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Perry Nuclear Power Plant
Docket No. 50-440
Submittal of Technical Specification Bases, Revision 5

Ladies and Gentlemen:

Pursuant to the requirements of Section 5.5.11.d of the Perry Nuclear Power Plant Technical Specifications, a copy of the Technical Specification Bases, Revision 5 is hereby submitted. This submittal reflects the changes made subsequent to the changes reported in the Revision 4 Technical Specification Bases submittal, which was provided by letter dated September 3, 2003. The changes are identified by sidebars.

If you have questions or require additional information, please contact Mr. Gregory A. Dunn, FirstEnergy Nuclear Operating Company, Fleet Licensing Manager, at (330) 315-7243.

Very truly yours,

Enclosure

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

A001

TECHNICAL SPECIFICATION BASES
REVISION 5
LIST OF EFFECTIVE PAGES

PAGE NUMBER	REVISION NUMBER	PAGE NUMBER	REVISION NUMBER
B3.0-5	5	B3.4-21a	5
B3.0-5a	5	B3.4-22	5
B3.0-5b	5	B3.4-36	5
B3.0-6	5	B3.4-37	5
B3.0-14	5	B3.4-42	5
B3.0-14a	5	B3.4-46	5
B3.0-15	5	B3.4-47	5
B3.2-1	5	B3.5-3	5
B3.2-2	5	B3.5-6	5
B3.2-3	5	B3.5-6a	5
B3.2-10	5	B3.5-12	5
B3.2-10a	5	B3.5-13	5
B3.2-11a	5	B3.5-13a	5
B3.3-13	5	B3.5-14	5
B3.3-41j	5	B3.5-21	5
B3.3-57	5	B3.5-22	5
B3.3-58	5	B3.5-22a	5
B3.3-61	5	B3.6-41	5
B3.3-62	5	B3.6-41a	5
B3.3-65	5	B3.6-42	5
B3.3-91	5	B3.6-89	5
B3.3-124	5	B3.6-98	5
B3.3-144	5	B3.6-103	5
B3.3-159	5	B3.7-2	5
B3.3-159a	5	B3.7-3	5
B3.3-160	5	B3.7-3a	5
B3.3-160a	5	B3.7-6	5
B3.3-173a	5	B3.7-15	5
B3.3-210	5	B3.8-4	5
B3.4-3	5	B3.8-4a	5
B3.4-4	5	B3.8-8	5
B3.4-5	5	B3.8-8a	5
B3.4-5a	5	B3.8-47	5
B3.4-6	5	B3.8-50	5

TECHNICAL SPECIFICATION BASES
REVISION 5
LIST OF EFFECTIVE PAGES

PAGE NUMBER	REVISION NUMBER	PAGE NUMBER	REVISION NUMBER
B3.10-6	5		

BASES

LCO 3.0.3 (continued)

assemblies in the associated fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.7 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.7 of "Suspend movement of irradiated fuel assemblies in the associated fuel storage pool(s)" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

(continued)

BASES

LCO 3.0.4 (continued)

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented.

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the

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BASES

LCO 3.0.4 (continued)

Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk, and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., RCS Specific Activity), and may be applied to other Specifications based on NRC plant-specific approval.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability. Startup with inoperable equipment should be the exception rather than the rule. The LCO 3.0.4.b allowance should be used only when it has been determined that there is a high likelihood that the LCO will be satisfied within the Required Action's Completion Time, after the Mode change.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the

(continued)

BASES

LCO 3.0.4 (continued)

provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2 or 3, MODE 2 to MODE 3, and MODE 3 to MODE 4.

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

(continued)

BASES

SR 3.0.3 (continued)	Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit.

SR 3.0.4 contains two exceptions which explain its interrelationship with SR 3.0.3 and LCO 3.0.4. The first exception is in the first sentence, and clarifies that SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3. A provision is also included in the second sentence of SR 3.0.4 to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to a Surveillance not being met, in accordance with LCO 3.0.4. In certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment or variables outside specified limits. When equipment is inoperable, or variables are outside their specified limits, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency, on equipment that is inoperable, or on variables that are outside specified limits, does not result in an SR 3.0.4 restriction to

(continued)

BASES

SR 3.0.4 (continued)

changing MODES or other specified conditions in the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2 or 3, MODE 2 to MODE 3, and MODE 3 to MODE 4.

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BASES

SR 3.0.4 (continued)	The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND	The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the fuel design limits are presented in the USAR, Chapters 4, 6, and 15, and in References 1 and 2. The analytical methods and assumptions used in evaluating LOCA and normal operations that determine APLHGR limits are presented in USAR, Chapters 4, 6, and 15, and in References 1, 2, 3, and 4. APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to 10 CFR 50.46 during the limiting LOCA. Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Ref. 5) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. $MAPFAC_f$ curves are provided based on the maximum credible flow runout transient for Loop Manual and Non Loop Manual operation. The result of a single failure or single operator error during Loop Manual operation is the runout of only one loop because both recirculation loops are under independent control. Non Loop Manual operational modes allow simultaneous runout of both loops because a single controller regulates core flow.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closure and turbine control valve fast closure scram signals are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 23.8% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and LOCA. A complete discussion of the analysis code is provided in Reference 6. The ECCS/LOCA analysis assumes the existence of MAPFAC.

LOCA analyses are performed to ensure that the above determined APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A discussion of the analysis code is provided in Reference 7. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor.

For single recirculation loop operation, the MAPFAC multiplier is limited to a maximum value which is specified in the COLR. This multiplier is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA.

The APLHGR satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The APLHGR limits specified in the COLR are a function of exposure and are a result of DBA analyses. For two recirculation loops operating, the limit is determined by multiplying the smaller of the $MAPFAC_f$ and $MAPFAC_p$ factors times the exposure dependent APLHGR limits. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent APLHGR limit by the smallest of $MAPFAC_f$, $MAPFAC_p$, and the limiting value specified for single recirculation loop operation in the COLR, which has been determined by a specific single recirculation loop analysis (Ref. 2).

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA analyses that are assumed to occur at high power levels. Design calculations and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 4.7% to 14.2% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor (IRM) scram function provides rapid scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels < 23.8% RTP, the reactor operates with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

A.1

If any APLHGR exceeds the required limit, an assumption regarding an initial condition of the DBA analysis may not be met. Therefore, prompt action is taken to restore the APLHGR(s) to within the required limit(s) such that the plant will be operating within analyzed conditions and within the design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limit and is acceptable based on the low probability of a LOCA occurring simultaneously with the APLHGR out of specification.

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND	The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs), and to ensure that the peak clad temperature (PCT) during postulated Design Basis Loss of Coolant Accident (LOCA) does not exceed the limits specified in 10 CFR 50.46. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure or inability to cool the fuel does not occur during the anticipated operating conditions identified in USAR Chapters 6 and 15.
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the fuel design limits are presented in the USAR, Chapters 4, 6, and 15, and in References 1 and 2. The analytical methods and assumptions used in evaluating AOOs and normal operation that determine the LHGR limits are presented in USAR Chapters 4 and 15, and in References 1 and 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are: a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and b. Severe overheating of the fuel rod cladding caused by inadequate cooling. A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 1).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGR up to the operating limit LHGR specified in the COLR.

(continued)

BASES (continued)

APPLICABILITY

The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At THERMAL POWER levels < 23.8% RTP, the reactor is operating with substantial margin to the LHGR limits and this LCO is not required.

ACTIONS

A.1

If any LHGR exceeds the required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action is taken to restore the LHGR(s) to within required limit(s) such that the plant will be operating within analyzed conditions and within the design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the LHGR(s) to within its limit and is acceptable based on the low probability of a transient or Design Basis LOCA occurring simultaneously with the LHGR out of specification.

