

1.14 SECONDARY CONTAINMENT INTEGRITY

Secondary containment integrity means that the reactor building is closed and the following conditions are met:

- A. At least one door at each access opening is closed.
(Note: Momentary opening and closing of the trunnion room door does not constitute a loss of secondary containment integrity.)
- B. The standby gas treatment system is operable.
- C. All automatic secondary containment isolation valves are operable or are secured in the closed position.

1.15 (DELETED)

1.16 RATED FLUX

Rated flux is the neutron flux that corresponds to a steady state power level of 1930 MW(t). Use of the term 100 percent also refers to the 1930 thermal megawatt power level.

1.17 REACTOR THERMAL POWER-TO-WATER

Reactor thermal power-to-water is the sum of (1) the instantaneous integral over the entire fuel clad outer surface of the product of heat transfer area increment and position dependent heat flux and (2) the instantaneous rate of energy deposition by neutron and gamma reactions in all the water and core components except fuel rods in the cylindrical volume defined by the active core height and the inner surface of the core shroud.

1.18 PROTECTIVE INSTRUMENTATION LOGIC DEFINITIONS

A. Instrument Channel

An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

B. Trip System

A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system (e.g., initiation of a core spray loop, automatic depressurization, isolation of an isolation condenser, offgas system isolation, reactor building isolation, standby gas treatment and rod block) or the coincident tripping of two trip systems (e.g., initiation of scram, isolation condenser, reactor isolation, and primary containment isolation).

parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluent, in the calculation of gaseous and liquid effluent monitoring Alarm/trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4; and (2) descriptions of the information that should be included in the Annual Radioactive Effluent Release Report AND Annual Radiological Environmental Operating Report required by Specifications 6.9.1.d and 6.9.1.e, respectively.

1.37 PURGE

PURGE OR PURGING is the controlled process of discharging air or gas from a confinement and replacing it with air or gas.

1.38 SITE BOUNDARY

The SITE BOUNDARY is the perimeter line around the OCNGS beyond which the land is neither owned, leased nor otherwise subject to control by AmerGen Energy Company, LLC (ref. ODCM). The area outside the SITE BOUNDARY is termed OFFSITE or UNRESTRICTED AREA.

1.39 REACTOR VESSEL PRESSURE TESTING

System pressure testing required by ASME Code Section XI, Article IWA-5000, including system leakage and hydrostatic test, with reactor vessel completely water solid, core not critical and section 3.2.A satisfied.

1.40 SUBSTANTIVE CHANGES

SUBSTANTIVE CHANGES are those which affect the activities associated with a document or the document's meaning or intent. Example of non-substantive changes are: (1) correcting spelling, (2) adding (but not deleting) sign-off spaces, (3) blocking in notes, cautions, etc, (4) changes in corporate and personnel titles which do not reassign responsibilities and which are not referenced in the Appendix A Technical Specifications, and (5) changes in nomenclature or editorial changes which clearly do not change function, meaning or intent.

1.41 DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 microcuries per gram which alone would produce the same thyroid dose as the quantity and isotopic mixture of I131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table E-7 or Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluences for the Purpose of Evaluating Compliance with 10 CFR Par 40 Appendix I."

SECTION 2

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT - FUEL CLADDING INTEGRITY

Applicability: Applies to the interrelated variables associated with fuel thermal behavior.

Objective: To establish limits on the important thermal hydraulic variables to assure the integrity of the fuel cladding.

Specifications:

- A. When the reactor pressure is greater than or equal to 800 psia and the core flow is greater than or equal to 10% of rated, the existence of a minimum CRITICAL POWER RATIO (MCPR) less than 1.11 for both four or five loop operation and 1.12 for three loop operation shall constitute violation of the fuel cladding integrity safety limit.
- B. When the reactor pressure is less than 800 psia or the core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.
- C. In the event that reactor parameters exceed the limiting safety system settings in Specification 2.3 and a reactor scram is not initiated by the associated protective instrumentation, the reactor shall be brought to, and remain in, the COLD SHUTDOWN CONDITION until an analysis is performed to determine whether the safety limit established in Specification 2.1.A and 2.1.B was exceeded.
- D. During all modes of reactor operation with irradiated fuel in the reactor vessel, the water level shall not be less than 4'8" above the TOP OF ACTIVE FUEL.

