PURGE - PURGING

1.23 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~2775 MWt.~~ 2900 Mwt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of hghest reactivity worth which is assumed to be fully withdrawn.

SLAVE RELAY TEST

1.29 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

1.30 Not Used

SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

PURGE - PURGING

1.23 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2900 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SLAVE RELAY TEST

1.29 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

1.30 Not Used

SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: LOWER SHELL
INITIAL RT_{NDT}: 10°F
ART AFTER ~~14~~ 13 EFPY: 1/4T, 96°F
3/4T, 83°F

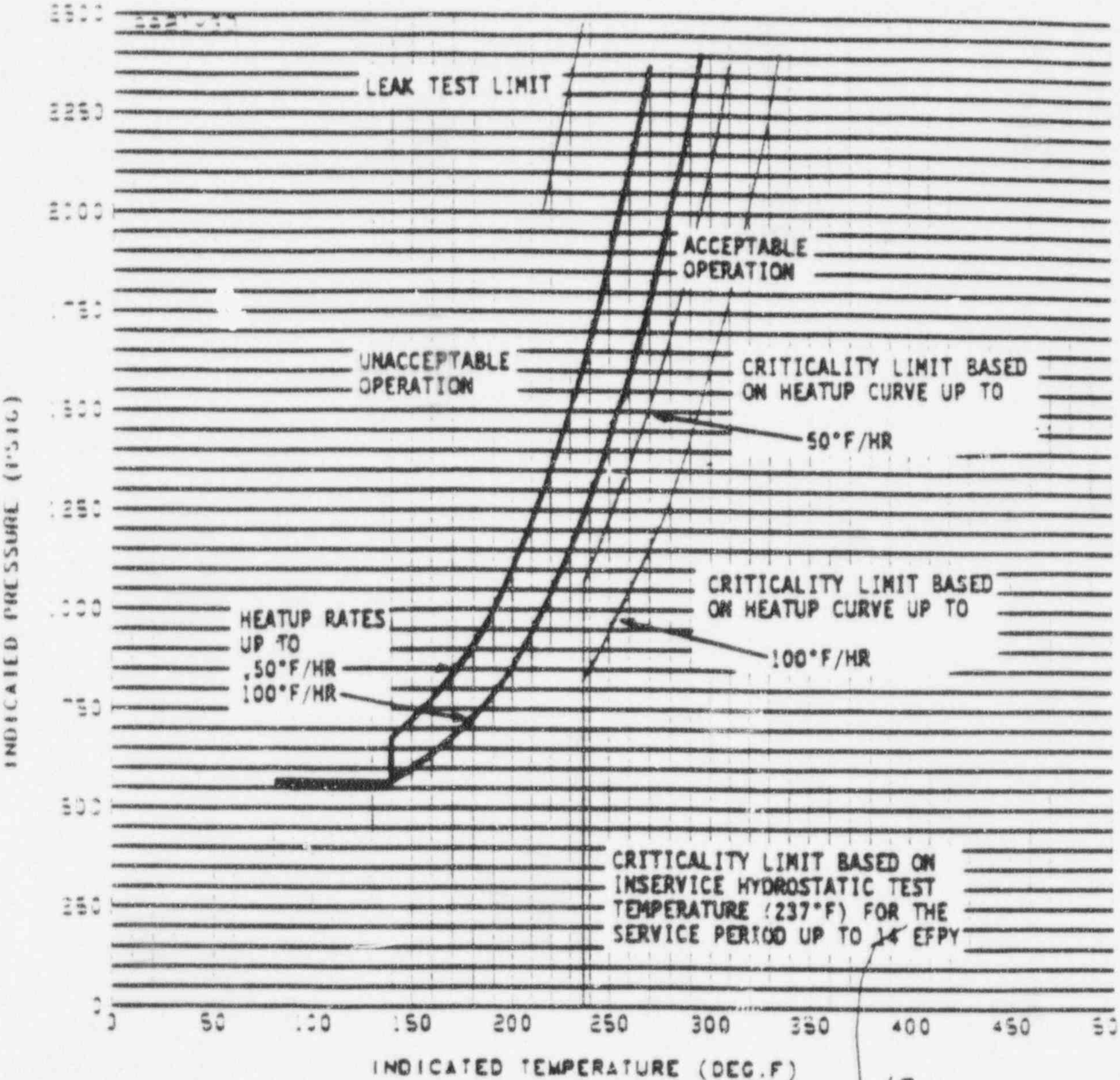

Figure 3.4-2 V. C. Summer Unit 1 Reactor Coolant System Heatup Limitations (Heatup rates up to 50 and 100°F/hr) Applicable for the First ~~14~~ EFPY (With Margins 10°F and 60 psig For Instrumentation Errors)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: LOWER SHELL
INITIAL RT_{NDT}: 10°F
ART AFTER 13 EFPY: 1/4T, 96°F
3/4T, 83°F

