an MSLB event. However, this failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed. The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a single steam generator blowdown event with continued feedwater, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modifications would not provide a cost beneficial improvement to plant safety.

The MFRVs and MFRV bypass valves satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

This LCO ensures that the MFRVs and MFRV bypass valves will isolate MFW flow to the steam generators following an MSLB. This LCO requires that both MFRVs and both MFRV bypass valves be OPERABLE. The MFRVs and MFRV bypass valves are considered OPERABLE when the isolation times are within limits, and are closed on an isolation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an MSLB inside containment.

BASES

APPLICABILITY

All MFRVs and MFRV bypass valves must be OPERABLE, or either closed and deactivated, or isolated by closed manually actuated valves, whenever there is significant mass and energy in the Primary Coolant System and steam generators.

In MODES 1, 2, and 3, the MFRVs or MFRV bypass valves are required to be OPERABLE, except when both MFRVs and both MFRV bypass valves are either closed and deactivated, or isolated by closed manually actuated valves, in order to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are either closed and deactivated, or isolated by closed manually actuated valves, they are already performing their safety function.

Once the valves are closed, deactivation can be accomplished by the removal of the motive force (e.g., electrical power, air) to the valve to prevent valve opening. Closing another manual valve in the flow path either remotely (i.e., control room hand switch) or locally by manual operation satisfies isolation requirements.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFRVs and MFRV bypass valves are not required to be OPERABLE.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one MFRV or MFRV bypass valve inoperable, action must be taken to close or isolate the inoperable valve(s) within 8 hours. When these valve(s) are closed or isolated, they are performing their required safety function (e.g., to isolate the line).

The 8 hour Completion Time is reasonable to close the MFRV or MFRV bypass valve, which includes performing a controlled plant shutdown to condition that supports isolation of the affected valve(s).

BASES

ACTIONS (continued)

B.1 and B.2

If the MFRVs or MFRV bypass valves cannot be restored to OPERABLE status, closed, or isolated in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies the closure time for each MFRV and MFRV bypass valve is ≤ 22.0 seconds on an actual or simulated actuation signal. Specific signals (e.g., steam generator low pressure and containment high pressure) are tested under Section 3.3, "Instrumentation." The MFRV and MFRV bypass valves closure times are bounding values assumed in the MSLB containment response and core response (DNB) analyses (Refs. 3 and 4). This SR is normally performed upon returning the plant to operation following a refueling outage. The MFRVs and MFRV bypass valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the plant generating power. As these valves are not stroke tested at power, they are exempt from the ASME Code, Section XI (Ref. 2) requirements during operation in MODES 1 and 2.

The Frequency is 18 months. The 18 month Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency.

BASES

REFERENCES

1. FSAR, Section 10.2.3
2. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400
3. FSAR, Section 14.18.2
4. FSAR, Section 14.14

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Primary Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction through a common suction line from the Condensate Storage Tank (CST) (LCO 3.7.6, "Condensate Storage and Supply") and pump to the steam generator secondary side via two separate and independent flow paths to a common AFW supply header for each steam generator. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the Main Steam Safety Valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") or Atmospheric Dump Valves (ADVs) (LCO 3.7.4, "Atmospheric Dump Valves (ADVs)"). If the main condenser is available, steam may be released via the turbine bypass valve.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into two trains. One train (A/B) consists of a motor driven pump (P-8A) and the turbine driven pump (P-8B) in parallel, the discharges join together to form a common discharge. The A/B train common discharge separates to form two flow paths, which feed each steam generator via each steam generator's AFW penetration. The second motor driven pump (P-8C) feeds both steam generators through separate flow paths via each steam generator AFW penetration and forms the other train (C). The two trains join together at each AFW penetration to form a common supply to the steam generators. Each AFW pump is capable of providing 100% of the required capacity to the steam generators as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system.

Each motor driven AFW pump is powered from an independent Class 1E power supply, and feeds both steam generators.

BASES

BACKGROUND (continued)

The steam turbine driven AFW pump receives steam from either main steam header upstream of the Main Steam Isolation Valve (MSIV). Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump. The steam supply from steam generator E-50A receives an open signal from the Auxiliary Feedwater Actuation Signal (AFAS) instrumentation. The steam supply from steam generator E-50B does not. This steam source is a manual backup. The turbine driven AFW pump feeds both steam generators through the same flow paths as motor driven AFW pump P-8A.

One pump at full flow is sufficient to remove decay heat and cool the plant to Shutdown Cooling (SDC) System entry conditions.

The AFW System supplies feedwater to the steam generators during normal plant startup, shutdown, and hot standby conditions.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs, with exception of AFW pump P-8C. If AFW pump P-8C is used, operator action may be required to either trip two of four Primary Coolant Pumps (PCPs), start an additional AFW pump, or reduce steam generator pressure. This will allow the required flowrates to the steam generators that are assumed in the safety analyses. Subsequently, the AFW System supplies sufficient water to cool the plant to SDC entry conditions, and steam is released through the ADVs, or the turbine bypass valve if the condenser is available.

The AFW System actuates automatically on low steam generator level by an AFAS as described in LCO 3.3.3, "Engineered Safety Feature (ESF) Instrumentation" and 3.3.4, "ESF Logic." The AFAS initiates signals for starting the AFW pumps and repositioning the valves to initiate AFW flow to the steam generators. The actual pump starts are on an "as required" basis. P-8A is started initially, if the pump fails to start, or if the required flow is not established in a specified period of time, P-8C is started. If P-8A and P-8C do not start, or if required flow is not established in a specified period of time, then P-8B is started.

The AFW System is discussed in the FSAR, Section 9.7 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with a loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat, by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest MSSV set pressure plus 3% with the exception of AFW pump P-8C. If AFW pump P-8C is used, operator action may be required to either trip two of the four PCPs, start an additional AFW pump or reduce steam generator pressure. This will allow the required flowrate to the steam generators that are assumed in the safety analyses.

The limiting Design Basis Accident for the AFW System is a loss of normal feedwater.

In addition, the minimum available AFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident.

The AFW System design is such that it can perform its function following loss of normal feedwater combined with a loss of offsite power with one AFW pump injecting AFW to one steam generator.

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

LCO

This LCO requires that two AFW trains be OPERABLE to ensure that the AFW System will perform the design safety function to mitigate the consequences of accidents that could result in overpressurization of the primary coolant pressure boundary. Three independent AFW pumps, in two diverse trains, ensure availability of residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two pumps from independent emergency buses. The third AFW pump is powered by a diverse means, a steam driven turbine supplied with steam from a source not isolated by the closure of the MSIVs.

BASES

APPLICABLE SAFETY ANALYSES

The Condensate Storage and Supply provides condensate to remove decay heat and to cool down the plant following all events in the accident analysis, discussed in the FSAR, Chapters 5 and 14. For anticipated operational occurrences and accidents which do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 30 minutes at MODE 3, steaming through the MSSVs followed by a cooldown to Shutdown Cooling (SDC) entry conditions at the design cooldown rate.

The Condensate Storage and Supply satisfies Criterion 3 of 10 CFR 50.36(c)(2).

LCO

To satisfy accident analysis assumptions, the CST and T-81 must contain sufficient cooling water to remove decay heat for 8 hours following a reactor trip from 102% RTP. This amount of time allows for cool down of the PCS to SDC entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this the CST and T-81 must retain sufficient water to ensure adequate net positive suction head for the AFW pumps, and makeup for steaming required to remove decay heat.

The combined CST and T-81 level required is a usable volume of at least 100,000 gallons, which is based on holding the plant in MODE 3 for 4 hours, followed by a cooldown to SDC entry conditions at approximately 75°F per hour. This basis was established by the Systematic Evaluation Program.

OPERABILITY of the Condensate Storage and Supply System is determined by maintaining the combined tank levels at or above the minimum required volume.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the Condensate Storage and Supply is required to be OPERABLE.

In MODES 5 and 6, the Condensate Storage and Supply is not required because the AFW System is not required.

BASES

ACTIONS

A.1 and A.2

If the condensate volume is not within the limit, the OPERABILITY of the backup water supplies must be verified by administrative means within 4 hours and once every 12 hours thereafter.

OPERABILITY of the backup feedwater supplies must include verification of the OPERABILITY of flow paths from the Fire Water System and SWS to the AFW pumps, and availability of the water in the backup supplies. The Condensate Storage and Supply volume must be returned to OPERABLE status within 7 days, as the backup supplies may be performing this function in addition to their normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the Fire Water System and SWS. Additionally, verifying the backup water supplies every 12 hours is adequate to ensure the backup water supplies continue to be available. The 7 day Completion Time is reasonable, based on OPERABLE backup water supplies being available, and the low probability of an event requiring the use of the water from the CST and T-81 occurring during this period.

B.1 and B.2

If the condensate volume cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generator for heat removal, within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System (SWS), and thus to the environment.

The CCW System consists of three pumps connected in parallel to common suction and discharge headers. The discharge header splits into two parallel heat exchangers and then combines again into a common distribution header which supplies various heat loads. A common surge tank provides the necessary net positive suction head for the CCW pumps and a surge volume for the system. A train of CCW is considered to be that equipment electrically connected to a common safety bus necessary to transfer heat acquired from the various heat loads to the SWS. There are two CCW trains, each associated with a Safeguards Electrical Distribution Train which are described in Specification 3.8.9, "Distribution Systems - Operating." The CCW train associated with the Left Safeguards Electrical Distribution Train consists of two CCW pumps (P-52A, P-52C), both CCW heat exchangers (E-54A, E-54B), the CCW surge tank (T-3), associated piping, valves, and controls for the equipment to perform their safety function. The CCW train associated with the Right Safeguards Electrical Distribution Train consists of one CCW pump (P-52B), both CCW heat exchangers (E-54A, E-54B), the CCW surge tank (T-3), associated piping, valves, and controls for the equipment to perform their safety function. The pumps and valves are automatically started upon receipt of a safety injection actuation signal and all essential valves are aligned to their post accident positions. CCW valve repositioning also occurs following a Recirculation Actuation Signal (RAS) which aligns associated valves to provide full cooling to the CCW heat exchangers during the recirculation phase following a design basis Loss of Coolant Accident (LOCA).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 (continued)

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.8.2

This SR verifies proper automatic operation of the SWS valves on an actual or simulated actuation signal. Specific signals (e.g., safety injection) are tested under Section 3.3, "Instrumentation." If the isolation valve for the noncritical service water header (CV-1359) or for containment air cooler VHX-4 (CV-0869) fail to close, then both trains of SWS are considered inoperable due to the diversion of cooling water flow. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This SR is modified by a Note which states this SR is not required to be met in MODE 4. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.3

The SR verifies proper automatic operation of the SWS pumps on an actual or simulated actuation signal in the "with standby power available" mode which tests the starting of the pumps by the SIS-X relays. The starting of the pumps by the sequencer is performed in Section 3.8, "Electrical Power Systems." This SR is modified by a Note which states this SR is not required to be met in MODE 4. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for process and operating heat from safety related components during a Design Basis Accident (DBA) or transient, as well as during normal operation. This is done utilizing the Service Water System (SWS).

The UHS has been defined as Lake Michigan. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

The basic performance requirements are that an adequate Net Positive Suction Head (NPSH) to the SWS pumps be available, and that the design basis temperatures of safety related equipment not be exceeded.

Additional information on the design and operation of the system along with a list of components served can be found in FSAR, Section 9.1 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the plant is cooled down and placed on shutdown cooling. Maximum post accident heat load occurs between 20 to 40 minutes after a design basis Loss of Coolant Accident (LOCA). Near this time, the plant switches from injection to recirculation, and the containment cooling systems are required to remove the core decay heat.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The minimum water level of the UHS is based on the NPSH requirements for the SWS pumps. The NPSH calculation assumes a minimum water level of 4 feet above the bottom of the pump suction bell which corresponds to an elevation of 557.25 ft. Violation of the SWS pump submergence requirement should never become a factor unless the Lake Michigan water level falls below the top of the sluice gate opening which is at elevation 568.25 ft. Early warning of a falling intake water level is provided by the intake structure level alarm. The nominal lake level is approximately 580 ft mean sea level. The minimum water temperature of the UHS is based on conservative heat transfer analyses for the worst case LOCA. FSAR, Section 14.18 (Ref. 2) and Design Basis Document (DBD) 1.02 (Ref. 3) provide the details of the analysis which forms the basis for these operating limits. The assumptions include: worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst case single active failure. The UHS satisfies Criterion 3 of 10 CFR 50.36(c)(2).
LCO	The UHS is required to be OPERABLE. The UHS is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate without the loss of NPSH, and without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the UHS temperature should not exceed 81.5°F and the level should not fall below 568.25 ft above mean sea level during normal plant operation.
APPLICABILITY	In MODES 1, 2, 3, and 4, the UHS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES. In MODES 5 and 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

BASES

ACTIONS

A.1 and A.2

If the UHS is inoperable, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies adequate cooling can be maintained. The level specified also ensures sufficient NPSH is available for operating the SWS pumps. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ 568.25 ft above mean sea level as measured within the boundaries of the intake structure.

SR 3.7.9.2

This SR verifies that the SWS is available to provide adequate cooling for the maximum accident or normal design heat loads following a DBA. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the water temperature from the UHS is ≤ 81.5°F.

REFERENCES

1. FSAR, Section 9.1
2. FSAR, Section 14.18
3. Design Basis Document (DBD) 1.02, "Service Water System"

B 3.7 PLANT SYSTEMS

B 3.7.12 Fuel Handling Area Ventilation System

BASES

BACKGROUND

The Fuel Handling Area Ventilation System filters airborne radioactive particulates from the area of the spent fuel pool following a fuel handling accident or a fuel cask drop accident. The fuel handling area is served by two separate subsystems one being part of the original plant design, and the other being added as part of the Auxiliary Building Addition.