B.1

If the LHGR cannot be restored to within its required limit within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 23.8% RTP within 4 hours. The allowed

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. Reactor Vessel Water Level-Low, Level 3 (continued)

The Function is required in MODES 1 and 2 where considerable energy exists in the RCS resulting in the limiting transients and accidents. ECCS initiations at Reactor Vessel Water Level-Low Low, Level 2 and Low Low Low, Level 1 provide sufficient protection for level transients in all other MODES.

An operating bypass of the reactor vessel low water level trip is provided with the PEI keylock switches in the 'BYPASS' position and the mode switch in the 'SHUTDOWN' (MODE 3) position. The interlock with the mode switch will ensure that the reactor is in the shutdown condition prior to bypassing the reactor water level 3 scram.

5. Reactor Vessel Water Level-High, Level 8

High RPV water level indicates a potential problem with the feedwater level control system, resulting in the addition of reactivity associated with the introduction of a significant amount of relatively cold feedwater. Therefore, a scram is initiated at Level 8 to ensure that MCPR is maintained above the MCPR SL. The Reactor Vessel Water Level-High, Level 8 Function is one of the many Functions assumed to be OPERABLE and capable of providing a reactor scram during transients analyzed in Reference 3. It is directly assumed in the analysis of feedwater controller failure, maximum demand (Ref. 4).

Reactor Vessel Water Level-High, Level 8 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. The Reactor Vessel Water Level-High, Level 8 Allowable Value is specified to ensure that the MCPR SL is not violated during the assumed transient.

Four channels of the Reactor Vessel Water Level-High, Level 8 Function, with two channels in each trip system arranged in a one-out-of-two logic, are available and are required to be OPERABLE when THERMAL POWER is ≥ 23.8% RTP to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. With THERMAL POWER < 23.8% RTP, this Function is not required since MCPR is not a concern below 23.8% RTP.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3.6 (continued)

As noted, neutron detectors are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time. RPS RESPONSE TIME tests are conducted on a 24 month STAGGERED TEST BASIS. This Frequency is based upon operating experience, which shows that random failures of instrumentation components causing serious time degradation, but not channel failure, are infrequent.

REFERENCES

1. NEDO-31960-A, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," November 1995.

2. NEDO 31960-A, Supplement 1, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," November 1995.

3. NRC Letter, A. Thadani to L. A. England, "Acceptance for Referencing of Topical Reports NEDO-31960 and NEDO-31960 Supplement 1, 'BWR Owners Group Long-Term Stability Solutions Licensing Methodology'," July 12, 1993.

4. Generic Letter 94-02, "Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors," July 11, 1994.

5. USAR Section 15B.4.4 Thermal and Hydraulic Design.

6. NEDO-32465-A, "BWR Owners' Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications," August 1996.

7. CENPD-400-P-A, Rev 01, "Generic Topical Report for the ABB Option III Oscillation Power Range Monitor (OPRM)," May 1995.

(continued)

BASES

LCO

9. Penetration Flow Path. Primary Containment Isolation Valve (PCIV) Position (continued)

penetration flow path is not needed to determine status. Therefore, the position indication for valves in an isolated penetration is not required to be OPERABLE.

10. Deleted.

11. Primary Containment Pressure

Primary containment pressure is a Category I variable provided to verify RCS and containment integrity and to verify the effectiveness of ECCS actions taken to prevent containment breach. Two wide range primary containment pressure signals are transmitted from separate pressure transmitters and are continuously recorded and displayed on two control room recorders. These recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

(continued)

BASES

LCO (continued)

12. Primary Containment Air Temperature

Containment air temperature is a Category I variable provided to detect breach of the RCPB and to verify ECCS functions that operate to maintain RCS integrity. Eight wide range primary containment air temperature signals (four per channel) are transmitted from separate temperature elements and are continuously recorded and displayed on two control room recorders. However, two primary containment air temperature signals provide sufficient information to perform the above functions, and therefore only two in each Function are required to be OPERABLE. The recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1 and 2. These variables are related to the diagnosis and preplanned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1 and 2. In MODES 3, 4, and 5, plant conditions are such that the likelihood of an event that would require PAM instrumentation is extremely low; therefore, PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

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

(continued)

BASES

ACTIONS

F.1 (continued)

NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels. The special report shall be submitted in accordance with 10 CFR 50.4 within 14 days of entering Condition F.

SURVEILLANCE REQUIREMENTS

The following SRs apply to each PAM instrumentation Function in Table 3.3.3.1-1, except as noted below.

SR 3.3.3.1.1

For all Functions, performance of the CHANNEL CHECK once every 31 days ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The Primary Containment and Drywell Gross Gamma Radiation Monitors should be compared to similar plant instruments located throughout the plant.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1.1 (continued)

The Frequency of 31 days is based upon plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of those displays associated with the required channels of this LCO.

SR 3.3.3.1.2

Deleted.

SR 3.3.3.1.3

For all Functions a CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop including the sensor. The test verifies that the channel responds to the measured parameter with the necessary range and accuracy. The CHANNEL CALIBRATION for the Penetration Flow Path, PCIV Position consists of the Position Indicator Test (PIT), which is conducted in accordance with the ASME inservice inspection and testing program. The CHANNEL CALIBRATION for primary Containment/Drywell Area Gross Gamma Radiation Monitors shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source. The Frequency is based on operating experience and consistency with the typical industry refueling cycles.

REFERENCES

1. Regulatory Guide 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 2, December 1980.

2. USAR, Table 7.1-4.

BASES (continued)

APPLICABILITY	The Remote Shutdown System LCO is applicable in MODES 1 and 2. This is required so that the plant can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room. This LCO is not applicable in MODES 3, 4, and 5. In these MODES, the plant is already subcritical and in a condition of reduced Reactor Coolant System energy. Under these conditions, considerable time is available to restore necessary instrumentation or control Functions if control room instruments or controls become unavailable. Consequently, the TS do not require OPERABILITY in MODES 3, 4, and 5.
ACTIONS	A Note has been provided to modify the ACTIONS related to Remote Shutdown System Functions. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable Remote Shutdown System Functions provide appropriate compensatory measures for separate Functions. As such, a Note has been provided that allows separate Condition entry for each inoperable Remote Shutdown System Function. <u>A.1</u> Condition A addresses the situation where one or more required Functions of the Remote Shutdown System is inoperable. This includes the control and transfer switches for any required Function.

(continued)

BACKGROUND

High Pressure Core Spray System (continued)

The HPCS System also monitors the water levels in the condensate storage tank (CST) and the suppression pool, since these are the two sources of water for HPCS operation. Reactor grade water in the CST is the normal and preferred source. However, only the capability to take suction from the suppression pool is required for OPERABILITY. Upon receipt of a HPCS initiation signal, the CST suction valve is automatically signaled to open (it is normally in the open position), unless the suppression pool suction valve is open. If the water level in the CST falls below a preselected level, first the suppression pool suction valve automatically opens, and then the CST suction valve automatically closes. Two level transmitters are used to detect low water level in the CST. Either transmitter and associated trip unit can cause the suppression pool suction valve to open and the CST suction valve to close. Similarly, two level transmitters are used to detect high water level in the suppression pool. The suppression pool suction valve also automatically opens and the CST suction valve closes if high water level is detected in the suppression pool. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.

The HPCS System provides makeup water to the reactor until the reactor vessel water level reaches the high water level (Level 8) trip, at which time the HPCS injection valve closes. The HPCS pump will continue to run on minimum flow. The logic is one-out-of-two taken twice to provide high reliability of the HPCS System. The injection valve automatically reopens if a low low water level signal is subsequently received.