Bases:

The fuel cladding integrity safety limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedure used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity safety limit is defined as the CRITICAL POWER RATIO in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR⁽¹⁾ is determined using the General Electric Thermal Analysis Basis, GETAB⁽²⁾, which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

The use of the GEXL correlation is not valid for the critical power calculations at pressures below 800 psia or core flows less than 10% of rated. Therefore, the fuel cladding integrity safety limit is protected by limiting the core thermal power.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psi or core flow less than 10% is conservative.

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 2.1.A or 2.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. Specification 2.1.C requires that appropriate analysis be performed to verify that backup protective instrumentation has prevented exceeding the fuel cladding integrity safety limit prior to resumption of POWER OPERATION. The concept of not approaching a Safety Limit provided scram signals are OPERABLE is supported by the extensive plant safety analysis.

If reactor water level should drop below the TOP OF ACTIVE FUEL, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. With a water level above the TOP OF ACTIVE FUEL, adequate cooling is maintained and the decay heat can easily be accommodated. It should be noted that during power generation there is no clearly defined water level inside the shroud and what actually exists is a mixture level. This mixture begins within the active fuel region and extends up through the moisture separators. For the purpose of this specification water level is defined to include mixture level during power operations.

The lowest point at which the water level can presently be monitored is 4'8" above the TOP OF ACTIVE FUEL. Although the lowest reactor water level limit which ensures adequate core cooling is the TOP OF ACTIVE FUEL, the safety limit has been conservatively established at 4'8" above the TOP OF ACTIVE FUEL.

REFERENCES

- (1) NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR II) (latest approved version as specified in the COLR) |
- (2) General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, NEDO-10958-A, January 1977. |

2.2 SAFETY LIMIT - REACTOR COOLANT SYSTEM PRESSURE

Applicability: Applies to the limit on reactor coolant system pressure.

Objective: Preserve the integrity of the reactor coolant system.

Specification: The reactor coolant system pressure shall not exceed 1375 psig whenever irradiated fuel is in the reactor vessel.

Bases:

The reactor coolant system(1) represents an important barrier in the prevention of the uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1375 psig was derived from the design pressures of the reactor pressure vessel, coolant piping, and isolation condenser. The respective design pressures are 1250 psig at 575°F, 1200 psig at 570°F and 1250 psig at 575°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code Section I for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III for the isolation condenser and the ASA Piping Code Section B31.1 for the reactor coolant system piping. The ASME Code permits pressure transients up to 10% over the design pressure ($110\% \times 1250 = 1375$ psig) and the ASA Code permits pressure transients up to 15% over the design pressure ($115\% \times 1200 = 1380$ psig).

The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 20,000 psi at an internal pressure of 1250 psig and temperature of 575°F; this is more than a factor of 2 below the yield strength of 42,300 psi at this temperature. At the pressure limit of 1375 psig, the general membrane stress increases to 22,000 psi, still almost a factor of 2 below the yield strength. The reactor coolant system piping provides a comparable margin of protection at the established pressure safety limit.

The normal operating pressure of the reactor coolant system is 1020 psig. An overpressurization analysis (2) is performed each cycle to assure that the pressure safety limit is not exceeded. The reactor fuel cladding can withstand pressures up to the safety limit, 1375 psig, without collapsing. (3) Finally, reactor system pressure is continuously monitored in the control room during reactor operation.

REFERENCES

- (1) FDSAR, Volume I, Section IV.
- (2) NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR II) (latest approved version as specified in the COLR).
- (3) FDSAR, Volume I, Section III-2.3.3

The solution saturation temperature varies with the concentration of sodium pentaborate. The solution will be maintained at least 5°F above the saturation temperature to guard against precipitation. The 5°F margin is included in Figure 3.2-2. Temperature and liquid level alarms for the system are annunciated in the control room.

The acceptable time out of service for a standby liquid control system pumping circuit as well as other safety features is determined to be 10 days. However, the allotted time out of service for a standby liquid control system pumping circuit is conservatively set at 7 days in the specification. Systems are designed with redundancy to increase their availability and to provide backup if one of the components is temporarily out of service.