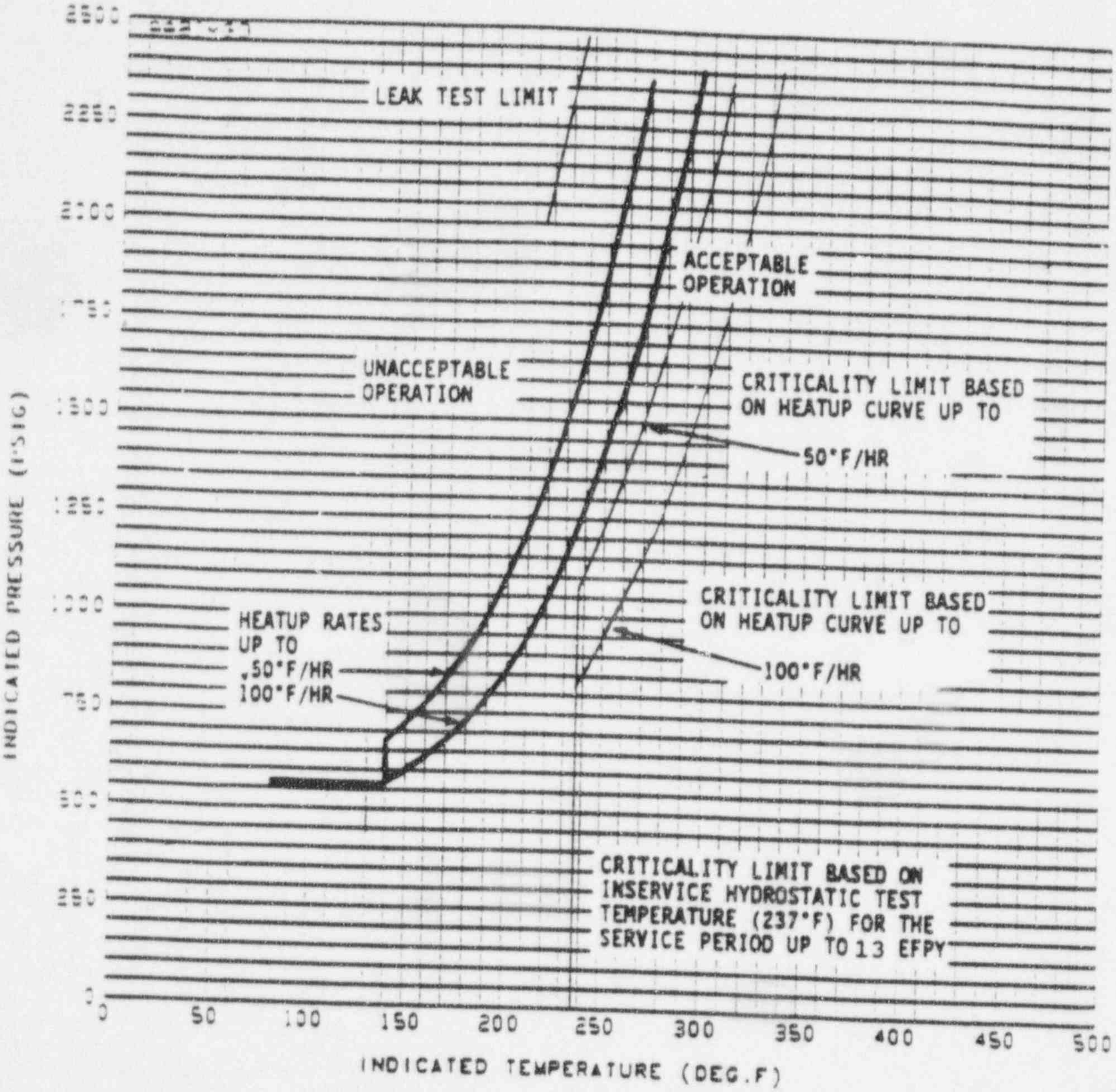

Figure 3.4-2 V. C. Summer Unit 1 Reactor Coolant System Heatup Limitations (Heat up rates up to 50 and 100°F/hr) Applicable for the First 13 EFPY (With Margins 10°F and 60 psig For Instrumentation Errors)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: LOWER SHELL

