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10CFR50.36
U-603532
November 5, 2001

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

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Transmittal of Revision 6 to the Clinton Power Station Technical Specification Bases

In accordance with Clinton Power Station (CPS) Technical Specification 5.5.11, "Technical Specification (TS) Bases Control Program," AmerGen Energy Company, LLC (i.e., AmerGen) is transmitting the revised pages constituting Revision 6 to the CPS TS Bases. The changes associated with this revision were processed in accordance with CPS TS 5.5.11 which became effective with Amendment No. 95 to the CPS Operating License. Compliance with CPS TS 5.5.11 requires updates to the TS Bases to be submitted to the NRC at a frequency consistent with 10CFR50.71(e).

Should you have any questions concerning this letter, please contact Mr. J. L. Peterson at (217) 937-2810.

Respectfully,

W. S. Iliff
W. S. Iliff
Regulatory Assurance Manager
Clinton Power Station

JLP\blf

Attachment – Attachment A: Revision 6 of the CPS Technical Specification Bases

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Clinton Power Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

A001

Attachment A
Clinton Power Station, Unit 1
Revision 6 to the CPS Technical Specification Bases

BASES

SAFETY LIMIT VIOLATIONS

2.2.2 (continued)

with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal. In the event reactor vessel water level is below the top of active irradiated fuel, water level would normally be restored by manually initiating Emergency Core Cooling Systems.

2.2.3

If any SL is violated, the CPS Plant Manager and the CPS Site Vice President shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 5). A copy of the report shall also be submitted to the CPS Plant Manager and the CPS Site Vice President.

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel, GESTAR-II," (latest approved revision).
3. 10 CFR 50.72.
4. 10 CFR 100.
5. 10 CFR 50.73.

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.2

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 5). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

includes allowances for short term transient operation above the operating limit to account for AOOs, plus an allowance for densification power spiking.

LHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during the limiting AOOs (Ref. 2). Flow dependent LHGR 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. The result of a single failure or single operator error is the runout of only one loop because both recirculation loops are under independent control.

Based on analyses of limiting plant transients (other than core flow increases) over 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 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 25% RTP and the previously mentioned bypass power level. The exposure dependent LHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and AOOs. A complete discussion of the analysis code is provided in Reference 6.

The LHGR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR.

The LHGR limits specified in the COLR are the result of fuel design and transient analyses. The limit is determined by multiplying the small of the $MAPFAC_f$ and $MAPFAC_p$ factors times the exposure dependent LHGR limits.

APPLICABILITY

The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 25% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ 25% RTP.

(continued)

BASES (continued)

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to restore the LHGR(s) to within its required limit(s) such that the plant is operating within analyzed conditions and within the design limits of the fuel rods. The 2 hour Completion Time is normally sufficient to restore the LHGR(s) to within its limit and is acceptable based on the low probability of a transient or Design Basis Accident 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 < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

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

With regard to LHGR values obtained pursuant to this SR, as determined from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 4).

REFERENCES

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel, GESTAR-II," (latest approved revision).
2. USAR, Section 15.0.
3. NUREG-0800, "Standard Review Plan," Section 4.2, II.A.2(g), Revision 2, July 1981.

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BASES (continued)

References (continued)

4. Calculation IP-0-0002.

5. NEDO-30130-P-A, "Steady State Nuclear Methods," April 1985.

6. NEDO-24154-A, "Qualification of the One Dimensional Core Transient Model for Boiling Water Reactors," August 1986.

BASES (continued)

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 15.2.8.
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, Table 6.3-8.
14. USAR, Section 7.3.1.1.1.4.
15. NEDO-32291-A, "System Analyses for Elimination of Selected Response Time Testing Requirements," January 1994.
16. Calculation IP-0-0044.
17. Calculations 01HP09/10/11, IP-C-0042.
18. Calculations 01LP08/11/14, IP-C-0043.
19. Calculations 01RH19/20/23/24, IP-C-0041.
20. Calculation IP-0-0024.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.3.3 and SR 3.5.3.4

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow is tested both at the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Since the required reactor steam pressure must be available to perform SR 3.5.3.3 and SR 3.5.3.4, sufficient time is allowed after adequate pressure and flow are achieved to perform these SRs. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time to satisfactorily perform the Surveillance is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure test has been satisfactorily completed and there is no indication or reason to believe that RCIC is inoperable. Therefore, these SRs are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test.

A 92 day Frequency for SR 3.5.3.3 is consistent with the Inservice Testing Program requirements. The 18 month Frequency for SR 3.5.3.4 is based on the need to perform this Surveillance under the conditions that apply just prior to or during startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

With regard to RCIC steam supply pressure values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 5).

With regard to the flow rate values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is not considered to be a nominal value with respect to instrument uncertainties. This requires additional margin to be added to the limit to compensate for instrument uncertainties, for implementation in the associated plant procedures. (Ref. 5)

With regard to the measured reactor pressure values used pursuant to this SR, the value obtained as read from plant indication instrumentation is not considered to be a nominal value with respect to instrument uncertainties. This requires additional margin to be added to the acceptance criteria to compensate for instrument uncertainties, for implementation in the associated plant procedures. (Ref. 5)

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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 system. It does not mean to perform the surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two Containment/ Drywell Hydrogen Mixing Systems inoperable for up to 7 days. Seven days is a reasonable time to allow two Containment/ Drywell Hydrogen Mixing Systems 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 amounts capable of exceeding the flammability limit.