During each fuel cycle, excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity is indicated by the integrated worth of control rods inserted into the core, referred to as the control rod inventory in the core. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of actual rod inventory with expected inventory based on appropriately corrected past data. Experience at Oyster Creek and other operating BWR's indicates that the control rod inventory should be predictable to the equivalent of one percent in reactivity. Deviations beyond this magnitude would not be expected and would require thorough evaluation. One percent reactivity limit is considered safe since an insertion of this reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

References:

- (1) FDSAR, Volume I, Section III-5.3.1
- (2) FDSAR, Volume I, Section VI-3
- (3) FDSAR, Volume I, Section III-5.2.1
- (4) FDSAR, Volume I, Section VII-9
- (5) NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR II) (latest approved version as specified in the COLR).
- (6) FDSAR, Volume I, Section III-5 and Volume II, Appendix B
- (7) FDSAR, Volume I, Sections VII-4.2.2 and VII-4.3.1
- (8) FDSAR, Volume I, Section VI-4
- (9) FDSAR, Amendment No. 55, Section 2
- (10) C. J. Paone, Banked Position Withdrawal Sequence, January 1988 (NEDO-21231)
- (11) UFSAR, Volume 4, Section 4.3.2.4.1

6. With one standby gas treatment system circuit inoperable:

a. During Power Operation:

- (1) Verify the operability of the other standby gas treatment system circuit within 2 hours. If testing is required to demonstrate operability and significant painting, fire, or chemical release has taken place in the reactor building within the previous 12 hours, then demonstration by testing shall take place within 1 hour of the expiration of the 12 hour period, and
- (2) Continue to verify the operability of the standby gas treatment system circuit once per 24 hours until the inoperable standby gas treatment circuit is returned to operable status.
- (3) Restore the inoperable standby gas treatment circuit to operable status within 7 days.

b. During Refueling:

- (1) Verify the operability of the other standby gas treatment system within 2 hours. If testing is required to demonstrate operability and significant painting, fire, or chemical release has taken place in the reactor building within the previous 12 hours, then demonstration by testing shall take place within 1 hour of the expiration of the 12 hour period, and
- (2) Continue to verify the operability of the redundant standby gas treatment system once per 7 days until the inoperable system is returned to operable status.
- (3) Restore the inoperable standby gas treatment system to operable status within 30 days or cease all spent fuel handling, core alterations or operation that could reduce the shutdown margin (excluding reactor coolant temperature changes);

7. If Specifications 3.5.B.5 and 3.5.B.6 are not met, reactor shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours and the condition of Specification 3.5.B.1 shall be met.

3.12 Alternate Shutdown Monitoring Instrumentation

Applicability: Applies to the operating status of alternate shutdown monitoring instrumentation.

Objective: To assure the operability of the alternate shutdown monitoring instrumentation.

Specification:

- A. The alternate shutdown monitoring instruments listed in Table 3.12-1 shall be operable during reactor power operations and when reactor coolant temperature exceeds 212°F.
- B. With less than the minimum number of operable channels specified in Table 3.12-1, either restore the inoperable channel to operable status within 30 days, or be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

Basis:

The operability of the alternate shutdown monitoring instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of hot shutdown of the plant from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with Appendix R and General Design Criteria 19 of 10 CFR 50.

- c. GPUN TR 033, Methods for the Generation of Core Kinetics Data for RETRAN-02, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)
 - d. GPUN TR 040, Steady-State and Quasi-Steady-State Methods Used in the Analysis of Accidents and Transients, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)
 - e. GPUN TR 045, BWR-2 Transient Analysis Model Using the Retran Code, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)
 - f. NEDE-31462P and NEDE-31462, Oyster Creek Nuclear Generating Station SAFER/CORECOOL/GESTR-LOCA Loss-of-Coolant Accident Analysis, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)
 - g. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, (GESTAR II) (The approved revision at the time reload analyses are performed shall be identified in the COLR.)
 - h. DELETED
 - i. XN-75-55-(A); XN-75-55, Supplement 1-(A); XN-75-55, Supplement 2-(A), Revision 2, "Exxon Nuclear Company WREM-Based NJP-BWR ECCS Evaluation Model and Application to the Oyster Creek Plant," April 1977
 - j. XN-75-36(NP)-(A); XN-75-36(NP), Supplement 1-(A), "Spray Cooling Heat Transfer Phase Test Results, ENC- 8x8 BWR Fuel 60 and 63 Active Rods, Interim Report," October 1975
3. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.
 4. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

Basis: 6.9.1.e - RELOCATED TO THE ODCM.

6.9.2 REPORTABLE EVENTS

The submittal of Licensee Event Reports shall be accomplished in accordance with the requirements set forth in 10 CFR 50.73.