Automatic Depressurization System

ADS may be initiated by either automatic or manual means. Automatic initiation occurs when signals indicating Reactor Vessel Water Level-Low Low Low, Level 1; confirmed Reactor Vessel Water Level-Low, Level 3; and either LPCS or LPCI Pump Discharge Pressure-High are all present, and the ADS Initiation Timer has timed out. There are two transmitters

(continued)

B 3.3 INSTRUMENTATION

B 3.3.5.2 Reactor Core Isolation Cooling (RCIC) System Instrumentation

BASES

BACKGROUND

The purpose of the RCIC System instrumentation is to initiate actions to ensure adequate core cooling when the reactor vessel is isolated from its primary heat sink (the main condenser) and normal coolant makeup flow from the Reactor Feedwater System is unavailable, such that initiation of the low pressure Emergency Core Cooling Systems (ECCS) pumps does not occur. A more complete discussion of RCIC System operation is provided in the Bases of LCO 3.5.3, "RCIC System."

The RCIC System may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level-Low Low, Level 2. The variable is monitored by four transmitters that are connected to four trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic arrangement. Once initiated, the RCIC logic seals in and can be reset by the operator only when the reactor vessel water level signals have cleared.

The RCIC CST first and second test return valves close on a RCIC initiation signal to allow full system flow. The RCIC System also monitors the water levels in the condensate storage tank (CST) and the suppression pool, since these are the two sources of water for RCIC operation. Reactor grade water in the CST is the normal source. However, only the capability to take suction from the suppression pool is required for OPERABILITY. Upon receipt of a RCIC initiation signal, the CST suction valve is automatically signaled to open (it is normally in the open position) unless the pump suction from the suppression pool valve is open. If the water level in the CST falls below a preselected level, first the suppression pool suction valve automatically opens and then the CST suction valve automatically closes. Two level transmitters are used to detect low water level in the CST. Either transmitter and associated trip unit can cause the suppression pool suction valve to open and the CST suction valve to close. Similarly, two level transmitters are used to detect high water level in the suppression pool. The suppression pool suction valve also automatically opens and the CST suction valve closes if high water level is detected in the suppression pool. To prevent losing suction to the pump,

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.d. Condenser Vacuum-Low (continued)

OPERABLE in MODES 2 and 3, when all turbine stop valves (TSVs) are closed, since the potential for condenser overpressurization is minimized. Switches are provided to manually bypass the channels when all TSVs are closed.

This Function isolates the Group 6 valves.

1.e, 1.f. Main Steam Line Pipe Tunnel Ambient Temperature-High, Main Steam Line Turbine Building Temperature-High

Ambient Temperature-High is provided to detect a leak in the RCPB, and provides diversity to the high flow instrumentation. The isolation occurs when a very small leak has occurred. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. However, credit for these instruments is not taken in any transient or accident analysis in the USAR, since bounding analyses are performed for large breaks such as MSLBs.

Ambient temperature signals are initiated from thermocouples located in the area being monitored. Four channels of both Main Steam Line Pipe Tunnel Temperature-High and Main Steam Line Turbine Building Temperature-High Functions are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The ambient temperature monitoring Allowable Value is chosen to detect a leak equivalent to 25 gpm in the steam tunnel and 280 gpm in the Turbine Building.

These Functions isolate the Group 6 valves.

1.g. Manual Initiation

The Manual Initiation push button channels introduce signals into the MSL isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific USAR safety analysis that takes credit for this Function. It is retained for the isolation function as required by the NRC in the plant licensing basis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.k. SLC System Initiation (continued)

Two channels (one from each pump) of SLC System Initiation Function are required to be OPERABLE only in MODES 1 and 2, since these are the only MODES where the reactor can be critical, and these MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

4.l. Manual Initiation

The Manual Initiation push button channels introduce signals into the RWCU System isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific USAR safety analysis that takes credit for this Function. It is retained for the isolation function as required by the NRC in plant licensing basis.

There are four push buttons for the logic, two manual initiation push buttons per trip system. There is no Allowable Value for this Function, since the channels are mechanically actuated based solely on the position of the push buttons.

Four channels of the Manual Initiation Function are required to be OPERABLE.

5. RHR System Isolation

5.a. Ambient Temperature-High

RCIC steam piping still remains in the RHR equipment areas even after elimination of the Steam Condensing Mode of RHR, and a break in that portion of the steam pipe could result in increased temperatures in the RHR equipment areas. The requirements over the Ambient Temperature-High circuits which address isolation of such RCIC steam leaks in the RHR areas are provided by Function 3.h, "RCIC System Isolation, RHR Equipment Area Ambient Temperature-High", rather than by this Function (Function 5.a, which is an "RHR System Isolation").

The function of the RHR System Isolation Ambient Temperature-High instrumentation is to provide a diverse method from the Reactor Vessel Water Level-Low Level 3

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.a. Ambient Temperature-High (continued)

instruments for detecting a leak from RHR system piping which connects to the Reactor Coolant Pressure Boundary (RCPB) (Ref. 14). When a small or large leak is detected, the function also initiates an automatic RHR isolation to terminate further leakage from the RCPB. If a small leak is allowed to continue without isolation, offsite dose limits may be reached. This Function is not assumed in any USAR transient or accident analysis, since bounding analyses are performed for large breaks such as MSLBs.

Ambient Temperature-High signals are initiated from thermocouples that are appropriately located to protect the system that is being monitored. Two instruments monitor each RHR equipment area. Four channels for RHR Ambient Temperature-High

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.a. Ambient Temperature-High (continued)

Function are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The RHR System Isolation Ambient Temperature - High Function is only required to be OPERABLE in MODES 2 and 3 when the reactor vessel steam dome pressure is less than the RHR cut in permissive pressure. This is the only period when the valves in Valve Groups 3 and 4 which serve to isolate the RCPB (i.e., 1E12-F008, F009, F023, F053A and B) can be open, since in MODE 1 and in MODES 2 and 3 when the reactor vessel pressure is above the RHR cut in permissive pressure, a diverse signal from the Reactor Steam Dome Pressure-High Function will maintain the RHR RCPB isolation valves in these groups isolated. There are additional valves other than the five RCPB isolation valves listed above which were assigned at PNPP to RHR Valve Groups 3 and 4. Because these other non-RCPB valves were assigned to Valve Groups 3 and 4, in addition to receiving Loss Of Coolant Accident (LOCA) isolation(s), they also receive an isolation from the Ambient Temperature instruments. Although some of these non-RCPB RHR system valves are not maintained closed by the high RPV pressure signal, the Ambient Temperature instruments providing Function 5.a are not required by the TS to be OPERABLE when RPV pressure is above the cut in permissive pressure. Although these non-RCPB RHR system valves do receive an isolation signal upon a high temperature, that is not the specified safety function of the Ambient Temperature portion of the Leak Detection System, since as described in Ref. 14, the Leak Detection System is designed to detect leakage from the RCPB, in piping which is in direct communication with the reactor vessel.

In MODES 4 and 5, high temperature leakage is not a concern, and significant water leakage is isolated by the Reactor Vessel Water Level - Low, Level 3 Function.

The Allowable Value is set low enough to detect a leak equivalent to 25 gpm.

This Function isolates the Group 3 and 4 valves.

5.b. Reactor Vessel Water Level-Low, Level 3

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.b. Reactor Vessel Water Level-Low, Level 3 (continued)

some reactor vessel interfaces occurs to begin isolating the potential sources of a break. The Reactor Vessel Water Level-Low, Level 3 Function associated with RHR Shutdown Cooling System isolation is not directly assumed in any transient or accident analysis, since bounding analyses are performed for large breaks such as MSLBs. The RHR Shutdown Cooling System isolation on Level 3 supports actions to ensure that the RPV water level does not drop below the top of the active fuel during a vessel draindown event through the 1E12-F008 and 1E12-F009 valves caused by a leak (e.g., pipe break or inadvertent valve opening) in the RHR Shutdown Cooling System. The Reactor Vessel Water Level-Low, Level 3 channels required to be OPERABLE by Function 5.b are only those channels which are combined with the Reactor Vessel Pressure-High Function to provide isolation of the RHR Shutdown Cooling System suction from the reactor vessel (i.e., the 1E12-F008 and 1E12-F009 valves.)