INITIAL RT_{NDT}: 10°F

ART AFTER ~~14~~ 13 EFPY: 1/4T, 96°F

3/4T, 83°F

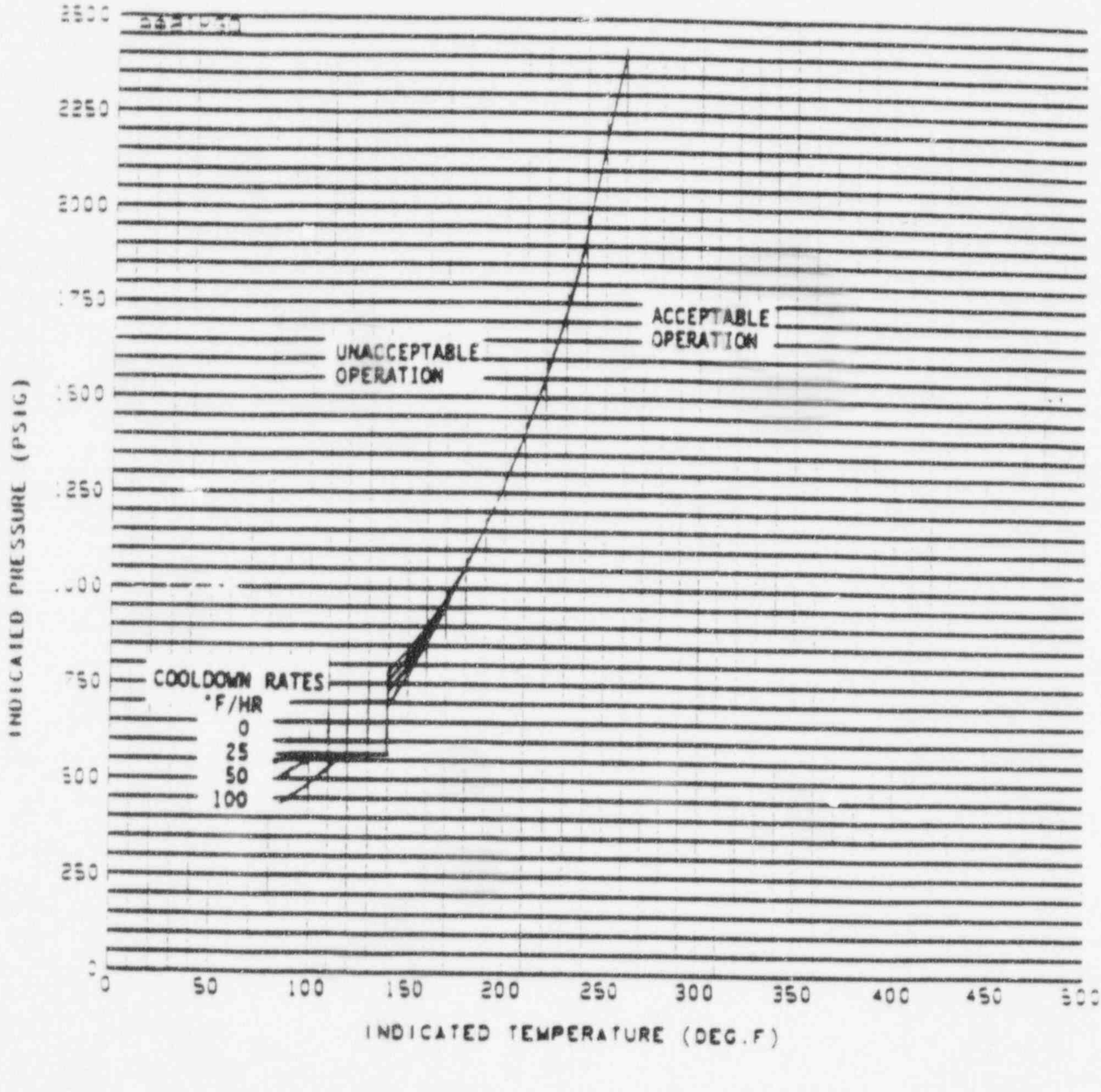

Figure 3.4-3 V. C. Summer Unit 1 Reactor Coolant System Cooldown (Cooldown rates up to 100°F/hr) Limitations Applicable for the First 13 ~~14~~ EFPY (With Margins 10°F and 60 psig For Instrumentation Errors)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: LOWER SHELL
INITIAL RT_{NDT}: 10°F
ART AFTER 13 EFPY: 1/4T, 96°F
3/4T, 83°F