The original plant design consists of a supply plenum and an exhaust plenum including associated ductwork, dampers, and instrumentation. The supply plenum contains one prefilter, two heating coils, and one supply fan. The exhaust plenum contains two filter banks (normal and emergency) configured in a parallel flow arrangement, and two independent exhaust fans which draw air from a common duct. The "normal filter bank" contains a prefilter and a High Efficiency Particulate Air (HEPA) filter. The "emergency filter bank" contains a prefilter, HEPA filter, and an activated charcoal filter.

The Auxiliary Building Addition, which was added to serve the spaces at the north end of the spent fuel pool, also consist of a supply plenum and exhaust plenum. The supply plenum is configured similar to the supply plenum provided in the original plant design and includes one prefilter, two heating coils, and one supply fan. The exhaust plenum is different from the original plant design in that it only contains one filter bank consisting of a prefilter and HEPA filter, and two common exhaust fans.

During normal plant operations, the Fuel Handling Area Ventilation System supplies filtered and heated (as needed) outside air to the fuel handling area. The exhaust fans draw air from the fuel handling area through the normally aligned prefilters and HEPA filters and discharge it to the unit stack by way of the main ventilation exhaust plenum.

BASES

BACKGROUND (continued)	During plant evolutions when the possibility for a fuel handling accident or fuel cask drop accident exist, the Fuel Handling Area Ventilation System is configured such that all fans are stopped except one exhaust fan in the original plant subsystem aligned to the "emergency filter bank." The "normal filter bank" in the original plant design is isolated by closing its associated inlet damper. Thus, in the event of a fuel handling accident, the fuel handling area atmosphere will be filtered for the removal of airborne fission products prior to being discharged to the outside environment. The Fuel Handling Area Ventilation System is discussed in the FSAR, Sections 9.8, 14.11 and 14.19 (Refs. 1, 2, and 3) because it may be used for normal, as well as post accident, atmospheric cleanup functions.
APPLICABLE SAFETY ANALYSES	The Fuel Handling Area Ventilation System is designed to mitigate the consequences of a fuel handling accident or fuel cask drop accident by limiting the amount of airborne radioactive material discharged to the outside atmosphere. The results and major assumptions used in the analysis of the fuel handling accident are presented in FSAR Section 14.19. For the purpose of defining the upper limit of the radiological consequences of a fuel handling accident, it is assumed that a fuel bundle is dropped during fuel handling activities and all the fuel rods in the equivalent of an entire assembly (216) fail. The bounding fuel handling accident is assumed to occur in containment two days after shutdown. No containment isolation is assumed to occur. As such, the released fission products escape to the environment with no credit for filtration. The results of this analysis have shown that the offsite doses resulting from this event are within the guideline of 10 CFR 100. In the event a fuel handling accident were to occur in the fuel handling area, the radioactive release would pass through the "emergency filter bank" significantly reducing the amount of radioactive material released to the environment. Thus, the consequences of a fuel handling accident in the fuel handling area are deemed acceptable with or without the "emergency filter bank" in operation since they are no more severe than the consequences of a fuel handling accident in containment.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The results and major assumptions used in the analysis of the fuel cask drop accident are presented in FSAR Section 14.11. For the purpose of defining the upper limit of the radiological consequences of a fuel cask drop accident, it is assumed that all 73 fuel assemblies in a 7 x 11 Westinghouse spent fuel pool rack with a minimum decay of 30 days are damaged and release their fuel rod gap inventories. Three fuel cask drop scenarios were analyzed to encompass all fuel cask drop events. They are:

1. A fuel cask drop onto 30 day decayed fuel with the Fuel Handling Area Ventilation System aligned for emergency filtration with a conservative amount of unfiltered leakage. All isolatable unfiltered leak path are assumed to be isolated prior to event initiation.

2. A fuel cask drop onto 30 day decayed fuel with the Fuel Handling Area Ventilation System aligned for emergency filtration with a conservative amount of unfiltered leakage. This scenario determined the maximum amount of non-isolatable unfiltered leakage than can exist and still meet offsite dose limits. This scenario also assumes isolation of isolable leak paths prior to event initiation.

3. A fuel cask drop onto 90 day decayed fuel without the Fuel Handling Area Ventilation System aligned for emergency filtration. This scenario needs no assumptions as to unfiltered leakage or post-accident unfiltered leak path isolation times since all radiation is assumed to be released unfiltered from the fuel handling area.

The results of the analysis show that the radiological consequences of a fuel cask drop in the spent fuel pool meet the acceptance criteria of Regulatory Guide 1.25 (Ref. 4) and NUREG-0800 Section 15.7.5 (Ref. 5) for all scenarios. In addition, the dose from all scenarios are less than 25% of the dose guidelines in 10 CFR 100. For scenario 2, the analysis shows that a maximum of 20% charcoal filter bypass from non-isolatable leak paths can be accommodated while still meeting 25% of the 10 CFR 100 guidelines.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Filtration of the fuel handling area atmosphere following a fuel handling accident is not necessary to maintain the offsite doses within the guidelines of 10 CFR 100. Thus, a total system failure would not impact the margin of safety as described in the safety analysis. However, analysis has shown that post accident filtration by the Fuel Handling Area Ventilation System provides significant reduction in offsite doses by limiting the release of airborne radioactivity. Therefore, for the fuel handling accident, the Fuel Handling Area Ventilation System satisfies Criterion 4 of 10 CFR 50.36(c)(2).

Filtration of the fuel handling area atmosphere following a fuel cask drop on irradiated fuel assemblies with < 90 days decay is required to maintain the offsite doses within the guidelines of 10 CFR 100. Therefore, for the fuel cask drop accident, the Fuel Handling Area Ventilation System satisfies Criterion 3 of 10 CFR 50.36(c)(2).

LCO

The LCO for the Fuel Handling Area Ventilation System ensures filtration of the fuel handling area atmosphere is immediately available in the event of a fuel handling accident, or a fuel cask drop accident. As such, the LCO requires the Fuel Handling Area Ventilation System to be OPERABLE with one fuel handling area exhaust fan aligned to the "emergency filter bank" and in operation.

The Fuel Handling Area Ventilation System is considered OPERABLE when the individual components necessary to control exposure in the fuel handling building are OPERABLE. The Fuel Handling Area Ventilation System is considered OPERABLE when:

a. One exhaust fan is aligned to the "emergency filter bank" and in operation to ensure the air discharged to the main ventilation exhaust plenum has been filtered. Operation of only one fuel handling area exhaust fan ensures the design flow rate of the "emergency filter bank" is not exceeded.

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

BASES

LCO (continued)

c. Ductwork and dampers are OPERABLE, and air circulation can be maintained. Inclusive to the requirement to align the "emergency filter bank" is that the "normal filter bank" is isolated by its associated inlet damper to prevent the release of unfilted air.

APPLICABILITY

The Fuel Handling Area Ventilation System must be Operable, aligned, and in operation whenever the potential exists for an accident that results in the release of radioactive material to the fuel handling area atmosphere that could exceed previously approved offsite dose limits if released unfiltered to the outside atmosphere. As such, the Fuel Handling Area Ventilation System is required; during movement of irradiated fuel assemblies in the fuel handling building when irradiated fuel assemblies with < 30 days decay time are in the fuel handling building; during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in containment when irradiated fuel assemblies with < 30 days decay time are in the containment with the equipment hatch open, and during movement of a fuel cask in or over the spent fuel pool when irradiated fuel assemblies with < 90 days are in the fuel handling building.

The requirement for the Fuel Handling Area Ventilation System does not apply during movement of irradiated fuel assemblies or CORE ALTERATIONS when all irradiated fuel assemblies in the fuel handling building, or all irradiated fuel assemblies in the containment with the equipment hatch open, have decayed for 30 days or greater since the dose consequences from a fuel handling accident would be of the same magnitude without the filters operating as the dose consequences would be with the filters operating and two days decay. In addition, the requirement for the Fuel Handling Area Ventilation System does not apply during fuel cask movement when all irradiated fuel assemblies in the fuel handling building have decayed 90 days or greater since the dose consequences remain less than 25% of the guidelines of 10 CFR 100.

BASES

ACTIONS

A.1 and A.2

If the Fuel Handling Area Ventilation System is not aligned to the "emergency filter bank", or one exhaust fan is not in operation, or the system is inoperable for any reason, action must be taken to place the unit in a condition in which the LCO does not apply. Therefore, activities involving the movement of irradiated fuel assemblies, CORE ALTERATIONS, and movement of a fuel cask in or over the spent fuel pool, must be suspended immediately to minimize the potential for a fuel handling accident.

The suspension of fuel movement, CORE ALTERATIONS, and fuel cask movement shall not preclude the completion of placing a fuel assembly, core component, or fuel cask in a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies the performance of Fuel Handling Area Ventilation System filter testing in accordance with the Ventilation Filter Testing Program. The Fuel Handling Area Ventilation System filter tests are in accordance with the Regulatory Guide 1.52 (Ref. 6) as described in Ventilation Filter Testing Program. The Ventilation Filter Testing Program includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the Ventilation Filter Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.2

This SR verifies the Fuel Handling Area Ventilation System has not degraded and is operating as assumed in the safety analysis. The flow rate is periodically tested to verify proper function of the Fuel Handling Ventilation System. When aligned to the "emergency filter bank", the Fuel Handling Area Ventilation System is designed to reduce the amount of unfiltered leakage from the fuel handling building which, in the event of a fuel handling accident, lowers the dose at the site boundary to well within the guidelines of 10 CFR 100. The Fuel Handling Area Ventilation System is designed to lower the dose to these levels at a flow rate of $\geq$ 5840 cfm and $\leq$ 8760 cfm. The Frequency of 18 months is consistent with the test for filter performance and other filtration SRs.

REFERENCES

1. FSAR, Section 9.8
2. FSAR, Section 14.11
3. FSAR, Section 14.19
4. Regulatory Guide 1.25, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurize Reactors.
5. NUREG-0800 Section 15.7.5, Spent Fuel Cask Drop Accidents.
6. Regulatory Guide 1.52, Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorbtion Units of Light-Water-Cooled Nuclear Power Plants.

BASES

LCO

Two ESRV Damper trains are required to be OPERABLE to ensure that each engineered safeguards room isolates upon receipt of its respective high radiation alarm. Total system failure could result in the atmospheric release from the engineered safeguards rooms exceeding the required limits in the event of a Design Basis Accident (DBA).

An ESRV Damper train is considered OPERABLE when its associated radiation monitor, instrumentation, ductwork, valves, and dampers are OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ESR-Damper trains are required to be OPERABLE consistent with the OPERABILITY requirements of the Emergency Core Cooling System (ECCS).

In MODES 5 and 6, the ESRV Damper trains are not required to be OPERABLE, since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

Condition A addresses the failure of one or both ESRV Damper trains. Operation may continue as long as action is immediately initiated to isolate the affected engineered safeguards room. With the inlet and exhaust dampers closed, or if the inlet and outlet ventilation plenums are adequately sealed, the engineered safeguards room is isolated and the intended safety function is achieved, since the potential pathway for radioactivity to escape to the environment from the engineered safeguards room has been minimized.

The Completion Time for this Required Action is commensurate with the importance of maintaining the engineered safeguards room atmosphere isolated from the outside environment when the ECCS pumps are circulating primary coolant after an accident.

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Pool (SFP) Water Level

BASES

BACKGROUND

The minimum water level in the SFP meets the assumptions of iodine decontamination factors following a fuel handling or cask drop accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the SFP design is given in the FSAR, Section 9.11 (Ref. 1), and the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section 9.4 (Ref. 2). The assumptions of fuel handling and fuel cask drop accidents are given in the FSAR, Section 14.19 and 14.11 (Refs. 3 and 4), respectively.

APPLICABLE SAFETY ANALYSES

The minimum water level in the SFP meets the assumptions of fuel handling or fuel cask drop accident analyses described in References 3 and 4 and are consistent with the assumptions of Regulatory Guide 1.25 (Ref. 5). The resultant 2 hour thyroid dose to a person at the exclusion area boundary is well within the 10 CFR 100 (Ref. 6) limits.

Reference 5 assumes there is 23 ft of water between the top of the damaged fuel assembly and the fuel pool surface for a fuel handling or fuel cask drop accident. This LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single assembly, dropped and lying horizontally on top of the spent fuel racks, there may be < 23 ft of water above the top of the assembly and the surface, by the width of the assembly. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rods fail from a hypothetical maximum drop.

The SFP water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2).

BASES

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool until a complete spent fuel pool verification of the stored assemblies has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

A.1, A.2.1, and A.2.2

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to within limit. Alternately, beginning a verification of the SFP fuel locations to ensure proper locations of the fuel can be performed.

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or used (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store 892 irradiated fuel assemblies, which includes storage for failed fuel canisters. The spent fuel storage racks are grouped into two regions, Region I and Region II per Figure 3.7.16-1. The racks are designed as a Seismic Category I structure able to withstand seismic events. Region I contains racks in the spent fuel pool having a 10.25 inch center-to-center spacing and a single rack in the north tilt pit having a 11.25 inch by 10.69 inch center-to-center spacing. Region II contains racks in both the spent fuel pool and the north tilt pit having a 9.17 inch center-to-center spacing. Because of the smaller spacing and poison concentration, Region II racks have more limitations for fuel storage than Region I racks. Further information on these limitations can be found in Section 4.0, "Design Features." These limitations (e.g., enrichment, burnup) are sufficient to maintain a k_{eff} of ≤ 0.95 for spent fuel of original enrichment of up to 4.40%.

APPLICABLE SAFETY ANALYSES

The spent fuel storage facility is designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans.

The spent fuel assembly storage satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Table 3.7.16-1, in the accompanying LCO, ensures that the k_{eff} of the spent fuel pool will always remain < 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool according to Table 3.7.16-1, in the accompanying LCO. Fuel assemblies not meeting the criteria of Table 3.7.16-1 shall be stored in accordance with Specification 4.3.1.1.

BASES

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region II of either the spent fuel pool or the north tilt pit.