(continued)

BASES

REFERENCES (continued)

11. Amendment No. 103 to Facility Operating License No. NPF-58, Perry Nuclear Power Plant, Unit 1.
12. USAR, Section 15.7.6
13. Amendment No. 122 to Facility Operating License No. NPF-58, Perry Nuclear Power Plant, Unit 1.
14. USAR, Section 7.6.1.3

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.1.3

The calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.7.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analyses of References 4, 5, and 6.

SR 3.3.7.1.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency is based on the assumption of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.7.1.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.7.3, "Control Room Emergency Recirculation (CRER) System," overlaps this Surveillance to provide complete testing of the assumed safety function.

The 24 month Frequency is based on the need to perform some of the Surveillance tests which satisfy this SR under the conditions that apply during a plant outage, and the potential for an unplanned transient if those particular tests were performed with the reactor at power. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

several seconds until the jet pump suction is uncovered (Ref. 1). The analyses assume that both loops are operating at the same flow prior to the accident. However, the LOCA analysis was reviewed for the case with a flow mismatch between the two loops, with the pipe break assumed to be in the loop with the higher flow. While the flow coastdown and core response are potentially more severe in this assumed case (since the intact loop starts at a lower flow rate and the core response is the same as if both loops were operating at a lower flow rate), a small mismatch has been determined to be acceptable based on engineering judgement.

The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during anticipated operational occurrences (AOOs) (Ref. 2), which are analyzed in Chapter 15 of the USAR.

A plant specific LOCA analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided THERMAL POWER is reduced to ≤ 2500 MWt, and the APLHGR and LHGR requirements are modified accordingly (Ref. 3).

The transient analyses of Chapter 15 of the USAR have also been performed for single recirculation loop operation (Ref. 3) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided THERMAL POWER is reduced to ≤ 2500 MWt, and the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System average power range monitor (APRM) instrument setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR, LHGR and MCPR limits for single loop operation are specified in the COLR. The APRM flow biased simulated thermal power setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation."

Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO	Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. Alternatively, with the limits specified in SR 3.4.1.1 not met, the recirculation loop with the lower flow must be considered to be not in operation. With only one recirculation loop in operation, THERMAL POWER must be ≤ 2500 MWt, and modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), LHGR limits (LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), and APRM Flow Biased Simulated Thermal Power-High setpoint (LCO 3.3.1.1) must be applied to allow continued operation consistent with the assumptions of Reference 3.
APPLICABILITY	In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

(continued)

BASES

APPLICABILITY (continued)	In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

A.1

With both recirculation loops operating but the recirculation loop flows not matched, Required Action A.1 requires that the recirculation loops must be restored to operation with matched flows within 2 hours. If the flow mismatch can not be restored to within limits within 2 hours, one recirculation loop must be declared to be "not in operation".

A recirculation loop is considered to be not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits. The loop with the lower flow must be considered not in operation. Should a LOCA or AOO occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA or AOO analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

Alternatively, if the single loop requirements of the LCO are applied to operating limits and RPS setpoints, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident or AOO sequence.

The 2 hour Completion Time is based on the low probability of an accident or AOO occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If large mismatches are detected, the condition should be alleviated by changing flow control valve position to re-establish forward flow or by tripping the pump, per plant procedures.

(continued)

BASES

ACTIONS (continued)

B.1

Should a LOCA or AOO occur with THERMAL POWER > 2500 MWt during single loop operation, the core response may not be bounded by the safety analyses. Therefore, only a limited time is allowed to reduce THERMAL POWER to ≤ 2500 MWt.

The 1 hour Completion Time is based on the low probability of an accident or AOO occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing changes in THERMAL POWER to be quickly detected.

(continued)

BASES

ACTIONS (continued)

C.1

If the required limit and setpoint modifications for single recirculation loop operation are not performed within 24 hours after transition from two recirculation loop operation to single recirculation loop operation, or requirements b.2, b.3, b.4 or b.5 of the LCO are not met for some other reason, the unit must be brought to a MODE in which the LCO does not apply (see Condition D). The 24 hour Completion Time of the Condition provides time before the required modifications to required limits and setpoints have to be in effect after a change in the reactor operating conditions from two recirculation loops operating to single recirculation loop operation. This time is provided due to the need to stabilize operation with one recirculation loop, including the procedural steps necessary to limit flow and adjust the flow control mode (to only Loop Manual mode) in the operating loop, and the complexity and detail required to fully implement and confirm the required limit and setpoint modifications. The 24 hour Completion Time is also based on the low probability of an accident or AOO occurring during this period, on a reasonable time to complete the Required Action, and on frequent monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

D.1

With no recirculation loops in operation, or the Required Action and associated Completion Time of Conditions A, B, or C not met, the unit is required to be brought to a MODE in which the LCO does not apply. The plant is required to be placed in MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.4.3

A manual actuation of each required S/RV is performed to verify that the valve is functioning properly. This SR can be demonstrated by one of two methods, which are described below. If performed by Method 1, plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements (Ref. 5), prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If performed by Method 2, valve OPERABILITY has been demonstrated for all installed S/RVs based upon successful offsite testing of the required test population.

Method 1:

Manual actuation of the S/RV with verification by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or any other method suitable to verify steam flow (e.g., tailpipe temperature or acoustic monitoring). Adequate reactor steam pressure must be available to perform this test to avoid damaging the valve. Also, adequate flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the S/RVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow is achieved to perform this test. Adequate pressure at which this test is performed is consistent with the pressure recommended by the valve manufacturer.

Method 2:

The required population of S/RVs tested will be stroked in the relief mode during testing at a qualified offsite facility to verify proper operation of the S/RV.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.4.3 (continued)

The successful performance of the S/RVs tested provides reasonable assurance that the remaining installed S/RVs will perform in a similar fashion. After the S/RVs are replaced, the power-operated actuator of all 19 S/RVs will be uncoupled from the S/RV stem, and cycled to ensure proper operation of the control circuit and actuator. Following cycling, the power-operated actuator is recoupled and the proper positioning of the stem nut is independently verified. This verifies that each S/RV will properly perform its intended function. If the valve actuator fails to operate due only to the failure of the solenoid but is capable of opening the valve on overpressure, the safety function of the S/RV is considered OPERABLE.

When removing and replacing the S/RVs, Foreign Material Exclusion controls will be in place to minimize the potential for unwanted materials from entering into any S/RV opening or the piping discharge lines.

SR 3.4.4.2 and the LOGIC SYSTEM FUNCTIONAL TEST performed in SR 3.3.6.4.4 overlap this surveillance to provide complete testing of the assumed safety function

The 24 months on a STAGGERED TEST BASIS Frequency ensures that each solenoid for each S/RV is alternately tested. The 24 month Frequency was developed based on the S/RV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 1). The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Sections III and XI.
2. USAR, Chapter 15, Appendix 15B.
3. USAR, Section 15.
4. NRC Safety Evaluation to NEDC-31753P, March 8, 1993.
5. ASME/ANSI OM-1995, Appendix I, Operations and Maintenance of Nuclear Power Plants, Part 1.

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, leakage detection systems are required to be OPERABLE to support LCO 3.4.5. This Applicability is consistent with that for LCO 3.4.5.

ACTIONS

A.1

With the drywell floor drain sump monitoring system inoperable, manual methods of determining the sump in leakage rate can provide the equivalent information to quantify leakage. In addition, the drywell atmospheric particulate or atmospheric gaseous monitor and the upper drywell air cooler condensate flow rate monitor will provide indications of changes in leakage.

With the drywell floor drain sump monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every 12 hours (SR 3.4.5.1) using alternate methods such as the pump timer, operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still operable.