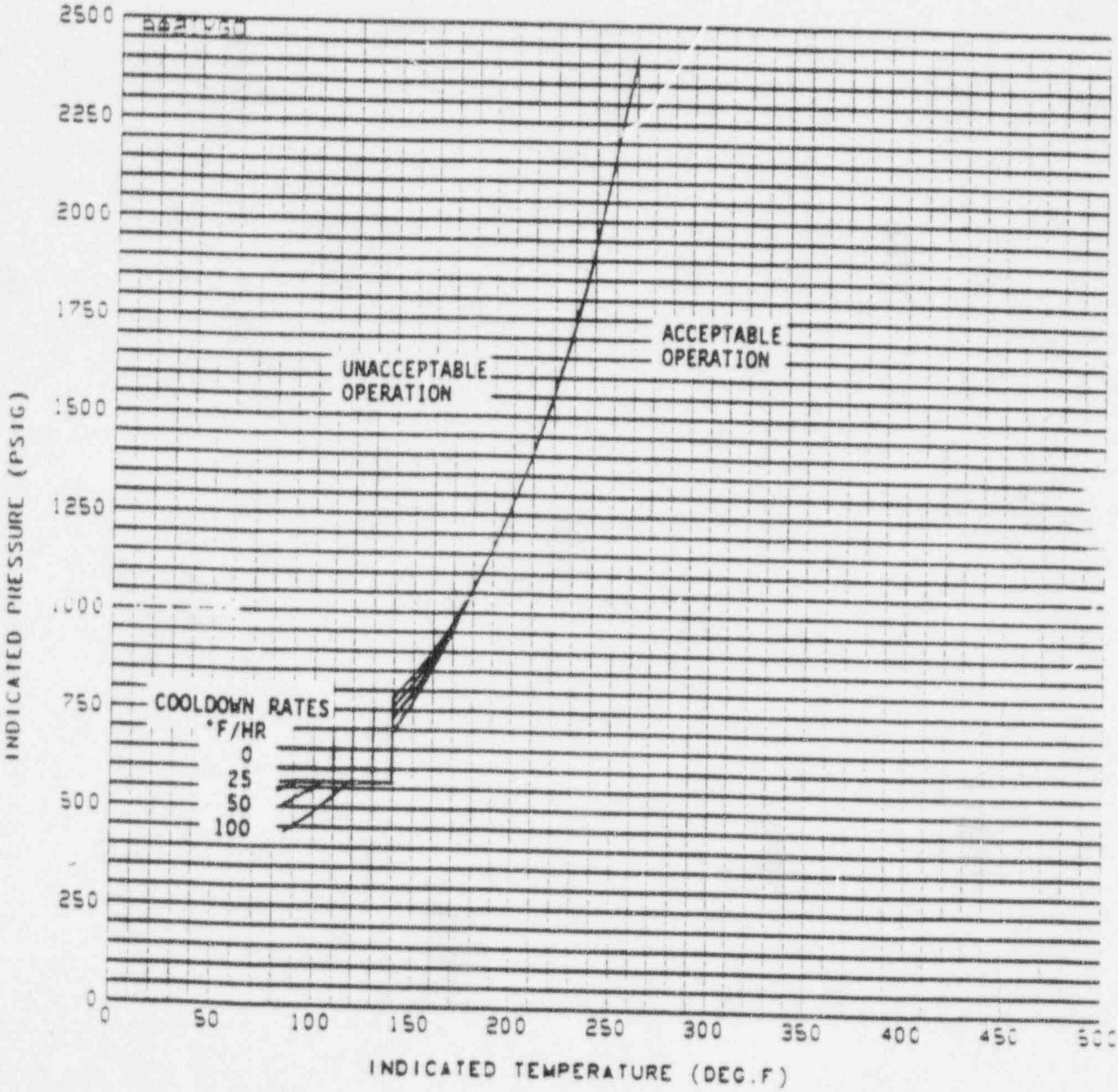

Figure 3.4-3 V. C. Summer Unit 1 Reactor Coolant System Cooldown (Cooldown rates up to 100°F/hr) Limitations Applicable for the First 13 EFPY (With Margins 10°F and 60 psig For Instrumentation Errors)

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to ~~160,000~~ 131,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 131,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least one per 24 hours when radioactive materials are being added to the tank.

CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limit, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

b. WCAP-10216-P-A, Rev. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", February 1994 (W Proprietary).

(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (FQ Methodology for W(Z) surveillance requirements).)

c. WCAP-10266-P-A, Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements there to shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

Including Addendum 2-A, "BASH METHODOLOGY IMPROVEMENTS AND RELIABILITY ENHANCEMENTS," MAY 1988, CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limit, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

b. WCAP-10216-P-A, Rev. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", February 1994 (W Proprietary).

(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (FQ Methodology for W(Z) surveillance requirements).)

c. WCAP-10266-P-A, Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987; Including Addendum 2-A, "BASH METHODOLOGY IMPROVEMENTS AND RELIABILITY ENHANCEMENTS," MAY 1988, (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements there to shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SAFETY EVALUATION
FOR REVISING THE SPECIFICATION FOR
UPRATE
VIRGIL C. SUMMER NUCLEAR STATION
TECHNICAL SPECIFICATIONS

Description of Amendment Request

South Carolina Electric & Gas Company (SCE&G) proposes to revise the following Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) pages: 1-5, 3/4 4-31, 3/4 4-32, 3/4.11-5, and 6-16a. These changes support the Uprate project and provide the following:

- a new definition of Rated Thermal Power (RTP) to incorporate the uprate power condition of 2900 MWt. This value represents the total heat transfer rate from the reactor core to the reactor coolant and does not include heat generated by the reactor coolant pumps.
- a revised limit for the quantity of radioactivity stored in any one gas storage tank. This new value is based on the methodology in NUREG 0133 and only affects the maximum quantity stored.
- a new reference to the Core Operating Limits Report (COLR) which is based on the BASH/BART methodology for Large Break Loss of Coolant Accident analysis.
- revision to the Pressure Temperature Limitations Curves due to effects of increased neutron fluence at 2900 MWt.

Many TS changes were required to support the Steam Generator Replacement (SGR), which were approved and issued via reference 1. Many of the TS changes expected for a plant Uprate were included in the SGR submittal. Most evaluations performed for SGR utilized 2900 MWt core power as an initial condition.

This Technical Specification Change Request (TSCR) primarily revises those areas in TS which were not included in Reference 3. The primary supporting analyses performed for uprate are: Large Break Loss of Cooling Accident (LOCA) utilizing the Westinghouse 1981 Evaluation Model with BASH, spent fuel pool cooling capacity analysis resulting from our outage practices, and Waste Gas Decay Tank Rupture analysis resulting from a comment included in the SER for SGR (Reference 1.). Other analyses and evaluations were performed to assess the capability of other systems and components to support Uprate, with the results indicating that both the Nuclear Steam Supply System (NSSS) and the Balance of Plant systems are capable of supporting uprate power operation assuming modifications to several balance of plant systems.

Increased neutron fluence resulting from uprate core conditions has an effect on the reactor vessel Pressure Temperature Curves. Their applicability will change from 14 Effective Full Power Years (EFPY) to 13 EFPY with no other changes at this time.

Safety Evaluation

The conditions that result from uprate power are increased heat transferred from the Reactor core, increased steam flow, increased feedwater flow, and increased electrical output. The additional heat load of approximately 4.5 percent can be met with the existing capacities of all NSSS and interfacing systems.

Modifications such as Closed Cycle Cooling are being planned to improve the capability of secondary systems to meet the additional load.

The increase in the secondary mass flow rates has been evaluated and does not present any concerns. The Δ75 steam generators are rated for this condition and comply with all ASME Code requirements. The condenser, piping, and valves have all been evaluated and have adequate margin to support uprate conditions. The same is true for Feedwater and Emergency Feedwater Systems. In addition to the code requirements, chrome-moly steel has been used in feedwater piping replaced during RF-8 to reduce the effects of erosion/corrosion.