ACTIONS

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

A.1

When the configuration of fuel assemblies stored in Region II the spent fuel pool is not in accordance with Table 3.7.16-1, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Table 3.7.16-1.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Table 3.7.16-1 in the accompanying LCO prior to placing the fuel assembly in a Region II storage location.

REFERENCES

None

B 3.7 PLANT SYSTEMS

B 3.7.17 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Primary Coolant System (PCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives, and thus is indication of current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak of primary coolant at the limit of 1.0 μCi/gm as assumed in the safety analyses with exception of the control rod ejection analysis which assumes 0.6 gpm. LCO 3.4.13, "PCS Operational LEAKAGE," is more restrictive in that the limit for a primary to secondary tube leak is 0.3 gpm. The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and primary coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

Operating a plant at the allowable limits would result in a 2 hour Exclusion Area Boundary (EAB) exposure well within the 10 CFR 100 (Ref. 1) limits.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the PCS and steam generators are at low pressure or depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant is an indication of a problem in the PCS and contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR ensures that the secondary specific activity is within the limits of the accident analysis. A gamma isotope analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in primary coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

7.5 AUXILIARY FEEDWATER (AFW) SYSTEM

3.~~5~~ ~~STEAM AND FEEDWATER SYSTEMS (Cont'd)~~ A.1

~~3.5.2 (Continued)~~

<SEE 3.7.6>

~~c. The fire water makeup to the Auxiliary Feedwater Pump Suction (P-8A and P-8B) may be inoperable for a period of 7 days provided the pump service water makeup to P-8C, pump P-8C, and its corresponding flow control valves are operable.~~

~~d. The service water makeup to the Auxiliary Feedwater Pump Suction (P-8C) may be inoperable for a period of 7 days provided the fire water makeup to P-8A & P-8B, pumps P-8A and P-8B and their corresponding flow control valves are operable.~~

COND B e. One ~~flow control valve on each~~ train may be inoperable for a period of 72 hours provided the ~~corresponding redundant flow control valve~~ ~~and a pump in the other pipe~~ train ~~are~~ IS operable. L.4

APPL 3.5.3 With the Primary Coolant System at a temperature greater than 300°F and if the system does not satisfy the requirements of Specification 3.5.1 or the conditions of Specification 3.5.2 except as noted in Specification 3.5.4, the reactor shall be placed ~~in hot standby within 6 hours, hot shutdown~~ within ~~the following~~ 6 hours and in ~~cold shutdown~~ MODE 4 ~~within~~ ~~the following~~ 24 hours. M.1 RA C.1 RA C.2 M.3 L.2

A.5 MODE 3 A.8 <INSERT>

COND D RA D.1 3.5.4 With all Auxiliary Feedwater Pumps inoperable immediately initiate corrective action to restore at least one Auxiliary Feedwater Pump to OPERABLE status as soon as possible ~~and reduce power within 24 hours to the lowest stable power level consistent with reliable Main Feedwater System operation.~~ L.5

<ADD. RA D1 NOTE> L.5

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August 12, 1994
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03/15/99

M.1

3.3 EMERGENCY CORE COOLING SYSTEM (Cont'd) A.1

3.3.2 During power operation the requirements of 3.3.1 may be modified to allow one of the following conditions to be true at any one time. If the system is not restored to meet the requirements of 3.3.1 within the time period specified below, the reactor shall be placed in a ~~hot shutdown~~ MODE 3 condition within ~~12~~ 6 hours. If the requirements of 3.3.1 are not met within an additional ~~48~~ hours, the reactor shall be placed in a ~~cold shutdown~~ MODE 5 condition ~~within 24 hours~~. 30 M.4

RAB.1 M.3 RAB.2 M.3 A.4

SEE 3.5

a. One safety injection tank may be inoperable for a period of no more than one hour.

b. One low-pressure safety injection pump may be inoperable provided the pump is restored to operable status within 24 hours.

c. One high-pressure safety injection pump may be inoperable provided the pump is restored to operable status within 24 hours.

SEE 3.5

d. One shutdown heat exchanger and one component cooling water heat exchanger may be inoperable for a period of no more than 24 hours. M.6

SEE 3.5

e. Any valves, interlocks or piping directly associated with one of the above components and required to function during accident conditions shall be deemed to be part of that component and shall meet the same requirements as listed for that component.

SEE 3.5

f. Any valve, interlock or pipe associated with the safety injection and shutdown cooling system and which is not covered under 3.3.2e above but, which is required to function during accident conditions, may be inoperable for a period of no more than 24 hours.

Amendment No. ~~21~~, ~~51~~, 172
September 26, 1996

Revised
03/15/99

~~4.2 EQUIPMENT SAMPLING AND TESTS~~

~~Table 4.2.3~~

~~VENTILATION SYSTEM TESTS~~

A.1

<SEE 3.7.12>

The Control Room Ventilation and Isolation System ~~and the Fuel Storage Area HEPA Charcoal Exhaust System~~ shall be demonstrated to be OPERABLE by the following tests:

<SEE 3.7.12>

3.7.10.2 1. Performing required Control Room Ventilation ~~and Fuel Storage Area filter testing~~ in accordance with the Ventilation Filter Testing Program.

A.8 ~~2. At least once per refueling cycle by:~~

ACTUAL OR SIMULATED ACTUATION SIGNAL A.7

SR 3.7.10.3 a. Verifying that on a ~~containment high pressure and high-radiation test signal~~, the Control Room Ventilation system automatically switches into the emergency mode of operation with flow through the HEPA filter and charcoal adsorber bank.

ADJACENT AREA A.6 0.125 INCHES A.6

SR 3.7.10.4 b. Verifying that the Control Room Ventilation system maintains the Control Room at a positive pressure ≥ ~~1/8 inch~~ WG relative to the ~~outside atmosphere~~ during system emergency mode operation. <AT A FLOW RATE ≥ 3040 AND ≤ 3520 cfm> M.2

<SEE 3.7.12> ~~c. Verifying that the Fuel Pool Ventilation System is OPERABLE by initiating flow through the HEPA filter and charcoal adsorbers from the control room.~~

<ADD SR 3.7.10.1> M.1

<ADD SR 3.7.10.3 NOTE> L.1

3.7 PLANT SYSTEMS A.1

3.7.12 Fuel Handling Area Ventilation System

3.8 REFUELING OPERATIONS (Continued)

g. During reactor vessel head removal and while refueling operations are being performed in the reactor, the refueling boron concentration shall be maintained in the primary coolant system and shall be checked by sampling on each shift.

h. Direct communication between personnel in the control room and at the refueling machine shall be available whenever changes in core geometry are taking place.

(SEE 3.9 3.3)

3.8.2 If any of the conditions in 3.8.1 are not met, all refueling operations shall cease immediately, work shall be initiated to satisfy the required conditions and no operations that may change the reactivity of the core shall be made.

3.8.3. Refueling operation shall not be initiated before the reactor core has decayed for a minimum of 48 hours if the reactor has been operated at power levels in excess of 2% rated power.

M.1

3.8.4 LCO 3.7.12 The ventilation system and charcoal filter A.2 in the fuel storage building shall be operating A.2 (OPERABLE and aligned in the emergency filtration mode with one exhaust fan) whenever irradiated fuel which has decayed less than 30 days is being handled by either of the following operations: A.3

Applic

a. Core Alterations ~~Refueling operation~~ with the equipment door open, or

b. Fuel handling in the fuel storage building.

LA.1

RA A.1 If both fans are unavailable, any fuel movements in progress shall be completed and further fuel movements over the spent fuel storage pool shall be terminated until one fan is returned to service. A.4

3.8.5. When spent fuel which has decayed less than one year is placed in the tilt pit storage racks, the bulk water temperature in the tilt pit storage area must be monitored continuously to assure that the water temperature does not exceed 150°F. Monitoring will continue for 24 hours after any addition of fuel to the main pool or the tilt pit or when a failure of the spent fuel pool cooling system occurs.

(SEE 3.9)

Basis

The equipment and general procedures to be utilized during refueling are discussed in the FSAR. Detailed instructions, the above specifications, and the design of the fuel handling equipment incorporating built-in interlocks and safety features provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.(1) Whenever changes are not being made in core geometry, one flux monitor is sufficient. This

<ADD Applicability FOR Fuel CASK movement> M.2

<ADD COND A 2nd PART> M.1

A.1

3.8 REFUELING OPERATIONS

Applicability

Applies to operating limitations during refueling operations.

Objective

To minimize the possibility of an accident occurring during refueling operations that could affect public health and safety.

3.8.1 Specifications

The following conditions shall be satisfied during any refueling operations:

SEE 3.9

a. ~~One door of the emergency air lock shall be properly closed.~~ Whenever both doors of the personnel air lock are open during refueling operations, ~~the equipment door shall be open and~~ the ventilating system and charcoal filter in the fuel storage building shall be operating.

LCO 3.7.12

b. All automatic containment isolation valves shall be operable or at least one valve in each line shall be closed.

SEE 3.9 3.3

c. The containment venting and purge systems, including two radiation monitors that initiate isolation, shall be tested and verified to both be operable immediately prior to refueling operations. The two monitors shall be located on the containment fuel handling area level (elevation 649'), shall be part of the plant area monitoring system and shall employ one-out-of-two logic for isolation. During normal operation, these monitors will not initiate an isolation signal. A switch shall be provided so that isolation action can be initiated during refueling only.

SEE 3.3 3.9

d. Radiation levels in the containment and spent fuel storage areas shall be monitored continuously.

SEE 3.9

e. Whenever core geometry is being changed, neutron flux shall be continuously monitored by at least two source range neutron monitors, with each monitor providing continuous visual indication in the control room. When core geometry is not being changed, at least one source range neutron monitor shall be in service.

f. At least one shutdown cooling pump and heat exchanger shall be in operation.

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Amendment No. 34
January 27, 1978

Revised
03/15/99

4.2 ~~EQUIPMENT SAMPLING AND TESTS~~

~~Table 4.2.3~~

~~VENTILATION SYSTEM TESTS~~

A.1

~~The Control Room Ventilation and Isolation System and~~ the Fuel Storage Area HEPA/Charcoal Exhaust System shall be demonstrated to be OPERABLE by the following tests: (See 3.7.10)

SR 3.7.12.1 1. Performing required ~~Control Room Ventilation and~~ Fuel Storage Area filter testing in accordance with the Ventilation Filter Testing Program.

2. At least once per refueling cycle by:

~~a. Verifying that on a containment high-pressure and high-radiation test signal, the Control Room Ventilation system automatically switches into the emergency mode of operation with flow through the HEPA filter and charcoal adsorber bank.~~

~~b. Verifying that the Control Room Ventilation system maintains the Control Room at a positive pressure ≥ 1/8 inch WG relative to the outside atmosphere during system emergency mode operation.~~

(See 3.7.10)

L.1

c. Verifying that the Fuel Pool Ventilation System is OPERABLE by initiating flow through the HEPA filter and charcoal adsorbers from the control room.

<ADD SR 3.7.12.2> M.3

A.6 A Note has been added to SR 3.7.5.3 which states "Not required to be met in MODES 2 or 3 when AFW is in operation." This Note is needed to prevent unnecessary entering of ACTIONS for LCO 3.7.5 during the startup or shutdown of the plant for not being able to meet the SR. Palisades uses the AFW system for steam generator level control during startup and shutdown in MODES 2, 3, and 4. During these operations the flow control valves used are in manual, and will not open automatically when an actuation signal is received, which would fail the SR. This change is administrative because CTS 4.9.b.1 states "each valve to actuates to its correct position (or that the specified flow is established) upon receipt of a simulated auxiliary feedwater pump start signal." During startup or shutdown the valves are providing the proper flow for the existing plant condition. This Note is appropriate because the valves are needed to be throttled in these conditions to prevent overfilling of the steam generators due to low steam flow conditions, also the Note clarifies current licensing basis requirements.

A.7 This change adds the additional "inservice requirements" as described in ASME Code, Section XI to CTS 4.9.a.1 and 2. This change is administrative in that these requirements are performed by current surveillances and also this change only combines the two requirements, Code and TS. This change is consistent with NUREG-1432.

A.8 CTS 3.5.4 provides corrective actions in the event all AFW pumps are inoperable. In this case, the capability to provide the required AFW flow to either steam generator has been lost. Proposed ITS 3.7.5 Condition D also provides corrective actions when the capability to provide the required AFW flow to either steam generator has been lost. Condition D is stated as "two AFW trains inoperable with both steam generators having less than 100% of the AFW flow equivalent to a single Operable AFW train available in Mode 1, 2, or 3 or, (the) required AFW train inoperable in Mode 4." Since the AFW system inoperabiliy addressed in ITS 3.7.5 Condition D (a loss of AFW function) is equivalent to the condition presented in CTS 3.5.4, this change has been characterized as administrative in nature.

ADMINISTRATIVE CHANGES (A)

A.1 All reformatting and renumbering are in accordance with NUREG-1432. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1432. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or implied) to the TS. Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1432. Since the design is already approved by the NRC, adding more details does not result in a technical change.

A.2 CTS 3.4.2 and 3.4.3 require that if a component(s) listed in Specification 3.4.1 is inoperable for more than the time specified, the plant must be placed in HOT SHUTDOWN. In proposed ITS 3.7.7 Required Action B.1, the CTS term is replaced with MODE 3. This is considered to be an administrative change since the effect on operations is similar. This change is consistent with NUREG-1432.

A.3 CTS 3.4.4 specifies that valves, interlocks and piping that are directly associated with the "above" (CTS 3.4.1) components shall meet the same requirements as listed for that component. CTS 3.4.5 specifies that valves, interlocks and piping which is associated with the containment cooling system and not covered by CTS 3.4.4 may be inoperable for no more than 24 hours if it is required to function during an accident. These requirements are addressed by the definition of OPERABILITY which requires that all associated equipment be OPERABLE. In the proposed ITS, all equipment in a particular train which is required to function during an accident must be OPERABLE and all equipment in the train will have the same Completion Time. This is an administrative change since the requirement remains that all equipment in a train of containment cooling must be OPERABLE. This change is consistent with NUREG-1432.