B.1

With both particulate and gaseous drywell atmospheric monitoring channels inoperable, grab samples of the drywell atmosphere shall be taken and analyzed to provide periodic leakage information. Provided a sample is obtained and analyzed every 24 hours, the plant may continue operation since at least one other form of drywell leakage detection (i.e., upper drywell air cooler condensate flow rate monitoring system) is available. The 24 hour interval provides periodic information that is adequate to detect LEAKAGE.

(continued)

BASES

ACTIONS (continued)

C.1

With the required upper drywell air cooler condensate flow rate monitoring system inoperable, SR 3.4.7.1 is performed every 8 hours to provide periodic information of activity in the drywell at a more frequent interval than the routine Frequency of SR 3.4.7.1. The 8 hour interval provides periodic information that is adequate to detect LEAKAGE and recognizes that other forms of leakage detection are available. However, this Required Action is modified by a Note that allows this action to be not applicable if the required drywell atmospheric monitoring system is inoperable. Consistent with SR 3.0.1, Surveillances are not required to be performed on inoperable equipment.

D.1 and D.2

With both the particulate and gaseous drywell atmospheric monitoring channels and the upper drywell air cooler condensate flow rate monitoring system inoperable, the only means of detecting LEAKAGE is the drywell floor drain sump monitoring system. This Condition does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable monitoring systems to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of a limiting event while exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to $\leq 0.2\ \mu Ci/gm$ within 48 hours, or if at any time it is $> 4.0\ \mu Ci/gm$, it must be determined at least every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR 100 during a postulated MSLB accident.

Alternately, the plant can be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for bringing the plant to MODES 3 and 4 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.

(continued)

BASES

APPLICABILITY (continued)	The requirements for decay heat removal in MODES 4 and 5 are discussed in LCO 3.4.10, "Residual Heat Removal (RHR) Shutdown Cooling System-Cold Shutdown"; LCO 3.9.8, "Residual Heat Removal (RHR)-High Water Level"; and LCO 3.9.9, "Residual Heat Removal (RHR) -Low Water Level."
ACTIONS	A Note has been provided to modify the ACTIONS related to RHR shutdown cooling subsystems. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable shutdown cooling subsystems provide appropriate compensatory measures for separate inoperable shutdown cooling subsystems. As such, a Note has been provided that allows separate Condition entry for each inoperable RHR shutdown cooling subsystem. <u>A.1, A.2, and A.3</u> With one required RHR shutdown cooling subsystem inoperable for decay heat removal, except as permitted by LCO Note 2, the inoperable subsystem must be restored to OPERABLE status without delay. In this condition, the remaining OPERABLE subsystem can provide the necessary decay heat removal. The overall reliability is reduced, however, because a single failure in the OPERABLE subsystem could result in reduced RHR shutdown cooling capability. Therefore, an alternate method of decay heat removal must be provided.

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities.

The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as contributing to the alternate method capability. Alternate methods that can be used include (but are not limited to) the Reactor Water Cleanup System; pathway(s) to the main condenser in combination with method(s) capable of returning water to the reactor pressure vessel (RPV); or use of Automatic Depressurization System (ADS) Safety/Relief Valve(s) (SRV) to the suppression pool, in combination with method(s) capable of returning water to the RPV and method(s) capable of removing the heat from the containment.

Per Required Action A.3, the plant is also required to enter Mode 4. This action is required because the alternate methods of decay heat removal may not be as reliable as the RHR shutdown cooling subsystems.

B.1, B.2, and B.3

With no RHR shutdown cooling subsystem and no recirculation pump in operation, except as is permitted by LCO Note 1, reactor coolant circulation by the RHR shutdown cooling subsystem or one recirculation pump must be restored without delay.

Until RHR or recirculation pump operation is re-established, an alternate method of reactor coolant circulation must be placed into service. This will provide the necessary circulation for monitoring coolant temperature. The 1 hour Completion Time is based on the reactor coolant circulation function and is modified such that the 1 hour is applicable separately for each occurrence involving a loss of coolant circulation. Furthermore, verification of the functioning of the alternate method must be reconfirmed every 12 hours thereafter. This will provide assurance of continued temperature monitoring capability.

(continued)

BASES

BACKGROUND (continued)

The HPCS System (Ref. 3) consists of a single motor driven pump, a spray sparger above the core, and piping and valves to transfer water from the suction source to the sparger. Suction piping is provided from the CST and the suppression pool. Pump suction can be aligned to either the suppression pool or the CST. However, only the capability to take suction from the suppression pool is required for OPERABILITY. If the CST volume is low or the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCS System. The HPCS System is designed to provide core cooling over a wide range of RPV pressures (0 psid to 1200 psid, vessel to suction source). Upon receipt of an initiation signal, the HPCS pump automatically starts after AC power is available and valves in the flow path begin to open. Since the HPCS System is designed to operate over the full range of expected RPV pressures, HPCS flow begins as soon as the necessary valves are open. Full flow test lines are provided to route water from and to the suppression pool or CST to allow testing of the HPCS System during normal operation without spraying water into the RPV.

The ECCS pumps are provided with minimum flow lines, which discharge to the suppression pool. The valves in these lines automatically open to prevent pump damage due to overheating when other discharge line valves are closed or RPV pressure is greater than the LPCS or LPCI pump discharge pressures following system initiation. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the ECCS discharge line "keep fill" systems are designed to maintain all pump discharge lines filled with water.

The ADS (Ref. 4) consists of 8 of the 19 S/RVs. It is designed to provide depressurization of the primary system during a small break LOCA if HPCS fails or is unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems (LPCS and LPCI), so that these subsystems can provide core cooling. Each ADS valve is supplied with pneumatic power from an air storage system, which consists of air accumulators located in the drywell.

(continued)

BASES (continued)

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable HPCS subsystem. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable HPCS subsystem, and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

If any one low pressure ECCS injection/spray subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced because a single failure in one of the remaining OPERABLE subsystems concurrent with a LOCA may result in the ECCS not being able to perform its intended safety function. The 7 day Completion Time is based on a reliability study (Ref. 12) that evaluated the impact on ECCS availability by assuming that various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (i.e., Completion Times).

B.1 and B.2

If the HPCS System is inoperable, and the RCIC System is verified to be OPERABLE (when RCIC is required to be OPERABLE), the HPCS System must be restored to OPERABLE status within 14 days. In this Condition, adequate core cooling is ensured by the OPERABILITY of the redundant and diverse low pressure ECCS injection/spray subsystems in conjunction with the ADS. Also, the RCIC System will automatically provide makeup water at most reactor operating pressures. Verification of RCIC OPERABILITY within 1 hour is therefore required when HPCS is inoperable and RCIC is required to be OPERABLE. This may be performed by an administrative check, by examining logs or other

(continued)

BASES

ACTIONS (continued)

information, to determine if RCIC is out of service for maintenance or other reasons. It is not necessary to perform the Surveillance needed to demonstrate the OPERABILITY of the RCIC System. However, if the OPERABILITY of the RCIC System cannot be verified and RCIC is required to be OPERABLE, Condition D must be entered. If a single active component fails concurrent with a design basis LOCA, there is a potential, depending on the specific failure, that the minimum required ECCS equipment will not be available. A 14 day Completion Time is based on the results of a reliability study (Ref. 12) and has been found to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.5 (continued)

This SR is modified by a Note that excludes vessel injection/spray during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the full flow test line, coolant injection into the RPV is not required during the Surveillance.

SR 3.5.1.6

The ADS designated S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify that the mechanical portions (i.e., solenoids) of the ADS function operate as designed when initiated either by an actual or simulated initiation signal. SR 3.5.1.7 and the LOGIC SYSTEM FUNCTIONAL TEST performed in SR 3.3.5.1.6 overlap this Surveillance to provide complete testing of the assumed safety function.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

This SR is modified by a Note that excludes valve actuation. This prevents an RPV pressure blowdown.