C.1

If any Required Action and associated Completion Time cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.3.1

Operating each Containment/Drywell Hydrogen Mixing System ensures that each system is OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, compressor failure, or excessive vibration can be detected for corrective action. The 92 day Frequency is consistent with Inservice Testing Program Frequencies, operating experience, the known reliability of the compressor and controls, and the two redundant subsystems available.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.3.2

Verifying that each Containment/Drywell Hydrogen Mixing System flow rate is ≥ 800 scfm ensures that each system is capable of maintaining drywell hydrogen concentrations below the flammability limit. In practice, verifying that the system differential pressure is less than 4.4 psid with the compressor running ensures that the system flow rate is greater than 800 scfm. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

With regard to system differential pressure values used to verify the required system flow rate as read from plant indication instrumentation, the procedural limit is considered to be not nominal and therefore requires compensation for instrument indication uncertainties (Ref. 3).

REFERENCES

1. Regulatory Guide 1.7.
2. USAR, Section 6.2.5.
3. Calculation IP-0-0076.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.4 and SR 3.6.4.1.5

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.4 verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary. This is confirmed by demonstrating that one SGT subsystem will draw down the secondary containment to ≥ 0.25 inches of vacuum water gauge within the time.

Specifically, the required drawdown time limit is based on ensuring that the SGT system will draw down the secondary containment pressure to ≥ 0.25 inches of vacuum water gauge within 188 seconds under LOCA conditions. Typically, however, the conditions under which drawdown testing is performed pursuant to SR 3.6.4.1.4 are different than those assumed for LOCA conditions. For this reason, and because test results are dependent on or influenced by certain plant and/or atmospheric conditions that may be in effect at the time testing is performed, it is necessary to adjust the test acceptance criteria (i.e., the required drawdown time) to account for such test conditions. Conditions or factors that may impact the test results include wind speed, whether the turbine building ventilation system is running, and whether the containment equipment hatch is open (when the test is performed during plant shutdown/outage conditions). The acceptance criteria for the drawdown test are thus based on a computer model (Ref. 6), verified by actual performance of drawdown tests, in which the drawdown time determined for accident conditions is adjusted to account for performance of the test during normal but certain plant conditions. The test acceptance criteria are specified in the applicable plant test procedure(s). Since the drawdown time is dependent upon secondary containment integrity, the drawdown requirement cannot be met if the secondary containment boundary is not intact.

SR 3.6.4.1.5 demonstrates that each SGT subsystem can maintain ≥ 0.25 inches of vacuum water gauge for 1 hour at a flow rate ≤ 4400 acfm. The 1-hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. Therefore, the tests required per SR 3.6.4.1.4 and SR 3.6.4.1.5 are performed to ensure secondary containment boundary integrity. Since these SRs are secondary containment tests, they need not be performed with each SGT subsystem and an inoperable SGT subsystem does not result in this SR being not met. The SGT subsystems are tested on a STAGGERED TEST BASIS, however, to ensure that in addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. Operating experience has

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SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.4 and SR 3.6.4.1.5 (continued)

shown these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

With regard to drawdown time values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Refs. 4, 5).

REFERENCES

1. USAR, Section 15.6.5.
2. USAR, Section 15.7.4.
3. Calculation IP-0-0082.
4. Calculation IP-0-0083.
5. Calculation IP-0-0084.
6. Calculation 3C10-1079-001.

BASES (continued)

APPLICABILITY

An SVC Protection System must be OPERABLE whenever its associated SVC is in operation, i.e., whenever the SVC's associated offsite circuit is energized with the SVC connected. Although the plant ESF busses are normally aligned together and to either the RAT or ERAT, an SVC Protection System must be OPERABLE if its associated SVC is connected to the associated auxiliary transformer (RAT or ERAT); the transformer is energized by the offsite network; and the transformer is supplying power to at least one ESF bus, or automatic transfer capability to that transformer exists such that it could supply power to at least one ESF bus.

The requirements for the offsite electrical power sources are addressed in LCO 3.8.1, "AC Sources-Operating," and LCO 3.8.2, "AC Sources-Shutdown."

ACTIONS

A.1

With one SVC protection subsystem of a required SVC Protection System inoperable, the inoperable subsystem must be restored to OPERABLE status within 30 days. With the SVC Protection System in this condition, the remaining subsystem is adequate to provide the protection function. However, the overall reliability of the SVC Protection System is reduced because a failure of the OPERABLE subsystem would result in a loss of the SVC failure protection function. The 30-day Completion Time is based on the low probability of an SVC failure occurring during this time period, and the fact that the remaining subsystem can provide the required protection function.

Required Action A.1 is modified by a note that states that the provisions of LCO 3.0.4 are not applicable. This exception allows entry into MODES or other specified conditions in the Applicability when one SVC protection subsystem is inoperable. This exception is acceptable due to the redundancy of the protection systems and the low probability of an SVC failure (fault) that could adversely affect the plant equipment.

B.1

If both SVC protection subsystems of a required SVC Protection System are inoperable, the backup protection system designed for the SVC is unavailable to provide its protection function. Though not all failure modes of the SVC would necessarily be unprotected or potentially damaging to ESF equipment with the required protection system unavailable, there is a significant increase in calculated risk based on conservative failure assumptions for the SVCs. Thus, at least one subsystem must be restored to OPERABLE

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