A.4 CTS 3.3.2, 3.4.2, and 3.4.3 require that with the Required Action and associated Completion Time not met the plant must be placed in COLD SHUTDOWN. In proposed ITS 3.7.7 Required Action B.2, the CTS term is replaced with MODE 5. This is considered to be an administrative change since the effect on operations is similar. This change is consistent with NUREG-1432.

A.5 CTS 3.4.3 states "....Continued power operation with one component out of service shall be as specified in Section 3.4.2, with the permissible period in inoperability starting at the time that the first of the two components became inoperable." This explanatory information on the usage rules of technical specifications is addressed in the proposed ITS Section 1.3, "Completion Times," and does not need to be addressed in the Actions of proposed ITS 3.7.7. This is considered to be an administrative change since the requirements on complying with the completion times is addressed in the proposed ITS. This change is consistent with NUREG-1432.

A.6 The Note added to proposed SR 3.7.7.1 to aid the operator in the prevention of entering an inappropriate LCO. The Note reminds the operator that loss of CCW flow to a component may render that component inoperable but does not affect the OPERABILITY of the CCW System. This change is considered administrative that this is a clarifier to the operator to prevent confusion. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE (M)

M.1 CTS 3.3.1, 3.3.2, 3.4.1, and 3.4.2 establish the Applicability for the various components which comprise the CCW by stating that "the reactor shall not be made critical.... unless all of the following conditions are met." The Applicability of the CCW in proposed ITS 3.7.7 is MODES 1, 2, 3, and 4. As such, the requirements associated with CTS 3.3.1, 3.3.2, 3.4.1, and 3.4.2 have been revised to be more restrictive by requiring the CCW to also be OPERABLE during the additional MODES 3 and 4. SRs 3.7.7.2 and 3.7.7.3 are modified by a Note which states that these SRs are not required to be met in MODE 4. This is due to the instrumentation providing the signals are not required in MODE 4. This change keeps consistency with ITS 3.3.3, "ESF Instrumentation," and current licensing basis. This change is an additional restriction on plant operations and is consistent with NUREG-1432.

LESS RESTRICTIVE CHANGES (L)

L.1 CTS 4.2, Table 4.2.3, item 2.a requires a verification that the Control Room Ventilation system automatically switches into the emergency mode of operation on a "containment high pressure and high radiation test signal." The Applicability of this requirement is "above COLD SHUTDOWN, during REFUELING OPERATIONS, during movement of irradiated fuel assemblies, and during movement of a fuel cask in or over the Spent Fuel Pool." Proposed SR 3.7.10.3 requires a verification that each CRV Filtration train actuates on an actual or simulated actuation signal. The requirement and Applicability of CTS 4.2, Table 4.2.3, item 2.a is similar to the requirement and Applicability of SR 3.7.10.3. However, SR 3.7.10.3 is further modified by a Note which states that the SR is "not required to be met during movement of irradiate fuel assemblies in the SFP, or during movement of a fuel cask in or over the SFP." The purpose of this Note is to exclude the requirement of the SR during those plant evolutions in which no instrumentation is available to actuate the CRV System. The CRV System is designed to automatically switch to the emergency mode of operation on a "containment high pressure or containment high radiation signal." The instruments used to initiate these actuation signals are not capable of detecting an increase in radiation levels in the fuel handling building, and as such, can not provide automatic actuation of the CRV System in the event of a fuel handling accident or cask drop accident in the SFP. Therefore, the addition of the Note in SR 3.7.10.3 establishes consistency with the design of the CRV System and the requirement of the SR. During movement of irradiate fuel assemblies in the SFP, or during movement of a fuel cask in or over the SFP, manual operator action is necessary to initiate the emergency filtration mode of the CRV System.

ADMINISTRATIVE CHANGES (A)

A.1 All reformatting and renumbering are in accordance with NUREG-1432. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1432. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or implied) to the TS. Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1432. Since the design is already approved by the NRC, adding more details does not result in a technical change.

A.2 CTS 3.8.4 requires that the "ventilation system and charcoal filter" be in operation during specified conditions. Proposed LCO 3.7.12 has the same requirements, but replaces "charcoal filter...be operating" with the phrase "aligned in the emergency filtration mode with one exhaust fan in operation." This change is considered editorial in that the wording describes the same components and provides a clearer description to the operator of what needs to be performed.

A.3 The CTS definition of REFUELING OPERATION forms the basis for the proposed ITS definition of CORE ALTERATIONS. This change is considered to be administrative since the term "CORE ALTERATIONS" is used to simply replace "REFUELING OPERATION," any additional clarification provided by the new definition gives additional clarification on its application. This change is consistent with NUREG-1432.

A.4 CTS 3.8.4 requires fuel movements be terminated if both (Fuel Handling Area exhaust) fans are unavailable "until one fan is returned to service." Proposed ITS 3.7.12 also requires fuel movements be suspended if the Fuel Handling Area Ventilation system is not in operation (i.e., one exhaust fan running) but, does not contain the stipulation "until one fan is returned to service." Inclusion of the phrase "until one fan is returned to service" is unnecessary since once a fan is returned to service compliance with the LCO is restored. Thus, omission of this phrase is considered administrative in nature since it does not alter the original intent of the CTS requirement. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE (M)

M.1 The requirements of CTS 3.8.4 have been revised to address the Operability of the Fuel Handling Area Ventilation system. LCO 3.7.12 requires, in part, that the Fuel Handling Area Ventilation system must be Operable. Although CTS 4.2, Table 4.2.3 item 2.c contains a test to verify the Operability of the Fuel Pool Ventilation System, the CTS does not contain an explicit requirement for Operability. As such, a new Condition, Required Action and Completion Time has been provided to address the situation when the Fuel Handling Area Ventilation system is inoperable. This change is consistent with NUREG-1432.

M.2 CTS 3.8.4 requires Fuel Handling Area Ventilation System to be in operation during REFUELING OPERATIONS with the equipment door open and also fuel handling in the fuel storage building, if fuel in either locations has decayed < 30 days. Proposed ITS 3.7.12 requires Fuel Handling Area Ventilation System in operation during CORE ALTERNATIONS (see DOC A.3) with the equipment door open, movement of irradiated fuel assemblies in the containment with the equipment door open, and movement of irradiated fuel assemblies in the fuel handling building, if fuel in either location has decayed < 30 days. In addition, proposed ITS 3.7.12 requires the operation of the Fuel Handling Area System during movement of a fuel cask in or over the spent fuel pool when irradiated fuel assemblies with < 90 days decay time are in the fuel handling building. This change clarifies the conditions when Fuel Handling Area Ventilation System shall be in operation and adds the new requirements for Fuel Handling Area System operation. The Design Basis Accidents are fuel handling accidents and a cask drop accident, both of which involve the release of airborne radioactive particles to the fuel handling area. The change is considered more restrictive because of the added requirement on plant operation.

M.3 CTS does not have a specific Surveillance for Fuel Handling Area Ventilation System as presented in proposed ITS SR 3.7.12.3. The proposed surveillance requires additional testing to be performed on the Fuel Handling Area Ventilation System to verify OPERABILITY. The proposed SR verifies that the Fuel Handling Area Ventilation System has not degraded and is operating at the flow rate assumed in the analysis. This change is considered more restrictive since the system is not required by CTS to be tested in this manner.

LESS RESTRICTIVE CHANGES - REMOVAL OF DETAILS TO LICENSEE CONTROLLED DOCUMENTS (LA)

LA.1 CTS 3.8.4 states if both (Fuel Handling Area Ventilation exhaust) fans are unavailable, then "any fuel movements in progress shall be completed...". The intent of this statement is to clarify that the actions do not preclude the movement of a fuel assembly to a safe position which ultimately minimizes the potential for a fuel handling accident. In proposed ITS 3.7.12 this same clarification is provided in the Bases. Placing this information in the Bases is acceptable since these details are not pertinent to the actual requirements, but merely describe safe operating practices. Placing these details in the Bases provides adequate assurance that they will be maintained since the Bases are controlled by the Bases Control Program proposed in ITS Chapter 5.0. This change is consistent with NUREG-1432.

LESS RESTRICTIVE CHANGES (L)

L.1 CTS 4.2, Table 4.2.3 item 2.c requires a verification "that the Fuel Pool Ventilation System is Operable by initiating flow through the HEPA filter and charcoal adsorbers from the control room at least once per refueling cycle." In proposed ITS 3.7.12, this surveillance requirement has been deleted since it is redundant to the actual requirement of the LCO. LCO 3.7.12 requires that the Fuel Handling Area Ventilation System be OPERABLE and aligned in the emergency filtration mode with one exhaust fan in operation. As such, in order to establish compliance with the LCO, flow must be initiated through the emergency filter bank which includes the HEPA and charcoal adsorbers. Specifying that flow be initiated from the control room is irrespective of the safety function performed by the Fuel Handling Area Ventilation System since the system must be aligned in the emergency filtration mode prior to movement of any irradiated fuel assemblies. Therefore, since the requirement of LCO 3.7.12 fulfills the requirement CTS 4.2, Table 4.2.3 item 2.c, this surveillance requirement can be deleted without an impact of safety.

ADMINISTRATIVE CHANGES (A)

A.1 All reformatting and renumbering are in accordance with NUREG-1432. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1432. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or implied) to the TS. Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1432. Since the design is already approved by the NRC, adding more details does not result in a technical change.

A.2 The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content consistent with NUREG-1432. The revised Bases are shown in the proposed Technical Specification Bases.

A.3 CTS 3.1.5c requires that with specific activity of the secondary coolant >0.1 μCi/gram DOSE EQUIVALENT I-131, the plant must be placed in HOT SHUTDOWN. In proposed ITS 3.7.17 Required Action A.1, the CTS term is replaced with MODE 3. This is considered to be an administrative change since the effect on operations is similar. This change is consistent with NUREG-1432.

A.4 CTS 3.1.5c requires that with specific activity of the secondary coolant >0.1 μCi/gram DOSE EQUIVALENT I-131, the plant must be placed in COLD SHUTDOWN. In proposed ITS 3.7.17 Required Action A.2, the CTS term is replaced with MODE 5. This is considered to be an administrative change since the effect on operations is similar. This change is consistent with NUREG-1432.

A.5 CTS 3.1.5c requires that with specific activity of the secondary coolant >0.1 μCi/gram DOSE EQUIVALENT I-131, the plant must be placed in COLD SHUTDOWN. In proposed ITS the term is replaced with MODE 5 (see DOC A.4). In proposed ITS 3.7.17 Applicability, the Specification is applicable in MODES 1, 2, 3, and 4. Placing the plant in COLD SHUTDOWN in CTS and having the Applicability in MODES 1, 2, 3, and 4 in proposed ITS is basically the same. This change is considered to be an administrative change since the effect on operations is similar. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE (M)

M.1 CTS 4.2 Table 4.2.1, item 7a, requires the specific activity of the secondary coolant system to be determined once per 31 days whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit, and once per 6 months whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. Proposed ITS SR 3.7.17.1 will require the specific activity to be determined once per 31 days. The proposed ITS SR will not contain the allowance to extend the SR interval to 6 months whenever the gross activity determination indicates iodine concentration below 10% of the allowable limit. This change does not adversely affect safety because the 31 day interval ensures that the specific activity is checked frequently enough to establish a trend to identify secondary activity problems in a timely manner. Deleting an allowance to extend an SR interval constitutes a more restrictive change. This change is consistent with NUREG-1432.

LESS RESTRICTIVE CHANGES - REMOVAL OF DETAILS TO LICENSEE CONTROLLED DOCUMENTS (LA)

There were no "Removal of Details" associated with this specification.

LESS RESTRICTIVE CHANGES (L)

L.1 CTS 4.2, Table 4.2.1 requires a sample of secondary coolant to be analyzed for gross radioactivity 3 times every 7 days with a maximum of 72 hours between samples. The CTS contains no LCO, limiting value, or Required Actions for secondary coolant gross radioactivity, only that sampling is required. The intent of this surveillance is to monitor the iodine concentration in the secondary coolant in order to determine the frequency at which an isotopic analysis for Dose Equivalent I-131 concentration in the secondary coolant is performed. The CTS requires an isotopic analysis for Dose equivalent I-131 of the secondary coolant once per 31 days whenever the gross activity indicates iodine concentrations greater than 10% of the allowable limit or, once per 6 months whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. However as discussed in DOC M.1 for this specification, the extended surveillance interval of 6 months for the determination of Dose Equivalent I-131 in the secondary coolant has been proposed for deletion and that future testing be performed every 31 days. Thus, the need to perform sampling of the secondary coolant for gross radioactivity is no longer necessary and has been delete in the ITS. This change is acceptable since gross radioactivity in the secondary coolant is not evaluated for radiological consequences in any of the accidents assumed in the FSAR, and the concentration of the Dose Equivalent I-131 in the secondary coolant will continue to be determined at an appropriate frequency. In addition, radiation monitoring instrumentation, controlled in accordance with the Offsite Dose Calculation Manual (e.g., SG blowdown monitors and condenser off gas monitor), is available to monitor increases in the radioactivity levels in the secondary coolant. This change is consistent with NUREG-1432.

LESS RESTRICTIVE CHANGE L.1

CTS 4.2, Table 4.2.3 item 2.c requires a verification "that the Fuel Pool Ventilation System is Operable by initiating flow through the HEPA filter and charcoal adsorbers from the control room at least once per refueling cycle." In proposed ITS 3.7.12, this surveillance requirement has been deleted since it is redundant to the actual requirement of the LCO. LCO 3.7.12 requires that the Fuel Handling Area Ventilation System be OPERABLE and aligned in the emergency filtration mode with one exhaust fan in operation. As such, in order to establish compliance with the LCO, flow must be initiated through the emergency filter bank which includes the HEPA and charcoal adsorbers. Specifying that flow be initiated from the control room is irrespective of the safety function performed by the Fuel Handling Area Ventilation System since the system must be aligned in the emergency filtration mode prior to movement of any irradiated fuel assemblies. Therefore, since the requirement of LCO 3.7.12 fulfills the requirement CTS 4.2, Table 4.2.3 item 2.c, this surveillance requirement can be deleted without an impact of safety.