SR 3.5.1.7

A manual actuation of each required ADS valve is performed to verify that the valve is functioning properly. This SR can be demonstrated by one of two methods, which are described below. If performed by Method 1, plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements (Ref. 17), prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If performed by Method 2, valve OPERABILITY has been demonstrated for all installed ADS valves based upon the successful offsite testing of the required test population.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.7 (continued)

Method 1:

Manual actuation of the ADS valve with verification by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or any other method suitable to verify steam flow (e.g., tailpipe temperature or acoustic monitoring). Adequate reactor steam pressure must be available to perform this test to avoid damaging the valve. Also, adequate flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the ADS valves divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow is achieved to perform this test. Adequate pressure at which this test is performed is consistent with the pressure recommended by the valve manufacturer.

Method 2:

The required population of ADS S/RVs tested will be stroked in the relief mode during testing at a qualified offsite facility to verify proper operation of the S/RV. The successful performance of the S/RVs tested provides reasonable assurance that the remaining installed S/RVs will perform in a similar fashion. After the S/RVs are replaced, the power-operated actuator of all 19 S/RVs will be uncoupled from the S/RV stem, and cycled to ensure proper operation of the control circuit and actuator. Following cycling, the power-operated actuator is recoupled and the proper positioning of the stem nut is independently verified. This verifies that each S/RV will properly perform its intended function. If the valve actuator fails to operate due only to the failure of the solenoid but is capable of opening the valve on overpressure, the safety mode of the S/RV is considered OPERABLE.

When removing and replacing the S/RVs, Foreign Material Exclusion controls will be in place to minimize the potential for unwanted materials from entering into any S/RV opening or the piping discharge lines.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.7 (continued)

SR 3.5.1.6 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1.6 overlap this Surveillance to provide complete testing of the safety function. The Frequency of 24 months on a STAGGERED TEST BASIS Frequency ensures that both solenoids for each ADS valve power-operated actuator are alternately tested. The Frequency of the required-power-operated actuator testing is based on the tests required by ASME OM, Part 1, (Ref. 17) as implemented by the Inservice Testing Program of Specification 5.5.6. The testing Frequency required by the Inservice Testing Program is based on operating experience and valve performance. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.1.8

This SR ensures that the ECCS RESPONSE TIMES are within limits for each of the ECCS injection and spray subsystems. This SR is modified by a note which identifies that the associated ECCS actuation instrumentation is not required to be response time tested. Response time testing of the remaining subsystem components is required. This is supported by Reference 15. Response time testing acceptance criteria are included in Reference 16.

ECCS RESPONSE TIME tests are conducted every 24 months. The 24 month Frequency is based on the need to perform this

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.8 (continued)

Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

REFERENCES

1. USAR, Section 6.3.2.2.3.
2. USAR, Section 6.3.2.2.4.
3. USAR, Section 6.3.2.2.1.
4. USAR, Section 6.3.2.2.2.
5. USAR, Section 15.6.6.
6. USAR, Section 15.6.4.
7. USAR, Section 15.6.5.
8. 10 CFR 50, Appendix K.
9. USAR, Section 6.3.3.
10. 10 CFR 50.46.
11. USAR, Section 6.3.3.3.
12. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCO's for ECCS Components," December 1, 1975.
13. USAR, Section 5.2.2.4.1.
14. ASME, Boiler and Pressure Vessel Code, Section XI.
15. NEDO-32291, "System Analyses for Elimination of Selected Response Time Testing Requirements," January 1994.
16. USAR, Section 6.3, Table 6.3-1.
17. ASME/ANSI OM-1995, Appendix I, Operations and Maintenance of Nuclear Power Plants, Part 1.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.3 RCIC System

BASES

BACKGROUND

The RCIC System is not part of the ECCS; however, the RCIC System is included with the ECCS section because of their similar functions.

The RCIC System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of RPV water level. Under these conditions, the High Pressure Core Spray (HPCS) and RCIC systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 1 are satisfied.

The RCIC System (Ref. 2) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the reactor vessel head spray nozzle. Suction piping is provided from the condensate storage tank (CST) and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. However, only the capability to take suction from the suppression pool is required for OPERABILITY. If the CST volume is low, or the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from main steam line A, upstream of the inboard main steam line isolation valve.

The RCIC System is designed to provide core cooling for a wide range of reactor pressures, 165 psia to 1215 psia. Rated flow is required up to 1118 psia, based on operation of the Safety Relief Valves in the Relief and Low-Low-Set modes (T.S. 3.3.6.4) during the vessel isolation transients for which RCIC is designed. Upon receipt of an initiation signal, the RCIC turbine accelerates to a specified speed. As the RCIC flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the RCIC System during normal operation without injecting water into the RPV.

(continued)

BASES

BACKGROUND (continued)	The RCIC pump is provided with a minimum flow line, which discharges to the suppression pool. The valve in this line automatically opens to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the RCIC System discharge line "keep fill" system is designed to maintain the pump discharge line filled with water.
APPLICABLE SAFETY ANALYSES	The function of the RCIC System is to respond to transient events by providing makeup coolant to the reactor. The RCIC System is not an Engineered Safety Feature System and no credit is taken in the safety analyses for RCIC System operation. Based on its contribution to the reduction of overall plant risk, however, the system is included in the Technical Specifications as required by the NRC Policy Statement.
LCO	The OPERABILITY of the RCIC System provides adequate core cooling such that actuation of any of the ECCS subsystems is not required in the event of RPV isolation accompanied by a loss of feedwater flow. The RCIC System has sufficient capacity to maintain RPV inventory during an isolation event.
APPLICABILITY	The RCIC System is required to be OPERABLE in MODE 1, and MODES 2 and 3 with reactor steam dome pressure > 150 psig since RCIC is the primary non-ECCS water source for core cooling when the reactor is isolated and pressurized. In MODES 2 and 3 with reactor steam dome pressure ≤ 150 psig, and in MODES 4 and 5, RCIC is not required to be OPERABLE since the ECCS injection/spray subsystems can provide sufficient flow to the vessel.
ACTIONS	A Note prohibits the application of LCO 3.0.4.b to an inoperable RCIC system. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable RCIC system, and the

(continued)

BASES

ACTIONS (continued)

provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1 and A.2

If the RCIC System is inoperable during MODE 1, or MODES 2 or 3 with reactor steam dome pressure > 150 psig, and the HPCS System is verified to be OPERABLE, the RCIC System must be restored to OPERABLE status within 14 days. In this Condition, loss of the RCIC System will not affect the overall plant capability to provide makeup inventory at high

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1

A manual actuation of each required LLS valve is performed to verify that the valve is functioning properly. This SR can be demonstrated by one of two methods, which is described below. If performed by Method 1, plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements (Ref. 4), prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If performed by Method 2, valve OPERABILITY has been demonstrated for all installed LLS valves based upon successful offsite testing of the required test population.

Method 1:

Manual actuation of the LLS valve with verification by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or any other method suitable to verify steam flow (e.g., tailpipe temperature or acoustic monitoring). Adequate reactor steam pressure must be available to perform this test to avoid damaging the valve. Also, adequate flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the LLS valves divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is performed is consistent with the pressure recommended by the valve manufacturer.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1

Method 2:

The required population of LLS S/RVs tested will be stroked in the relief mode during testing at a qualified offsite facility to verify proper operation of the S/RV. The successful performance of the S/RVs tested provides reasonable assurance that the remaining installed S/RVs will perform in a similar fashion. After the S/RVs are replaced, the power-operated actuator of all 19 S/RVs will be uncoupled from the S/RV stem, and cycled to ensure proper operation of the control circuit and actuator. Following cycling, the power-operated actuator is recoupled and the proper positioning of the stem nut is independently verified. This verifies that each S/RV will properly perform its intended function. If the valve actuator fails to operate due only to the failure of the solenoid but is capable of opening the valve on overpressure, the safety mode of the S/RV is considered OPERABLE.