The additional heat produced will generate additional electricity. The turbine-generator has been evaluated and is capable, with a modification to the Stator Water Cooling System to adequately meet the demands of uprate.

With a RATED CORE POWER level of 2900 MWt, the calculated results (i.e., DNBR, Pressure, Peak Clad Temperature, Metal Water Reaction, Environmental Conditions Inside and Outside Containment, etc.) are acceptable and remain within applicable regulatory acceptance criteria. The results further show that the integrity of the primary/secondary/containment pressure boundary is not challenged and that the extent of fuel failures during Condition III and IV events remains bounded by assumptions within the dose analyses. The calculated radiological consequences remain well within applicable regulatory limits.

Offsite Dose Limits will be maintained with the revision to the gas storage specification. Although this is not specifically an uprate concern, it affects the radiological consequences section in the SGR submittal (Ref. 3). The TS 3.11.2.6 limit will decrease from 160,000 curies Noble Gas to 131,000 curies Noble Gas. However, the station administrative limit of 90,000 curies Noble Gas is unchanged and has never been exceeded. These gas tanks are sampled daily when adding to the tank to assure this limit is not exceeded.

The uprate conditions will produce additional heat loads on the Spent Fuel Cooling System due to increased decay heat. Analyses indicate that the system has sufficient capacity to limit the pool temperature to less than 150°F during limiting Normal heat loads and to less than bulk boiling during limiting Abnormal heat loads. In the event of a loss of spent fuel cooling, adequate time remains available to restore spent fuel cooling to preclude the onset of boiling. For the postulated condition of an extended loss of normal cooling, various makeup water sources are available on site with sufficient capacity to match the pool boiloff rate, thus precluding fuel uncovery.

The Pressure Temperature Limitations Curves are derived using NRC Approved Methodology to comply with 10 CFR 50, Appendix G. These curves provide an acceptable range of operating temperatures and pressures for heatup, cooldown, low temperature overpressure, criticality, and inservice leak and hydrostatic testing conditions. The reduction in applicability for these curves has no effect on the curves themselves. Only the amount of time between the next scheduled specimen capusle analysis and the next revision to these curves will be effected.

Uprate power will not adversely affect the operation of the Reactor Protection System, Engineering Safety Features, or other systems or components that are required for accident mitigation. The revised operating conditions will not affect these systems' performance or qualification for either normal operation or accident conditions. The calculated results to VCSNS FSAR Chapter 15 Analyses demonstrate that there are no challenges to the integrity of the primary/secondary/containment pressure boundaries and that the plant remains within the regulatory acceptance criteria applied to the VCSNS current licensing basis.

SIGNIFICANT HAZARDS EVALUATION FOR REVISING THE SPECIFICATION FOR UPRATE VIRGIL C. SUMMER NUCLEAR STATION TECHNICAL SPECIFICATIONS

Description of Amendment Request

South Carolina Electric & Gas Company (SCE&G) proposes to revise the following Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) pages: 1-5, 3/4 4-31, 3/4 4-32, 3/4.11-5, and 6-16a. These changes support the Uprate project and provide the following:

- a new definition of Rated Thermal Power (RTP) to incorporate the uprate power condition of 2900 MWt. This value represents the total heat transfer rate from the reactor core to the reactor coolant and does not include heat generated by the reactor coolant pumps.

- a revised limit for the quantity of radioactivity stored in any one gas storage tank. This new value is based on the methodology in NUREG 0133 and only affects the maximum quantity stored.

- a new reference to the Core Operating Limits Report (COLR) which is based on the BASH/BART methodology for Large Break Loss of Coolant Accident analysis.

- revision to the Pressure Temperature Limitations Curves due to effects of increased neutron fluence at 2900 MWt.

Many TS changes were required to support the Steam Generator Replacement (SGR), which were approved and issued via reference 1. Many of the TS changes expected for a plant Uprate were included in the SGR submittal. Most evaluations performed for SGR utilized 2900 MWt core power as an initial condition.

This Technical Specification Change Request (TSCR) primarily revises those areas in TS which were not included in Reference 3. The primary supporting analyses performed for uprate are: Large Break Loss of Cooling Accident (LOCA) utilizing the Westinghouse 1981 Evaluation Model with BASH, spent fuel pool cooling capacity analysis resulting from our outage practices, and Waste Gas Decay Tank Rupture analysis resulting from a comment included in the SER for SGR (Reference 1.). Other analyses and evaluations were performed to assess the capability of other systems and components to support Uprate, with the results indicating that both the Nuclear Steam Supply System (NSSS) and the Balance of Plant systems are capable of supporting uprate power operation assuming modifications to several balance of plant systems.