1. **Does the change involve a significant increase in the probability or consequence of an accident previously evaluated?**

   Analyzed events are assumed to be initiated by the failure of plant structures, systems, or components. The proposed change eliminates the requirement to perform a test on the Fuel Pool Ventilation System on the basis that the intent of the test is adequately fulfilled by complying with the associated LCO prior to establishing a condition where the system would be required to function. The proposed change does not involve a change to any accident initiators or precursor. Therefore, this change does not involve a significant increase in the probability of an accident previously evaluated.

   The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event. The proposed change does not alter the function or operability of the Fuel Pool Ventilation System. As such, the consequences of an accident involving operation of the Fuel Pool Ventilation System remain unchanged. Therefore this change does not involve a significant increase in the consequence of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not involve a physical alteration of the plant. No new equipment is being introduced, and no installed equipment is being operated in a new or different manner. The proposed change does not alter the accident mitigative function of the Fuel Pool Ventilation System, nor does it create an opportunity for new or different accident beyond those previously evaluated. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The margin of safety is determined by equipment design, operating parameters, and the set points at which automatic actions are initiated within analyzed limits. There are no design changes or equipment performance parameter changes associated with this change. No accident or transient analysis are affected by this change. Therefore, this change does not involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L.1

CTS 4.2, Table 4.2.1 requires a sample of secondary coolant to be analyzed for gross radioactivity 3 times every 7 days with a maximum of 72 hours between samples. The CTS contains no LCO, limiting value, or Required Actions for secondary coolant gross radioactivity, only that sampling is required. The intent of this surveillance is to monitor the iodine concentration in the secondary coolant in order to determine the frequency at which an isotopic analysis for Dose Equivalent I-131 concentration in the secondary coolant is performed. The CTS requires an isotopic analysis for Dose equivalent I-131 of the secondary coolant once per 31 days whenever the gross activity indicates iodine concentrations greater than 10% of the allowable limit or, once per 6 months whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. However as discussed in DOC M.1 for this specification, the extended surveillance interval of 6 months for the determination of Dose Equivalent I-131 in the secondary coolant has been proposed for deletion and that future testing be performed every 31 days. Thus, the need to perform sampling of the secondary coolant for gross radioactivity is no longer necessary and has been delete in the ITS. This change is acceptable since gross radioactivity in the secondary coolant is not evaluated for radiological consequences in any of the accidents assumed in the FSAR, and the concentration of the Dose Equivalent I-131 in the secondary coolant will continue to be determined at an appropriate frequency. In addition, radiation monitoring instrumentation, controlled in accordance with the Offsite Dose Calculation Manual (e.g., SG blowdown monitors and condenser off gas monitor), is available to monitor increases in the radioactivity levels in the secondary coolant. This change is consistent with NUREG-1432.

1. **Does the change involve a significant increase in the probability or consequence of an accident previously evaluated?**

   Analyzed events are assumed to be initiated by the failure of plant structures, systems, or components. The proposed change deletes the sample requirement for gross radioactivity of the secondary coolant. This sample does not have a detrimental impact on the integrity of any plant structure, system, or component. Deletion of this sample requirement will not alter the operation of any plant equipment, or otherwise increase its failure probability. As such, the probability of occurrence for a previously analyzed accident is not significantly increased.

The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event. Gross radioactivity of the secondary coolant is not an initial condition input assumed for any analyzed event. The amount of Dose Equivalent I-131 in the secondary coolant is the assumed parameter. The limit requirement for Dose Equivalent I-131 remains unchanged and the sampling requirement has become more restrictive (see M.1). The deletion of the gross radioactivity sampling requirement does not affect the assumptions of an analyzed event. This change does not affect the performance of any credited equipment since the sample requirement is for an unassumed parameter. As a result, no analysis assumptions are violated. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not involve a physical alteration of the plant. No new equipment is being introduced, and no installed equipment is being operated in a new or different manner. There is no alteration to the parameters within which the plant is normally operated or in the set points which initiate protective or mitigative actions. No change is being proposed to the procedures governing normal plant operation or those procedures relied upon to mitigate a design basis event. Deleting the sample requirement for gross radioactivity of the secondary coolant does not have a detrimental impact on the manner in which plant equipment operates or responds to an actuation signal. As such, no new failure modes are being introduced. In addition, the change does not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety is established through equipment design, operating parameters, and the set points at which automatic actions are initiated. Deleting the requirement to sample the secondary coolant for gross radioactivity does not significantly impact these factors. There are no design changes or equipment performance parameter changes associated with this change. Therefore, this change does not involve a significant reduction in the margin of safety.

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The UHS shall be OPERABLE.

New ↓

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

	CONDITION	REQUIRED ACTION		COMPLETION TIME
6	~~A. One or more cooling towers with one cooling tower fan inoperable.~~	~~A.1~~	~~Restore cooling tower fan(s) to OPERABLE status.~~	~~7 days~~
6 3 1	~~B. Required Action and associated Completion Time of Condition A not met.~~ ~~OR~~ A, UHS inoperable ~~[for reasons other than Condition A]~~.	~~B~~ A.1 5	Be in MODE 3.	6 hours
		AND		
		~~B~~ A.2 3	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
2	SR 3.7.9.1	Verify water level of UHS is ≥ ~~[562]~~ 568.25 ft above ~~[~~mean sea level~~]~~.	24 hours	1

(continued)

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3.7 PLANT SYSTEMS

4 3.7.13 ~~Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)~~ Engineered Safeguards Room Ventilation (ESRV) Dampers

New

LCO 3.7.13 Two ECCS PREACS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ECCS PREACS trains inoperable.	A.1 ~~Restore ECCS PREACS train to OPERABLE status.~~ Initiate action to isolate associated ESRV Damper train.	~~7 days~~ Immediately
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 5.~~	~~6 hours~~ ~~36 hours~~

New 2 6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
~~SR 3.7.13.1 Operate each ECCS PREACS train for [≥ 10 continuous hours with the heater operating or (for systems without heaters) ≥ 15 minutes].~~	~~31 days~~

6

(continued)

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④ FBAC = Fuel Handling Area Ventilation System

3.7 PLANT SYSTEMS

③ 3.7.~~14~~ 12 Fuel ~~Building Air Cleanup System (FBACS)~~ Handling Area Ventilation System

CTS 3.8.4 ⑤ LCO 3.7.~~14~~ 12 INSERT 1 ~~Two FBACS trains shall be OPERABLE.~~

⑥ INSERT 2 APPLICABILITY: ~~[MODES 1, 2, 3, and 4,] During movement of irradiated fuel assemblies in the fuel building.~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One FBACS train inoperable.	A.1 Restore FBACS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4. OR Two FBACS trains inoperable in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and Associated Completion Time [of Condition A] not met during movement of irradiated fuel assemblies in the fuel building.	C.1 Place OPERABLE FBACS train in operation. OR C.2 Suspend movement of irradiated fuel assemblies in the fuel building.	Immediately Immediately

CTS 3.8.4

INSERT 3 ⑦

(continued)

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SECTION 3.7

INSERT 1

The Fuel Handling Area Ventilation System shall be OPERABLE with one fuel handling area exhaust fan, aligned to the emergency filter bank, in operation.

INSERT 2

During movement of irradiated fuel assemblies in the fuel handling building when irradiated fuel assemblies with < 30 days decay time are in the fuel handling building,

During movement of a fuel cask in or over the SFP when irradiated fuel assemblies with < 90 days decay time are in the fuel handling building,

During CORE ALTERATIONS when irradiated fuel assemblies with < 30 days decay time are in the containment with the equipment hatch open,

During movement of irradiated fuel assemblies in the containment when irradiated fuel assemblies with < 30 days decay time are in the containment with the equipment hatch open.

INSERT 3

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel Handling Area Ventilation System not aligned or in operation. OR Fuel Handling Area Ventilation System inoperable.	A.1 Suspend movement of fuel assemblies. AND A.2 Suspend CORE ALTERATIONS. AND A.3 Suspend movement of a fuel cask in or over the SFP.	Immediately Immediately Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
7 D. Two FBACS trains inoperable during movement of irradiated fuel assemblies in the fuel building.	D.1 Suspend movement of irradiated fuel assemblies in the fuel building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
8 SR 3.7.14.1	Operate each FBACS train for [≥ 10 continuous hours with the heaters operating or (for systems without heaters) ≥ 15 minutes].	31 days
TS 4.2 TBL 4 2.3 item 1 — 2 SR 3.7.14.~~2~~ 1	Perform required FBACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].	In accordance with the ~~[VFTP]~~ Ventilation Filter Testing Program
9 [SR 3.7.14.3	Verify each FBACS train actuates on an actual or simulated actuation signal.	[18] months]
10 New SR 3.7.14.~~4~~ 2	Verify ~~one FBACS train can maintain a negative pressure ≥ [ ] inches water gauge with respect to atmospheric pressure, during the [post accident] mode of operation at a flow rate ≤ [3000] cfm.~~ the flow rate of the, when aligned to the emergency filter bank, is ≥ 5840 cfm and ≤ 8760 cfm.	[18] months ~~on a STAGGERED TEST BASIS~~

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.7.14.5~~	~~Verify each FBACS filter bypass damper can be opened.~~	~~[18] months~~

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3.7.18 = 3.7.16 4

3.7 PLANT SYSTEMS

3.7.~~18~~ 16 Spent Fuel Assembly Storage

CTS 5.4.2.1 2 3 4

LCO 3.7.~~18~~ 16 The combination of initial enrichment and burnup of each spent fuel assembly stored in [Region ~~2~~] II shall be within the ~~acceptable [burnup domain]~~ requirements of ~~Figure~~ Table 3.7.~~18~~ 16-1 ~~[or in accordance with Specification 4.3.1.1]~~.

New

APPLICABILITY: Whenever any fuel assembly is stored in [Region ~~2~~] II of either the spent fuel storage pool or the north tilt pit. 1 2 4

ACTIONS 2

New

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met. 1 2	A.1 --------NOTE-------- LCO 3.0.3 is not applicable. --------------------- Initiate action to move the noncomplying fuel assembly from [Region ~~2~~] II.	Immediately

SURVEILLANCE REQUIREMENTS

CTS 5.4.2d

SURVEILLANCE	FREQUENCY
SR 3.7.~~18~~ 16.1 Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with ~~Figure~~ Table 3.7.~~18~~ 16-1 ~~or Specification 4.3.1.1~~. 3 11 7 8 1	Prior to storing the fuel assembly in [Region ~~2~~] II.

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B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

(4) (DOWNSTREAM OF THE MSIV)

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). (2) MSIV closure terminates flow from the unaffected (intact) steam generator. (FOR BREAKS UPSTREAM OF THE OTHER MSIV) (4)

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the (2) main steam safety valves (MSSVs), atmospheric dump valves, and auxiliary feedwater pump turbine steam supplies to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam (TURBINE) (4) (VALVE) generator from the other, and isolates the turbine, [redacted] Bypass [redacted], and other auxiliary steam supplies from the (8) <INSERT 1> steam generators. (2)

The MSIVs close on [redacted] isolation signal generated (5) (4) (4) (ISOLATION SIGNAL) by either low steam generator pressure or high containment pressure. The MSIVs fail closed on loss of ~~control or~~ (AIR) ~~actuation power~~. The ~~MSIS~~ also actuates the main feedwater (2) (11) (REGULATING) ~~isolation~~ valves (MFIVs) to close. The MSIVs may also be actuated manually. (R) (AND MFRV BYPASS VALVES) (11)

(1) A description of the MSIVs is found in the FSAR, Section 10.3 (Ref. 1). (2)

(7) (MAIN)

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the (M) (14.18) containment analysis for the ~~large~~ steam line break (SLB) (2) (1) inside containment, as discussed in the FSAR, Section ~~6.2~~ (7) (M) (Ref. 2). It is also influenced by the accident analysis of the SLB events presented in the FSAR, Section ~~15.1.5~~ (14.14) (2) (Ref. 3). ~~The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).~~ (6)

(7) <INSERT 2> (M) (7) (2) (4) (FOR CONTAINMENT PRESSURE

The limiting case for the containment analysis is the hot zero power SLB inside containment ~~with a loss of offsite power following turbine trip, and failure of the MSIV on the affected steam generator to close.~~ (4) At zero power, the steam generator inventory and temperature are at their maximum, (and fuel integrity)

(continued)

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SECTION 3.7

INSERT 1

.....assuming the normally closed MSIV bypass valves are closed. The MSIV bypass valves do not receive an isolation signal and might be open during zero power conditions.

INSERT 2

The MSIVs are swing disc check valves. The inherent characteristic of this type of valve allows for reverse flow through the valve on a differential pressure even if the valve is closed. In the event of an MSLB, if the MSIV associated with the unaffected steam generator fails to close, both steam generators may blowdown. This failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed. The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a double steam generator blowdown event, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modifications would not provide a cost beneficial improvement to plant safety.

There are three different limiting MSLB cases that have been evaluated, one for fuel integrity and two for containment analysis (one for containment temperature and one for containment pressure). The limiting case for containment temperature is the hot full power MSLB inside containment following a turbine trip. At hot full power, the stored energy in the primary coolant is maximized.

BASES

DUE TO THE OPEN MSIV BYPASS VALVES 4

APPLICABLE SAFETY ANALYSES (continued)

INSERT 1

maximizing the analyzed mass and energy release to the containment. ~~Due to~~ reverse flow ~~, failure of the MSIV to close~~ contributes to the total release of the additional mass and energy ~~in the steam headers, which are downstream of the other MSIV. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the borated water injection delivered by the Emergency Core Cooling System. Other failures considered are the failure of an MFIV to close, and failure of an emergency diesel generator to start.~~ 9

The accident analysis compares several different SLB events against different acceptance criteria. The ~~large~~ SLB (M) 7 outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The ~~large~~ SLB inside containment at hot ~~zero~~ FULL power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip.