When removing and replacing the S/RVs, Foreign Material Exclusion controls will be in place to minimize the potential for unwanted materials from entering into any S/RV opening or the piping discharge lines.

The STAGGERED TEST BASIS Frequency ensures that both solenoids for each LLS valve power-operated actuator are alternately tested. The 24 Month Frequency of the required power-operated actuator testing is based on the tests required by ASME OM Part 1 (Ref. 4) as implemented by the Inservice Testing Program of Specification 5.5.6. The testing Frequency required by the Inservice Testing Program is based on operating experience and valve performance. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENT (continued)

SR 3.6.1.6.2

The LLS function S/RVs are required to actuate automatically upon receipt of specific initiation signals. A functional test is performed to verify that the mechanical portions (i.e., solenoids) of the automatic LLS function operate as designed when initiated either by an actual or simulated automatic initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.4.4 overlaps this SR to provide complete testing of the safety function.

The 24 month Frequency is based on the need to perform this Surveillance during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

This SR is modified by a Note that excludes valve actuation. This prevents a reactor pressure vessel pressure blowdown.

REFERENCES

1. GESSAR-II, Appendix 3B, Attachment A, Section 3BA.8.
2. USAR, Section 7.6.1.11.
3. ASME, Boiler and Pressure Vessel Code, Section XI.
4. ASME/ANSI OM-1995, Appendix I, Operations and Maintenance of Nuclear Power Plants, Part 1.

BASES (continued)

REFERENCES	1.	USAR, Section 6.2.
	2.	USAR, Chapter 15.

BASES

ACTIONS

A.1 (continued)

accumulation from exceeding the flammability limit, and the low probability of failure of the OPERABLE hydrogen igniter division.

B.1 and B.2

With two primary containment and drywell hydrogen igniter divisions inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by at least one primary containment hydrogen recombiner and one combustible gas mixing subsystem. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. The verification may be performed as an administrative check by examining logs or other information to determine the availability of the alternate hydrogen control capabilities. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control capabilities. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two igniter divisions inoperable for up to 7 days. Seven days is a reasonable time to allow two igniter divisions to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

(continued)

BASES

APPLICABILITY (continued)

calculated for the DBA LOCA. Also, because of the limited time in this MODE, the probability of an accident requiring the Combustible Gas Mixing System is low. Therefore, the Combustible Gas Mixing System is not required in MODE 3.

In MODES 4 and 5, the probability and consequences of a LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, the Combustible Gas Mixing System is not required in these MODES.

ACTIONS

A.1

With one combustible gas mixing subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the hydrogen mixing function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced hydrogen mixing capability. The 30 day Completion Time is based on the low probability of failure of the OPERABLE combustible gas mixing subsystem, the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit, and the amount of time available after the event for operator action to prevent hydrogen accumulation from exceeding this limit.

B.1 and B.2

With two combustible gas mixing subsystems inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by one primary containment hydrogen recombiner or one division of the hydrogen

(continued)

BASES

BACKGROUND (continued)	Following a DBA or transient, the ESW System will operate automatically without operator action. Manual initiation of supported systems (e.g., suppression pool cooling) is, however, performed for long term cooling operations.
APPLICABLE SAFETY ANALYSES	The volume of Lake Erie is such that sufficient water inventory is available for all ESW System post LOCA cooling requirements for a 30 day period with no additional makeup water source available (Ref. 1). The ability of the ESW System to support long term cooling of the reactor or containment is assumed in evaluations of the equipment required for safe reactor shutdown presented in the USAR, Sections 9.2.1, 6.2.1.1.3.3, and Chapter 15, (Refs. 2, 4, and 5, respectively). These analyses include the evaluation of the long term primary containment response after a design basis LOCA. The ESW System provides cooling water for the RHR suppression pool cooling mode to limit suppression pool temperature and primary containment pressure following a LOCA. This ensures that the primary containment can perform its intended function of limiting the release of radioactive materials to the environment following a LOCA. The ESW System also provides cooling to other components assumed to function during a LOCA (e.g., RHR and Low Pressure Core Spray Systems via the Emergency Closed Cooling Water System). Also, the ability to provide onsite emergency AC power is dependent on the ability of the ESW System to cool the DGs. The safety analyses for long term containment cooling were performed, as discussed in the USAR, Sections 6.2.1.1.3.3 and 6.2.2 (Refs. 4 and 6, respectively), for a LOCA, concurrent with a loss of offsite power, and minimum available DG power. The worst case single failure affecting the performance of the ESW System is the failure of one of the two standby DGs, which would in turn affect one ESW subsystem. Reference 2 discusses ESW System performance during these conditions. The ESW System satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The OPERABILITY of the Division 1 and 2 ESW subsystems is required to ensure the effective operation of the RHR System in removing heat from the reactor, and the effective operation of other safety related equipment during a DBA or transient. Requiring both ESW subsystems to be OPERABLE ensures that either subsystem will be available to provide adequate capability to meet cooling requirements of the equipment required for safe shutdown in the event of a single failure.

An ESW subsystem is considered OPERABLE when:

a. The associated pump is OPERABLE; and

b. The associated piping, valves, instrumentation, and controls required to perform the safety related function are OPERABLE.

The isolation of the ESW System to components or systems may render those components or systems inoperable, but may not affect the OPERABILITY of the ESW System.

During the performance of maintenance, repair, or testing activities on an ESW sluice gate, the safety function of the ESW system must be maintained by ensuring that should a loss of the normal ESW intake occur, the alternate water source (discharge tunnel via a sluice gate) remains OPERABLE. The following two approaches describe how this can be done while taking into account the application of the single failure criterion discussed in Generic Letter (GL) 80-30.

First, the ESW loop in the division associated with the closed and inoperable sluice gate can be declared inoperable and the appropriate LCO Condition and Required Actions entered. Per GL 80-30, it is then not necessary to postulate a single failure of the OPERABLE sluice gate while the plant is operating in this time-limited condition. Should the normal ESW intake fail, the OPERABLE sluice gate would open, aligning the ESW loops to the alternate water source. Operators would then align the three ESW loops to the swale per plant procedures.

(continued)

BASES

LCO (continued)

An alternate method would be to align the 3 ESW loops to the swale and then deactivate at least one of the two ESW sluice gates in the open position. In this way either the operable or inoperable sluice gate may be used to maintain all ESW loops OPERABLE provided the sluice gate remains deactivated in the open position. In this way, should the normal ESW intake be lost, the alternate water source (discharge tunnel via the open sluice gate) would be available. This method may only be utilized when there is assurance that the recirculation of warm water from the normal Service Water (SW) system discharge into the ESW Forebay will not result in the ESW inlet temperature exceeding its maximum design limit.

OPERABILITY of the Division 3 ESW subsystem is addressed by LCO 3.7.2, "ESW System-Division 3."

APPLICABILITY

In MODES 1, 2, and 3, the Division 1 and 2 ESW subsystems are required to be OPERABLE to support OPERABILITY of the equipment serviced by these ESW subsystems and required to be OPERABLE in these MODES.

In MODES 4 and 5, the requirements of the ESW System are determined by the systems they support.

ACTIONS

A.1

If one Division 1 or Division 2 ESW subsystem is inoperable, it must be restored to OPERABLE status within 72 hours. With the unit in this condition, the remaining OPERABLE Division 1 or Division 2 ESW subsystem is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ESW subsystem could result in loss of ESW function.