Increased neutron fluence resulting from uprate core conditions has an effect on the reactor vessel Pressure Temperature Curves. Their applicability will change from 14 Effective Full Power Years (EFPY) to 13 EFPY with no other changes at this time.

Basis for No Significant Hazards Consideration Determination

South Carolina Electric & Gas Company (SCE&G) has evaluated the proposed changes to the VCSNS TS described above against the Significant Hazards Criteria of 10 CFR 50.92 and has determined that the changes do not involve any significant hazard for the following reasons:

1. The probability or consequences of an accident previously evaluated is not significantly increased.

   Implementation of uprate power operation does not contribute to any accident evaluated in the FSAR. The NSSS Components (RV, RCPs, CRDMs, SGs, and piping) are compatible with the revised operating conditions. These components have been reanalyzed and the results show that ASME Code requirements remain satisfied and are within the current Licensing Basis.

   Interfacing Systems which are important to safety are not adversely impacted and will continue to perform their design function. Overall secondary plant performance is not significantly altered by the proposed changes.

   The revision to the Pressure Temperature Limits will not adversely impact the RCS Pressure Boundary. The length of time these curves will be applicable, due to increased neutron fluence, is being reduced. Before the 13 Effective Full Power Years have elapsed, new curves will be generated to reflect the analysis of the specimen capsule and will be derived utilizing NRC approved methodology.

   Therefore, since the Reactor Coolant pressure boundary integrity and system functions are not adversely impacted, the probability of occurrence of an accident evaluated in the VCSNS FSAR will be no greater than the original design basis of the plant.

   An extensive analysis has been performed to evaluate the consequences of the following accident types currently evaluated in the VCSNS FSAR:

   - Non-LOCA Events
   - Large Break and Small Break LOCA
   - Steam Generator Tube Rupture

   With the Δ75 SGs and revised operating conditions, the calculated results (i.e., DNBR, Primary and Secondary System Pressure, Peak Clad Temperature, Metal Water Reaction, Challenge to Long Term Cooling, Environmental Conditions Inside and Outside containment, etc.) for the accidents are similar to those currently reported in the VCSNS FSAR and remain within applicable Regulatory Acceptance Criteria. Select results (i.e., Containment Pressure during a Steam Line Break, Minimum DNBR for Rod Withdrawal from Subcritical, etc.) are slightly more limiting than those currently reported in the FSAR due to the use of the assumed operating conditions with the Δ75 SGs and in some cases, use of an uprated core power of 2900 MWt. However, in all cases, the calculated results do not challenge the integrity of the primary/secondary/ containment pressure boundary and remain within the regulatory acceptance criteria applied to VCSNS's current licensing basis.

Given that calculated radiological consequences are not significantly higher than current FSAR results and remain well within 10CFR100 limits, it is concluded that the consequences of an accident previously evaluated in the FSAR are not significantly increased.

2. The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   Uprate power operation will not introduce any new accident initiator mechanisms. Structural integrity of the RCS is maintained during all plant conditions through compliance with the ASME code and 10 CFR 50 Appendix G requirements. Design requirements of auxiliary systems are met with the RSGs and uprate power operation. No new failure modes or limiting single failures have been identified. Since the safety and design requirements continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, no new accident scenarios have been created. Therefore, the types of accidents defined in the FSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation.

3. The proposed license amendment does not involve a significant reduction in a margin of safety.

   Although uprate power operation will require changes to the VCSNS Technical Specifications, the proposed changes are supported by extensive LOCA, NON-LOCA and SGTR analyses. These analyses show acceptable consequences with margin to the applicable regulatory limits. All equipment required to function during accident conditions has been shown to remain qualified and thus will perform their design function, and all components remain in compliance with the codes and standards in effect when VCSNS was originally licensed (with the exception of the replacement steam generators which use the 1986 ASME Code Section III Edition).

   Low Temperature Overpressure transients which could challenge RCS structural integrity are not impacted by the revision to the Pressure Temperature Limitations Curves. The curves are not directly impacted, the changes do not reduce any margin of safety.

Based on the above, it is concluded that there is no significant reduction in a margin of safety.