With offsite power available, the ~~reactor~~ PRIMARY 4 coolant pumps continue to circulate coolant through the steam generators, maximizing the ~~Reactor~~ Coolant System (RCS) cooldown. (P) With a loss of offsite power, the response of mitigating systems, such as the ~~High Pressure Safety Injection~~ (HPSI) pumps, is delayed. 2 ~~Significant single failures considered include: failure of a MSIV to close, failure of an emergency diesel generator, and failure of a HPSI pump.~~ 6

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. An ~~HELB~~ MSLB inside containment. ~~In order to maximize the mass and energy release into the containment, the analysis assumes that the MSIV in the affected steam generator remains open.~~ 4 For this accident scenario, steam is discharged into containment from both steam generators until closure of the MSIV in the intact steam generator occurs. After MSIV closure, steam is discharged into containment only from the affected steam generator ~~and from the residual steam in the~~ 4

2

(continued)

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SECTION 3.7

INSERT 1

With the most reactive control rod assumed stuck in the fully withdrawn position, there is an increased possibility that the core will return to power. The core is ultimately shut down by a combination of doppler feedback, steam generator dry out, and borated water injection delivered by the Emergency Core Cooling System.

SECTION 3.7

INSERT 1

.....and MFRV bypass valves in conjunction with feed pump speed, control Main Feedwater (MFW) flow to the steam generators for level control during normal plant operation. The valves also

INSERT 2

The MFRVs and MFRV Bypass valves are non-safety grade valves located on non-safety grade piping that fails "as-is" on a loss of air. If required, MFW isolation can be accomplished using manually operated valves located upstream or downstream of the MFRVs and MFRV Bypass valves. In addition, each MFRV is equipped with a handwheel that can be used to isolate this MFW flowpath.

BASES

AND MFRV BYPASS VALVES

BACKGROUND (continued)

A description of the MFIVs is found in the FSAR, Section ~~[10.4.7]~~ (Ref. 1).

1 2.3

MSLB containment response analysis

an assumption in

Closure

8 2 MITIGATE

APPLICABLE SAFETY ANALYSES

~~The design basis~~ of the MFIVs is ~~established by~~ the analysis for the ~~large~~ SLB. ~~It is also influenced by the accident analysis for the large FWLB.~~ Closure of the MFIVs and ~~their~~ bypass valves ~~may~~ also ~~be~~ relied on to ~~terminate~~ a ~~steam break for core response analysis~~ ~~and an excess feedwater flow event upon receipt of a MSIS on high steam generator level.~~

MFRV

assumed in the MSLB core response (DNB) analysis

6

OR MFRV 2

Failure of an MFIV ~~and the~~ bypass valve to close following an SLB ~~, FWLB, or excess feedwater flow event~~ can result in additional mass and energy to the steam generators contributing to cooldown. This failure also results in additional mass and energy releases following an SLB ~~or FWLB~~ event.

8 M 4

INSERT

M 8

The MFIVs satisfy Criterion 3 of the ~~NRC Policy Statement~~.

2 AND MFRV BYPASS VALVES 10 CFR 50.36(c)(2)

BOTH BOTH MFRV

LCO

This LCO ensures that ~~the~~ MFIVs and ~~the~~ bypass valves will isolate MFW flow to the steam generators, following an ~~FWLB or~~ SLB. ~~these valves will also isolate the nonsafety related portions from the safety related portions of the system.~~ This LCO requires that ~~[two]~~ MFIVs and ~~[MFIV]~~ bypass valves ~~in each feedwater line~~ be OPERABLE. The MFIVs and ~~the~~ bypass valves are considered OPERABLE when the isolation times are within limits, and are closed on an isolation ~~actuation~~ signal.

2 8 M 10 BOTH 1 MFRV 4 10 BOTH MFRV

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB ~~or FWLB~~ inside containment. ~~If an MSIS on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.~~

8 M 6

MFRV EITHER

APPLICABILITY

The MFIVs and ~~the~~ bypass valves must be OPERABLE whenever there is significant mass and energy in the ~~Reactor~~ Coolant

4 PRIMARY

(continued)

2

, CLOSED AND DEACTIVATED, OR ISOLATED BY CLOSED MANUALLY ACTUATED VALVES,

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SECTION 3.7

INSERT

However, this failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed. The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a single steam generator blowdown event with continued feedwater, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modifications would not provide a cost beneficial improvement to plant safety.

BASES

APPLICABILITY (continued)

System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator.

10 BOTH MFRV AND BOTH MFRV BYPASS VALVES ARE EITHER,

In MODES 1, 2, and 3, the MFIV [or [MFIV] bypass valves] are required to be OPERABLE, except when they are closed and deactivated or isolated by a closed manual valve in order to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and deactivated or isolated by a closed manual valve, they are already performing their safety function.

EITHER

1Y ACTUATED

2 INSERT

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs and the bypass valves are normally closed since MFW is not required. TO BE OPERABLE

MFRV

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one MFIV or the bypass valve inoperable, action must be taken to close or isolate the inoperable valve(s) within [8 or 72] hours. When these valve(s) are closed or isolated, they are performing their required safety function (e.g., to isolate the line).

MFRV

~~For units with only one MFIV per feedwater line:~~ The [8] hour Completion Time is reasonable to close the MFIV or its bypass valve, which includes performing a controlled unit shutdown to MODE 2.

MFRV

which includes performing a controlled plant shutdown to a condition that supports isolation of the affected valve

~~The [72] hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves, and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.~~

~~B.1~~

~~If more than one MFIV or [MFIV] bypass valve in the same flow path cannot be restored to OPERABLE status, then there~~

(continued)

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BASES

response and core response (DNB)

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 (continued)

AND MFRV BYPASS VALVES (2)

S ARE bounding Values

actuation signal. The MFIV closure time ~~is~~ assumed in the ~~accident~~ and containment analyses. This ~~Surveillance~~ is normally performed upon returning the ~~unit~~ to operation following a refueling outage. The MFIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the ~~unit~~ generating power. As these valves are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 2) requirements during operation in MODES 1 and 2.

MSLB

SR

PLANT (4)

(Refs. 3 and 4)

(2) STROKE

(1) (2)

The Frequency is ~~in accordance with the [Inservice Testing Program or]~~ [18] months]. The [18] month Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the SR when performed at the [18] month Frequency.

REFERENCES

1. FSAR, Section [~~10.4.7~~]. 10.2.3 (1)
2. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400.
3. FSAR, Section 14.8.2 (2)
4. FSAR, Section 14.14

(2) SPECIFIC SIGNALS (E.G., STEAM GENERATOR LOW PRESSURE AND CONTAINMENT HIGH PRESSURE) ARE TESTED UNDER SECTION 3.3, "INSTRUMENTATION."

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BASES

LCO (continued) 8

~~Systems Branch 5-1 (Ref. 4) and exceeds the volume required by the accident analysis.~~

OPERABILITY of the CST CONDENSATE STORAGE AND SUPPLY is determined by maintaining the COMBINED tank level 5 at or above the minimum required level VOLUME. 5

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator 1 is being relied upon for heat removal, the CST CONDENSATE STORAGE AND SUPPLY 4 is required to be OPERABLE.

In MODES 5 and 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2 CONDENSATE VOLUME 5

If the CST level is not within the limit, the OPERABILITY of the backup water supply IES 6 must be verified by administrative means within 4 hours AND ONCE EVERY 12 HOURS THEREAFTER.

OPERABILITY of the backup feedwater supply must include verification of the OPERABILITY of flow paths from the CONDENSATE 5 backup supply Fire Water System and SWS to the AFW pumps, and availability of the required volume of water in the backup supply. The CST level 5 VOLUME must be returned to OPERABLE status within 7 days, as the backup supply 6 IES may be performing this function in addition to its 2 THEIR normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. 9 <INSERT> The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply 6 IES being available, and the low probability of an event requiring the use of the water from the CST AND T-81 5 occurring during this period.

B.1 and B.2 5 CONDENSATE VOLUME OR THE BACKUP SUPPLY

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the unit 4 PLANT must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generator 1 for heat removal, within [18] 30 hours. The allowed Completion

(continued)

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SECTION 3.7

INSERT

The CCW System consists of three pumps connected in parallel to common suction and discharge headers. The discharge header splits into two parallel heat exchangers and then combines again into a common distribution header which supplies various heat loads. A common surge tank provides the necessary net positive suction head for the CCW pumps and a surge volume for the system. A train of CCW is considered to be that equipment electrically connected to a common safety bus necessary to transfer heat acquired from the various heat loads to the Service Water System (SWS). There are two CCW trains, each associated with a Safeguards Electrical Distribution Train which are described in Specification 3.8.9, "Distribution Systems - Operating." The CCW train associated with the Left Safeguards Electrical Distribution Train consists of two CCW pumps (P-52A, P-52C), both CCW heat exchangers (E-54A, E-54B), the CCW surge tank (T-3), associated piping, valves, and controls for the equipment to perform their safety function. The CCW train associated with the Right Safeguards Electrical Distribution Train consists of one CCW pump (P-52B), both CCW heat exchangers (E-54A, E-54B), the CCW surge tank (T-3), associated piping, valves, and controls for the equipment to perform their safety function. The pumps and valves are automatically started upon receipt of a safety injection actuation signal and all essential valves are aligned to their post accident positions. CCW valve repositioning also occurs following receipt a recirculating actuation signal (RAS) which aligns associated valves to provide full cooling to the component cooling water heat exchangers during the recirculation phase following a design basis Loss of Coolant Accident (LOCA).

SECTION 3.7

INSERT 1

Specific signals (e.g., safety injection) are tested under Section 3.3, "Instrumentation." If the isolation valve for the noncritical service water header (CV-1359) or for the containment air cooler VHX-4 (CV-0869) fail to close, then both trains of SWS are considered inoperable due to the diversion of cooling water flow.

INSERT 2

.....in the "with standby power available" mode which tests the starting of the pumps by the SIS-X relays. The starting of the pumps by the sequencer is performed in Section 3.8, "Electrical Power Systems."

INSERT 3

This SR is modified by a Note which states this SR is not required to be met in MODE 4. The instrumentation providing the input signal is not required in MODE 4, therefore to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE.

BASES

BACKGROUND (continued)

~~are illustrative of a cooling tower UHS without a makeup requirement.~~ 6

Additional information on the design and operation of the system along with a list of components served can be found in ~~Reference 1~~. FSAR, SECTION 9.1 (REF. 1) 2

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the ~~unit~~ PLANT 4 is cooled down and placed on shutdown cooling. ~~For those units using it as the normal heat sink for condenser cooling via the Circulating Water System, unit operation at full power is its maximum heat load. Its~~ ~~maximum~~ post accident heat load occurs ~~20~~ BETWEEN 20 TO 40 4 minutes after a design basis loss of coolant accident (LOCA). Near this time, the ~~unit~~ PLANT 4 switches from injection to recirculation, and the containment cooling systems are required to remove the core decay heat.

~~The operating limits are~~ INSERT 4 WHICH FORMS THE BASIS FOR THE OPERATING LIMITS based on 4 conservative heat transfer analyses for the worst case LOCA. ~~Reference 1~~ FSAR, SECTION 14.18 (REF. 2) AND DESIGN BASIS DOCUMENT (DBD), 1.02 (REF. 3) provides the details of ~~the assumptions used in~~ the analysis~~:~~. The assumptions include: ~~worst expected meteorological conditions,~~ conservative uncertainties when calculating decay heat, and the worst case SINGLE ACTIVE failure ~~(e.g., single failure of a manmade structure). The UHS is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water in the UHS~~. 7

The UHS satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10CFR 50.36(c)(2) 2.

LCO

The UHS is required to be OPERABLE. The UHS is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate ~~for at least 30 days following the design basis LOCA~~ 7 without the loss of ~~net positive suction head~~ (NPSH)~~,~~ and without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the UHS temperature should not exceed ~~[90]~~ 81.5 1 °F and the level should not fall below ~~[562 ft]~~ 568.25 ft. ABOVE 2 mean sea level~~]~~ during normal ~~unit~~ PLANT 4 operation.

(continued)

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INSERT

The minimum water level of the UHS is based on the NPSH requirements for the SWS pumps. The NPSH calculation assumes a minimum water level of 4 feet above the bottom of the pump suction bell which corresponds to an elevation of 557.25 ft. Violation of the SWS pump submergence requirement should never become a factor unless the Lake Michigan water level falls below the top of the sluice gate opening which is at elevation 568.25 ft. Early warning of a falling intake water level is provided by the intake structure level alarm. The nominal lake level is approximately 580 ft mean sea level. The minimum water temperature of the UHS is ...

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1 (continued)

1

trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ ~~[562]~~ 568.25 ft ~~[mean sea level]~~ ABOVE MSL AS MEASURED WITHIN THE BOUNDARIES OF THE INTAKE STRUCTURE. 4

SR 3.7.9.2

This SR verifies that the SWS is available to ~~cool the CCW System to at least its maximum design temperature within~~ PROVIDE ADEQUATE COOLING FOR the maximum accident or normal design heat loads ~~for 30 days~~ 7 following a DBA. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the ~~UHS~~ water temperature FROM THE UHS is ≤ ~~[92]~~ 81.5 °F. 7 1

6

~~SR 3.7.9.3~~

~~Operating each cooling tower fan for ≥ [15] minutes verifies that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the UHS cooling tower fans occurring between surveillances.~~

REFERENCES

1. FSAR, Section ~~[~~9.~~[2.5]~~ 1 . 1

~~2. Regulatory Guide 1.27.~~ 4

2. FSAR, SECTION 14.18

3. DBD, 1.02

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SECTION 3.7

INSERT 1

.....isolate the safeguards rooms by closing the inlet and exhaust plenum dampers on the initiation of a high radiation alarm from their respective airborne particulate monitor. This isolation lowers the offsite dose to well within 10 CFR 100 (Ref. 1) limits if a leak should occur. Typically, high radiation would only be expected due to excessive leakage during the recirculation phase of operation following a loss of coolant accident (LOCA).