(continued)

BASES

REFERENCES (continued)	4.	USAR, Section 6.2.1.1.3.3.
	5.	USAR, Chapter 15.
	6.	USAR, Section 6.2.2.
	7.	Deleted
	8.	USAR, Section 2.4.11

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.2 (continued)

parameters of the filtration system. (Note: Values identified in the VFTP are Surveillance Requirement values.) Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.3.3

This SR verifies that each CRER subsystem starts and operates on an actual or simulated initiation signal and the isolation dampers close within 10 seconds. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.5 overlaps this SR to provide complete testing of the safety function. The 24 month Frequency is based on the need to perform some of the surveillance tests which satisfy this SR under the conditions that apply during a plant outage, and the potential for an unplanned transient if those particular tests were performed with the reactor at power. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

SR 3.7.3.4

This SR verifies the integrity of the control room enclosure and the assumed inleakage rates of potentially contaminated air. The Control Room HVAC System is designed so that, when operating in the normal mode, the system automatically maintains a positive differential pressure between the control room and the outside environment. During an emergency, when the CRER System is operating, the supply (M25-F010A and M25-F020B for one train and M25-F010B and M25-F020A for the other train) and exhaust (M25-F130A and M25-F130B) dampers of the Control Room HVAC System are closed (no design admittance of outside air). When in the emergency recirculation mode of operation no attempt is made to pressurize the control room. Thus the leakage through the intake and exhaust dampers is the primary source of leakage into the control structure. The Frequency of 24 months is appropriate since it is consistent with most other valve leak tests, and since significant degradation of the dampers is not expected over this period of time.

(continued)

BASES

LCO (continued)

Proper sequencing of loads, including tripping of nonessential loads, is a required function for both offsite circuit and DG OPERABILITY for Divisions 1 and 2.

The AC sources in one division must be separate and independent (to the extent possible) of the AC sources in the other division(s). For the DGs, the separation and independence are complete. For the offsite AC sources, the separation and independence are to the extent practical.

APPLICABILITY

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

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

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

A Note has been added taking exception to the Applicability requirements for Division 3 AC sources, provided the HPCS System is declared inoperable. This exception is intended to allow declaring of the HPCS System inoperable either in lieu of declaring the Division 3 AC source inoperable, or at any time subsequent to entering ACTIONS for an inoperable Division 3 AC source. This exception is acceptable since, with the HPCS System inoperable and the associated ACTIONS entered, the Division 3 AC sources provide no additional assurance of meeting the above criteria.

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

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable DG. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable DG, and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

(continued)

BASES

ACTIONS (continued)

A.1

To ensure a highly reliable power source remains, it is necessary to verify the availability of the remaining required offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in the Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

(continued)

BASES

ACTIONS

B.3.1 and B.3.2 (continued)

In the event the inoperable DG is restored to OPERABLE status prior to completing either Required Actions B.3.1 or B.3.2, the corrective actions program will continue to evaluate the common cause failure possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is reasonable time to confirm that the OPERABLE DG(s) are not affected by the same problem as the inoperable DG.

B.4

In Condition B, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E distribution system. The 72 hour Completion Time for Division 3 and the 14 day Completion Time for Divisions 1 and 2 take into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

The Division 1 and 2 Emergency Diesel Generators (EDGs) have a 14 day Completion Time. The period of the Completion Time that is ≥ 72 hours is considered to be the "risk-informed" portion of the Completion Time. The provisions of the Configuration Risk Management Program (CRMP)in Specification 5.5.13.1 must be applied to use the risk-informed portion of the Completion Time. The risk-informed portion of the Completion Time is meant to be entered in a controlled fashion whenever possible with a work scope that limits the risk to the plant. The CRMP is controlled by plant administrative procedures.

One preventive maintenance outage using the risk-informed portion of the 14 day Completion Time may be planned for one EDG, Division 1 or 2, within a period of 365 days (one year). The basis of the "once per year" frequency is to minimize the number of times that major intrusive maintenance is performed on the diesels. In accordance with the CRMP and the acceptance criteria of RG 1.177, a qualitative assessment may be used to assess time periods between planned extended preventive maintenance outages of less than one year.

(continued)

BASES

ACTIONS

B.4 (continued)

The risk-informed provisions in the CRMP also apply if entry into the risk-informed portion of the Completion Time is necessary for corrective maintenance. In addition, the 10CFR50.65 (Maintenance Rule) program includes performance criteria for cumulative out-of-service time of the diesels. Entries into the risk-informed portion of the Completion Time for corrective maintenance do not affect the "once per year" frequency of planned extended outages described above, although such entries may shorten the allowable length of a planned preventive maintenance outage due to the Maintenance Rule performance criteria.

The third Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 72 hours. This situation could lead to a total of 17 days, since initial failure to meet the LCO, to restore the DG. At this time, an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 72 hours (for a total of 20 days) allowed prior to complete restoration of the LCO. The 17 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the Completion Times means that the three Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.3 (continued)

These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between the sample (and corresponding results) of new fuel, and addition of new fuel oil to the storage tanks to exceed 31 days. The limits and applicable ASTM Standards for the tests listed in the Diesel Fuel Oil Testing Program of Specification 5.5.9 are as follows:

a. Sample the new fuel oil in accordance with ASTM D4057-95 (Reapproved 2000)(Ref. 6);

b. Verify in accordance with the tests specified in ASTM D1298-85 (Ref. 6) that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89; an API gravity at 60°F of ≥ 26° and ≤ 39°; or an API gravity of within 0.3° at 60°F, or a specific gravity within 0.0016 at 60/60°F when compared to the supplier's certificate;

c. Verify in accordance with the tests specified in ASTM D975-89 (Ref. 6), a flash point of ≥ 125°F;

d. Verify in accordance with the tests specified in ASTM D975-89 (Ref. 6), if gravity was not determined by comparison with the supplier's certification, a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes; and

e. Verify that the new fuel oil has no visible free water or particulate contamination when tested in accordance with ASTM D4176-86 (Ref. 6).

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO since the fuel oil is not added to the storage tanks.

Following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-89 (Ref. 6) are met for new fuel oil when tested in accordance with ASTM D975-89 (Ref. 6). These additional analyses are required by Specification 5.5.9, Diesel Fuel Oil Testing Program, to be performed within 31 days following sampling and addition. This 31 days is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.6 (continued)

preventive maintenance. The presence of sediment does not necessarily represent a failure of this SR provided that accumulated sediment is removed during performance of the Surveillance.

REFERENCES

1. USAR, Section 9.5.4.
2. Regulatory Guide 1.137.
3. ANSI N195, Appendix B, 1976.
4. USAR, Chapter 6.
5. USAR, Chapter 15.
6. ASTM Standards: D4057-95 (Reapproved 2000); D1298-85; D975-89; D4176-86; D2276-88.
7. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.10 SPECIAL OPERATIONS

B 3.10.2 Reactor Mode Switch Interlock Testing

BASES

BACKGROUND

The purpose of this Special Operations LCO is to permit operation of the reactor mode switch from one position to another to confirm certain aspects of associated interlocks during periodic tests and calibrations in MODES 3, 4, and 5.

The reactor mode switch is a conveniently located, multiposition, keylock switch provided to select the necessary scram functions for various plant conditions (Ref. 1). The reactor mode switch selects the appropriate trip relays for scram functions and provides appropriate bypasses. The mode switch positions and related scram interlock functions are summarized as follows:

a. Shutdown-Initiates a reactor scram; bypasses main steam line isolation, reactor high water level scrams; and reactor low water level PEI bypass control switches become active (i.e., the PEI switches can BYPASS the level 3 trip if taken to the 'BYPASS' position.

b. Refuel-Selects Neutron Monitoring System (NMS) scram function for low neutron flux level operation (but does not disable the average power range monitor scram); bypasses main steam line isolation and reactor high water level scrams;

c. Startup/Hot Standby-Selects NMS scram function for low neutron flux level operation (intermediate range monitors and average power range monitors); bypasses main steam line isolation and reactor high water level scrams; and

d. Run-Selects NMS scram function for power range operation.

The reactor mode switch also provides interlocks for such functions as control rod blocks, scram discharge volume trip bypass, refueling interlocks, and main steam isolation valve isolations.

APPLICABLE SAFETY ANALYSES

The acceptance criterion for reactor mode switch interlock testing is to prevent fuel failure by precluding reactivity excursions or core criticality.

(continued)