INSERT 2

.....supply plenum damper, an exhaust plenum damper, a radiation monitor, and associated piping, valves, and duct work.

INSERT 3

.....which is addressed in LCO 3.3.10, "Engineered Safeguards Room Ventilation (ESRV) Instrumentation"

INSERT 4

.....shut, isolating the affected safeguards room(s) from the rest of the auxiliary building ventilation system lowering the leakage to the environment from the auxiliary building.

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BASES

LCO (continued)

~~ECCS PREACS is considered OPERABLE when the individual components necessary to maintain the ECCS Pump Room filtration are OPERABLE in both trains.~~

(4) An ECCS PREACS train is considered OPERABLE when its associated ~~:~~ RADIATION MONITOR, INSTRUMENTATION, ~~DUCTWORK~~, VALVES, AND DAMPERS ARE OPERABLE

(6)
~~a. Fan is OPERABLE;~~

~~b. HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions; and~~

~~c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.~~

(2) APPLICABILITY

In MODES 1, 2, 3, and 4, the ECCS PREACS ~~is~~ TRAINS ARE required to be OPERABLE consistent with the OPERABILITY requirements of the EMERGENCY CORE COOLING SYSTEM ~~(ECCS)~~.

In MODES 5 and 6, the ECCS PREACS ~~is~~ TRAINS ARE not required to be OPERABLE, since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

<INSERT 1>

(6)
~~With one ECCS PREACS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ECCS PREACS function.~~

~~The 7 day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72 hour Completion Time) and this system is not a direct support system for the ECCS. The 7 day Completion Time is reasonable, based on the low probability of a DBA occurring during this time period, and the consideration that the remaining train can provide the required capability.~~

(continued)

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SECTION 3.7

INSERT 1

Condition A addresses the failure of one or both ESRV Damper train(s). Operation may continue as long as action is immediately initiated to isolate the affected engineered safeguards room. With the inlet and exhaust dampers closed or if the inlet and outlet ventilation plenums are adequately sealed, the engineered safeguards room is isolated and the intended safety function is achieved, since the potential pathway for radioactivity to escape to the environment from the engineered safeguards room has been minimized.

The Completion Time for this Required Action is commensurate with the importance of maintaining the engineered safeguards room atmosphere isolated from the outside environment when the ECCS pumps are circulating primary coolant after an accident.

4 FBAC = Fuel Handling Area Ventilation System

3 B 3.7.14 = B 3.7.12

B 3.7 PLANT SYSTEMS

B 3.7.~~14~~ 12 Fuel ~~Building Air Cleanup~~ Handling Area Ventilation System (FBACS)

BASES

BACKGROUND (4) The FBACS filters airborne radioactive particulates from the area of the spent fuel pool following a fuel handling accident or a fuel cask drop accident ~~loss of coolant accident. The FBACS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the fuel pool area.~~

(11)

~~The FBACS consists of two independent, redundant trains. Each train consists of a heater, a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters, functioning to reduce the relative humidity of the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case of failure of the main HEPA filter bank. The downstream HEPA filter is not credited in the analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the fuel handling building following receipt of a high radiation signal.~~

INSERT 1 →

~~The FBACS is a standby system, part of which may also be operated during normal unit operations. Upon receipt of the actuating signal, normal air discharges from the fuel handling building, the fuel handling building is isolated, and the stream of ventilation air discharges through the system filter trains. The prefilters or demisters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.~~

(1) (2) The FBACS is 14.19 discussed in the FSAR, Sections ~~6.5.1~~ 9.8, 14.11, ~~9.4.5~~ and ~~15.7.4~~ (Refs. 1, 2, and 3 ~~respectively~~), because it may be used for normal, as well as post accident, atmospheric cleanup functions.

(continued)

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SECTION 3.7

INSERT 1

The fuel handling area is served by two separate subsystems one being part of the original plant design, and the other being added as part of the Auxiliary Building Addition.

The original plant design consists of a supply plenum and an exhaust plenum including associated ductwork, dampers, and instrumentation. The supply plenum contains one prefilter, two heating coils, and one supply fan. The exhaust plenum contains two filter banks (normal and emergency) configured in a parallel flow arrangement, and two independent exhaust fans which draw air from a common duct. The "normal filter bank" contains a prefilter and a High Efficiency Particulate Air (HEPA) filter. The "emergency filter bank" contains a prefilter, HEPA filter, and an activated charcoal filter.

The Auxiliary Building Addition, which was added to serve the spaces at the north end of the spent fuel pool, also consist of a supply plenum and exhaust plenum. The supply plenum is configured similar to the supply plenum provided in the original plant design and includes one prefilter, two heating coils, and one supply fan. The exhaust plenum is different from the original plant design in that it only contains one filter bank consisting of a prefilter and HEPA filter, and two common exhaust fans.

During normal plant operations, the Fuel Handling Area Ventilation System supplies filtered and heated (as needed) outside air to the fuel handling area. The exhaust fans draw air from the fuel handling area through the normally aligned prefilters and HEPA filters and discharge it to the unit stack by way of the main ventilation exhaust plenum.

During plant evolutions when the possibility for a fuel handling accident exists, the Fuel Handling Area Ventilation System is configured such that all fans are stopped except one exhaust fan in the original plant subsystem aligned to the "emergency filter bank." The "normal filter bank" in the original plant design is isolated by closing its associated inlet damper. Thus, in the event of a fuel handling accident, the fuel handling area atmosphere will be filtered for the removal of airborne fission products prior to being discharged to the outside environment.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The FBACS is designed to mitigate the consequences of a fuel handling accident in which [all] rods in the fuel assembly are assumed to be damaged. The analysis of the fuel handling accident is given in Reference 3. The Design Basis Accident analysis of the fuel handling accident assumes that only one train of the FBACS is functional, due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from the fuel handling building is determined for a fuel handling accident. These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 4).

INSERT 1

12

The FBACS satisfies Criterion 3 of the NRC Policy Statement.

LCO

INSERT 2

5

Two independent and redundant trains of the FBACS are required to be OPERABLE to ensure that at least one is available, assuming a single failure that disables the other train coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the fuel building exceeding the 10 CFR 100 limits (Ref. 5) in the event of a fuel handling accident.

2

The FBACS is considered OPERABLE when the individual components necessary to control exposure in the fuel handling building are OPERABLE in both trains. An FBACS train is considered OPERABLE when its associated: The

4

a. one exhaust Fan is OPERABLE; INSERT 3

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

5

c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

INSERT 4

APPLICABILITY

INSERT 5

In MODES 1, 2, 3, and 4, the FBACS is required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a LOCA (refer to LCO 3.7.13, "Emergency

6

(continued)

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SECTION 3.7

INSERT 1

...or fuel cask drop accident by limiting the amount of airborne radioactive material discharged to the outside atmosphere.

The results and major assumptions used in the analysis of the fuel handling accident are presented in FSAR Section 14.19. For the purpose of defining the upper limit of the radiological consequences of a fuel handling accident, it is assumed that a fuel bundle is dropped during fuel handling activities and all the fuel rods in the equivalent of an entire assembly (216) fail. The bounding fuel handling accident is assumed to occur in containment two days after shutdown. No containment isolation is assumed to occur. As such, the released fission products escape to the environment with no credit for filtration. The results of this analysis have shown that the offsite doses resulting from this event are within the guideline of 10 CFR 100. In the event a fuel handling accident were to occur in the fuel handling area, the radioactive release would pass through the "emergency filter bank" significantly reducing the amount of radioactive material released to the environment. Thus, the consequences of a fuel handling accident in the fuel handling area are deemed acceptable with or without the "emergency filter bank" in operation since they are no more severe than the consequences of a fuel handling accident in containment.

The results and major assumptions used in the analysis of the fuel cask drop accident are presented in FSAR Section 14.11. For the purpose of defining the upper limit of the radiological consequences of a fuel cask drop accident, it is assumed that all 73 fuel assemblies in a 7 x 11 Westinghouse spent fuel pool rack with a minimum decay of 30 days are damaged and release their fuel rod gap inventories. Three fuel cask drop scenarios were analyzed to encompass all fuel cask drop events. They are:

1. A fuel cask drop onto 30 day decayed fuel with the Fuel Handling Area Ventilation System aligned for emergency filtration with a conservative amount of unfiltered leakage. All isolatable unfiltered leak path are assumed to be isolated prior to event initiation.

2. A fuel cask drop onto 30 day decayed fuel with the Fuel Handling Area Ventilation System aligned for emergency filtration with a conservative amount of unfiltered leakage. This scenario determined the maximum amount of non-isolatable unfiltered leakage than can exist and still meet offsite dose limits. This scenario also assumes isolation of isolable leak paths prior to event initiation.

3. A fuel cask drop onto 90 day decayed fuel without the Fuel Handling Area Ventilation System aligned for emergency filtration. This scenario needs no assumptions as to unfiltered leakage or post-accident unfiltered leak path isolation times since all radiation is assumed to be released unfiltered from the fuel handling area.

SECTION 3.7

INSERT 1 Con't

The results of the analysis show that the radiological consequences of a fuel cask drop in the spent fuel pool meet the acceptance criteria of Regulatory Guide 1.25 (Ref. 4) and NUREG-0800 Section 15.7.5 (Ref. 5) for all scenarios. In addition, the dose from all scenarios are less than 25% of the dose guidelines in 10 CFR 100. For scenario 2, the analysis shows that a maximum of 20% charcoal filter bypass from non-isolatable leak paths can be accommodated while still meeting 25% of the 10 CFR 100 guidelines.

Filtration of the fuel handling area atmosphere following a fuel handling accident is not necessary to maintain the offsite doses within the guidelines of 10 CFR 100. Thus, a total system failure would not impact the margin of safety as described in the safety analysis. However, analysis has shown that post accident filtration by the Fuel Handling Area Ventilation System provides significant reduction in offsite doses by limiting the release of airborne radioactivity. Therefore, for the fuel handling accident, the Fuel Handling Area Ventilation System satisfies Criterion 4 of 10 CFR 50.36(c)(2).

Filtration of the fuel handling area atmosphere following a fuel cask drop on irradiated fuel assemblies with < 90 days decay is required to maintain the offsite doses within the guidelines of 10 CFR 100. Therefore, for the fuel cask drop accident, the Fuel Handling Area Ventilation System satisfies Criterion 3 of 10 CFR 50.36(c)(2).

INSERT 2

The LCO for the Fuel Handling Area Ventilation System ensures filtration of the fuel handling area atmosphere is immediately available in the event of a fuel handling accident, or a fuel cask drop accident. As such, the LCO requires the Fuel Handling Area Ventilation System to be OPERABLE with one fuel handling area exhaust fan, aligned to the "emergency filter bank", in operation.

INSERT 3

...aligned to the "emergency filter bank" and in operation to ensure the air discharged to the main ventilation exhaust plenum has been filtered. Operation of only one fuel handling area exhaust fan ensures the design flow rate of the "emergency filter bank" is not exceeded.

SECTION 3.7

INSERT 4

Inclusive to the requirement to align the "emergency filter bank" is that the "normal filter bank" is isolated by its associated inlet damper to prevent the release of unfilter air.

INSERT 5

The Fuel Handling Area Ventilation System must be Operable, aligned, and in operation whenever the potential exists for an accident that results in the release of radioactive material to the fuel handling area atmosphere that could exceed previously approved offsite dose limits if released unfiltered to the outside atmosphere. As such, the Fuel Handling Area Ventilation System is required; during movement of irradiated fuel assemblies in the fuel handling building when irradiated fuel assemblies with < 30 days decay time are in the fuel handling building; during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in containment when irradiate fuel assemblies with < 30 days decay time are in the containment with the equipment hatch open, and during movement of a fuel cask in or over the spent fuel pool when irradiated fuel assemblies with < 90 days are in the fuel handling building.

The requirement for the Fuel Handling Area Ventilation System does not apply during movement of irradiated fuel assemblies or CORE ALTERATIONS when all irradiated fuel assemblies in the fuel handling building, or all irradiated fuel assemblies in the containment with the equipment hatch open, have decayed for 30 days or greater since the dose consequences from a fuel handling accident would be of the same magnitude without the filters operating as the dose consequences would be with the filters operating and two days decay. In addition, the requirement for the Fuel Handling Area Ventilation System does not apply during fuel cask movement when all irradiated fuel assemblies in the fuel handling building have decayed 90 days or greater since the dose consequences remain less than 25% of the guidelines of 10 CFR 100.

BASES

APPLICABILITY (continued)

~~Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)) for units that use this system as part of their ECCS PREACS.~~

(6)

~~During movement of irradiated fuel assemblies in the fuel building, the FBACS is required to be OPERABLE to mitigate the consequences of a fuel handling accident.~~

~~In MODES 5 and 6, the FBACS is not required to be OPERABLE, since the ECCS is not required to be OPERABLE.~~

ACTIONS

A.1 and A.2

INSERT 1

(7)

~~If one FBACS train is inoperable, action must be taken to restore OPERABLE status within 7 days. During this time period, the remaining OPERABLE train is adequate to perform the FBACS function. The 7 day Completion Time is reasonable, based on the risk from an event occurring requiring the inoperable FBACS train, and ability of the remaining FBACS train to provide the required protection.~~

~~B.1 and B.2~~

~~In MODE 1, 2, 3, or 4, when Required Action A.1 cannot be completed within the Completion Time, or when both FBACS trains are inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.~~

~~C.1 and C.2~~

~~When Required Action A.1 cannot be completed within the required Completion Time during movement of irradiated fuel in the fuel building, the OPERABLE FBACS train must be started immediately or fuel movement suspended. This action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failure will be readily detected.~~

(continued)

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SECTION 3.7

INSERT 1

If the Fuel Handling Area Ventilation System is not aligned to the "emergency filter bank", or one exhaust fan is not in operation, or the system is inoperable for any reason, action must be taken to place the unit in a condition in which the LCO does not apply. Therefore, activities involving the movement of irradiated fuel assemblies and CORE ALTERATIONS and movement of a fuel cask in or over the spent fuel pool must be suspended immediately to minimize the potential for a fuel handling accident.

The suspension of fuel movement and CORE ALTERATIONS, and fuel cask movement shall not preclude the completion of placing a fuel assembly, core component, or fuel cask in a safe position.

BASES

ACTIONS

C.1 and C.2 (continued)

If the system is not placed in operation, this action requires suspension of fuel movement, which precludes a fuel handling accident. This does not preclude the movement of fuel to a safe position.

7

D.1

When two trains of the FBACS are inoperable during movement of irradiated fuel assemblies in the fuel building, action must be taken to place the unit in a condition in which the LCO does not apply. This LCO involves immediately suspending movement of irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

8

3 SR 3.7.14.2 1

This SR verifies the performance of FBACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)]. The FBACS filter tests are in accordance with the Regulatory Guide 1.52 (Ref. 1). The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific

1 spell out

as described in the Ventilation Filter Testing Program

(continued)

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BASES

③

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1 (continued)

operations). Specific test frequencies and additional information are discussed in detail in the [VFTP]. Spell-out

(9)

~~SR 3.7.14.3~~

~~This SR verifies that each FBACS train starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with that specified in Reference 6.~~

③

SR 3.7.14.2

INSERT 1 → This SR verifies the integrity of the fuel building enclosure. The ability of the fuel building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FBACS. During the post accident mode of operation, the FBACS is designed to maintain a slight negative pressure in the fuel building, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The FBACS is designed to maintain this negative pressure at a flow rate of ≤ [3000] cfm to the fuel building. The Frequency of [18] months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 7).

(10) INSERT 2

INSERT 3

≥ 5840 cfm and ≤ 8760

This test is conducted with the tests for filter penetration; thus, an [18] month Frequency, on a STAGGERED TEST BASIS is consistent with and other filtration SRs.

①

Performance

(9)

~~SR 3.7.14.5~~

~~Operating the FBACS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the FBACS filter bypass damper is verified if it can be closed. The 18 month Frequency is consistent with that specified in Reference 6.~~

(continued)

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SECTION 3.7

INSERT 1

Fuel Handling Area Ventilation System has not degraded and is operating as assumed in the safety analysis. The flow rate

INSERT 2

When aligned to the "emergency filter bank", the Fuel Handling Area Ventilation System is designed to reduce the amount of unfiltered leakage from the fuel handling building which, in the event of a fuel handling accident, lowers the dose at the site boundary to well within the guidelines of 10 CFR 100.

INSERT 3

.....lower the dose to these levels .....

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BASES (continued)

REFERENCES

4

1. FSAR, Section ~~6.7.1~~ 9.8.
2. FSAR, Section ~~9.4.5~~ 14.11.
3. FSAR, Section ~~15.7.4~~ 14.19.
4. Regulatory Guide 1.25, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurize Reactors.
5. 10 CFR 100.
6 ~~6~~. Regulatory Guide 1.52, Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and ADsorption Units of Light-Water Cooled Nuclear Power Plants.
5 ~~7~~. NUREG-0800, Section 15.7.5, Spent Fuel Cask Drop Accidents, ~~6.5.1, July 1981.~~

Revised
03/15/99

6 3

B 3.7 PLANT SYSTEMS

B 3.7.18 Spent Fuel Assembly Storage

6 3

BASES

BACKGROUND

used 4

The spent fuel storage facility is designed to ~~store~~ either new (nonirradiated) nuclear fuel assemblies, or ~~burned~~ (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store ~~[735]~~ 892 irradiated fuel assemblies, which includes storage for [15] failed fuel ~~containers~~ canisters. ~~The spent fuel storage cells are installed in parallel rows with center to center spacing of [12 31/32] inches in one direction, and [13 3/16] inches in the other orthogonal direction. This spacing and "flux trap" construction, whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans, is~~ <INSERT> sufficient to maintain a k_{eff} of ≤ 0.95 for spent fuel of original enrichment of up to ~~[1.3]~~ 4.40%. ~~However, as higher initial enrichment fuel assemblies are stored in the spent fuel pool, they must be stored in a checkerboard pattern taking into account fuel burnup to maintain a k_{eff} of 0.95 or less.~~

1 4 4 1 6 5

APPLICABLE SAFETY ANALYSES

The spent fuel storage facility is designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans.

The spent fuel assembly storage satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2).

2

LCO

Table 6 3

The restrictions on the placement of fuel assemblies within the spent fuel pool, according to ~~[Figure]~~ Table 3.7.18-1], in the accompanying LCO, ensures that the k_{eff} of the spent fuel pool will always remain < 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool according to ~~[Figure]~~ Table 3.7.18-1], in the accompanying LCO. Fuel assemblies not meeting the criteria of ~~[Figure]~~ 3.7.18-1] shall be stored in accordance with Specification 4.3.1.1.

4 6 1 6 1

(continued)

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BASES (continued)

APPLICABILITY — This LCO applies either whenever any fuel assembly is stored in [Region 2] II of the spent fuel pool or the north tilt pit.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in [Region 2] II the spent fuel pool is not in accordance with Figure Table [3.7.18-1], immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure Table [3.7.18-1].

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.18.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure Table [3.7.18-1] in the accompanying LCO prior to placing the fuel assembly in a Region II storage location. ~~For fuel assemblies in the unacceptable range of [Figure 3.7.18-1], performance of this SR will ensure compliance with Specification 4.3.1.1.~~

REFERENCES — None.

< INSERT >

Revised
03/15/99

Change	Discussion
Note:	This attachment provides a brief discussion of the deviations from NUREG-1432 that were made to support the development of the Palisades Nuclear Plant ITS. The Change Numbers correspond to the respective deviation shown on the "NUREG MARKUPS." The first five justifications were used generically throughout the markup of the NUREG. Not all generic justifications are used in each specification.

1. The brackets have been removed and the proper plant specific information or value has been provided.

2. Deviations have been made for clarity, grammatical preference, or to establish consistency within the Improved Technical Specifications. These deviations are editorial in nature and do not involve technical changes or changes of intent.

3. The requirement/statement has been deleted since it is not applicable to this facility. The following requirements have been renumbered, where applicable, to reflect this deletion.

4. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the facility specific nomenclature, number, reference, system description, or analysis description.

5. This change reflects the current licensing basis/technical specification. The design of the Fuel Handling Area Ventilation System is such that there is only one "train" not two as NUREG-1432 describes. The cleanup portion of the system contains two filter banks (normal and emergency) configured in a parallel flow arrangement, and two independent exhaust fans drawing air from a common duct. The "normal filter bank" contains a prefilter and a High Efficiency Particulate Air (HEPA) filter. The "emergency filter bank" contains a prefilter, HEPA filter, and an activated charcoal filter. As such, all statements concerning two trains, the reference to the second train, or single failure proof are deleted.

Change	Discussion
6.	The Applicability of ISTS 3.7.14 has been revised to match the plant conditions when the potential for a fuel handling accident exist. This includes Core Alterations, or movement of any fuel assembly in the containment, when irradiated fuel assemblies with < 30 days decay time are in the containment and the equipment hatch is opened. With the equipment hatch opened, the containment atmosphere is in direct contact with the fuel handling building atmosphere. In the event of a fuel handling accident in containment, the Fuel Handling Area Ventilation system is capable of filtering the airborne radioactive material in the containment atmosphere prior to being released to the outside atmosphere. In addition, the Applicability also includes movement of a fuel cask in or over the spent fuel pool. The fuel cask drop accident is presented in FSAR Section 14.11 and forms the bounding heavy load accident involving damage to stored irradiated fuel in the spent fuel pool. Conforming changes have been made to the Bases.
7.	The Actions of ISTS 3.7.14 have been revised to address the most probable causes for failure to meet the requirements of proposed LCO 3.7.12. As such, the Actions address the conditions when the Fuel Handling Area Ventilation system is inoperable, not properly aligned, or not in operation. Since the Fuel Handling Area Ventilation system consists of a single train aligned in its accident mitigation mode, the only appropriate Required Actions upon failure to meet the LCO is to immediately suspend fuel handling, CORE ALTERATIONS, and fuel cask movement activities. Conforming changes have been made to the Bases.
8.	ISTS SR 3.7.14.1 requires each FBAC train be operated for ≥10 hours (for plants with heaters), or ≥15 minutes (for plants without heaters) every 31 days. The intent of this SR is to ensure the standby FBAC system functions properly. For plants that rely on automatic actuation signals, or whose Applicability includes Modes 1, 2, 3, and 4, performance of this SR fulfill the intended function. However, for the Palisades plant, the Fuel Handling Area Ventilation system is required to be in operation whenever the plant is in the condition specified in the Applicability. Since SRs are only required to be met during the condition specified in the Applicability, performance of a system functional test would be redundant to the requirements of the LCO. Therefore, specifying this SR in the ITS is not necessary.

Change **Discussion**

9. This change reflects current plant design. The Fuel Handling Area Ventilation system does not include an automatic actuation feature. The system is manually configured in the emergency filtration mode prior to entering the conditions specified in the Applicability. As such, the tests required by ISTS 3.7.14.3 and ISTS 3.7.14.5 are not applicable and have not been included in the ITS.

10. ISTS SR 3.7.14.5 has been changed to reflect current analysis assumptions and methods of performance that prove the Fuel Handling Area Ventilation system is operating as required. The analysis that credits the Fuel Handling Area Ventilation System assumes a flow rate of 7300 cfm +/- 20%. No specific assumptions are made to the internal pressure of the fuel handling building relative to atmospheric pressure. Performance of this test on a Staggered Test Basis is not applicable since the Fuel Handling Area Ventilation system consists of a single train.

11. The Bases Background section for ISTS 3.7.14 has been revised to reflect the design of the Palisade's Fuel Handling Area Ventilation system. The level of detail provided in the revised Bases is comparable to the level of detail provided in the ISTS.

12. The Bases Applicable Safety Analyses section for ISTS 3.7.14 has revised to reflect specific plant analyses. The fuel handling accident in the fuel handling area is bounded by the fuel handling accident in containment which assumes all fission products release to the containment atmosphere are released to the outside environment with no credit for filtration. The NRC has previously concluded that the consequences of a fuel handling accident in the spent fuel area are acceptable with or without the charcoal filters operating. As part of Amendment 81 to the Palisade's Technical Specifications the NRC stated "the dose with the filter system operating was calculated to be 9 rem to the thyroid. If the filtration system was not operating, the dose would have been 91 rem which is still appropriately within the guidelines of 10 CFR 100 (i.e., < 100 rem thyroid)." Since operation of the Fuel Handling Area Ventilation System is not part of a primary success path that functions to mitigate a design basis accident, but instead, has been shown to be significant to public health and safety, the criterion satisfied in 10 CFR 50.36 for the fuel handling accident has been stated as Criterion 4.

Change **Discussion**

12 (continued)

The fuel cask drop accident forms the basis for a heavy load drop in the spent fuel pool that results in damage to stored irradiated fuel assemblies. This analysis was performed to support storage of spent fuel assemblies in the Independent Spent Fuel Storage Installation and is discussed in FSAR Section 14.11. The analysis shows acceptable radiological consequences when crediting filtration by the Fuel Handling Area Ventilation System for certain scenarios. Since operation of the Fuel Handling Area Ventilation System is part of a primary success path that functions to mitigate a design basis accident, the criterion satisfied in 10 CFR 50.36 for the cask drop accident has been stated as Criterion 3.

Change **Discussion**

Note: This attachment provides a brief discussion of the deviations from NUREG-1432 that were made to support the development of the Palisades Nuclear Plant ITS. The Change Numbers correspond to the respective deviation shown on the "NUREG MARKUPS." The first five justifications were used generically throughout the markup of the NUREG. Not all generic justifications are used in each specification.

1. The brackets have been removed and the proper plant specific information or value has been provided.

2. Deviations have been made for clarity, grammatical preference, or to establish consistency within the Improved Technical Specifications. These deviations are editorial in nature and do not involve technical changes or changes of intent.

3. The requirement/statement has been deleted since it is not applicable to this facility. The following requirements have been renumbered, where applicable, to reflect this deletion.

4. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the facility specific nomenclature, number, reference, system description, or analysis description.

5. This change reflects the current licensing basis/technical specification.

6. The storage of failed fuel is accomplished by the use of canisters that fit in the same storage racks as the fuel assemblies themselves. Therefore, the storage pool does not have any specifically designed rack(s) for failed fuel. The reference to a specific number of storage locations for failed fuel is deleted.

<u>Change</u> <u>Discussion</u>

7. ISTS 3.7.18 applies to plants which restrict the storage of fuel assemblies in high density storage locations based on meeting an acceptable combination of initial enrichment and discharge burnup. For fuel assemblies which do not meet the initial enrichment and discharge burnup requirements, the assemblies may be stored in compliance with other NRC approved methods or configurations as stipulated in ISTS 4.3.1.1. ISTS SR 3.7.18.1 requires an administrative verification of the initial enrichment and discharge burnup of a fuel assembly prior to storing any assembly in a Region 2 location. For the Palisades Plant, storage of fuel assemblies in high density racks (Region II) is only permitted for fuel assemblies which meet the initial enrichment and discharge burnup requirements. Alternate storage methods or configurations (e.g., checkerboading) in Region II has not been approved by the NRC. Therefore, reference to storage of fuel assemblies in accordance with Specification 4.3.1.1 in the LCO, SR, and SR Bases has been deleted. Assurance that fuel assembly enrichments do not exceed the limits of Region I locations (ITS 4.3.1.1) is controlled administratively in the design of new cores and the procurement of new fuel.