(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. are hereby incorporated into the license. The licensee shall operate
the facility in accordance with the Technical Specifications.

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

c. Performance of any test at a power level different from there described; and

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.23 DELETED

PURGE - PURGING

1.24 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

1.25 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.

RATED THERMAL POWER (RTP)

1.26 RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 3455 MWt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its (RTS) trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by NRC.

REPORTABLE EVENT

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

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	62
2	45
3	28

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1194 psig) of the system design pressure during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

In Mode 1 above 28% RTP, the number of MSSVs per steam generator required to be operable must be according to Table 3.7-1 in the accompanying LCO. At or below 28% RTP in Modes 1, 2, and 3, only two MSSVs per steam generator are required to be operable.

In Modes 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in Moαes 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be operable in these modes.

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 1.6×10^7 lbs/hr at 1170 psig which is 106.4 percent of the total secondary steam flow of 1.514×10^7 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

To calculate this setpoint, the governing equation is the relationship $q = m\Delta h$, where q is the heat input from the primary side, m is the steam flow rate and Δh is the heat of vaporization at the steam relief pressure (assuming no subcooled feedwater). Thus, an algorithm for use in defining the revised Technical Specification table setpoint values would be:

$$\text{Hi } \phi = (100 / Q) \frac{(w_s h_{fg} N)}{K}$$

Where:

Hi ϕ = Safety Analysis power range high neutron flux setpoint, percent

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.   are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

c. Performance of any test at a power level different from there described; and

RATED THERMAL POWER (RTP)

1.26 RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 3455 MWt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its (RTS) trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by NRC.

REPORTABLE EVENT

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

SHIELD BUILDING INTEGRITY

1.29 SHIELD BUILDING INTEGRITY shall exist when:

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

b. The emergency gas treatment system is OPERABLE.

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

SHUTDOWN MARGIN

1.30 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.

SITE BOUNDARY

1.31 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee (see figure 5.1-1).

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR
FOR THE SERVICE PERIOD UP TO 14.5 EFPY. MARGINS OF 60 PSIG
AND 10°F ARE INCLUDED FOR POSSIBLE INSTRUMENT ERRORS.

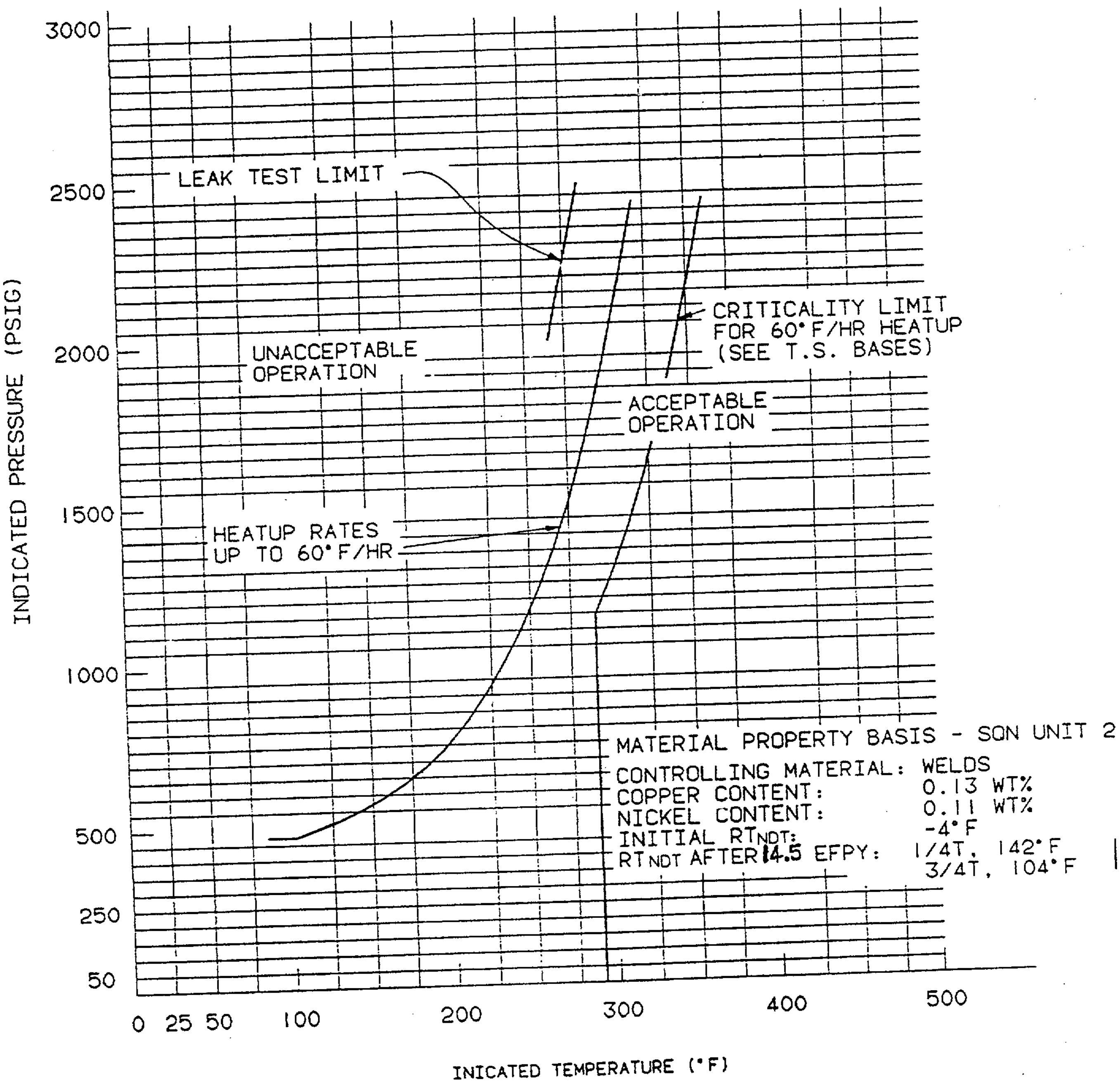

FIGURE 3.4-2 SEQUOYAH UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
APPLICABLE UP TO 14.5 EFPY

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 14.5 EFPY. MARGINS OF 60 PSIG AND 10°F ARE INCLUDED FOR POSSIBLE INSTRUMENT ERRORS.

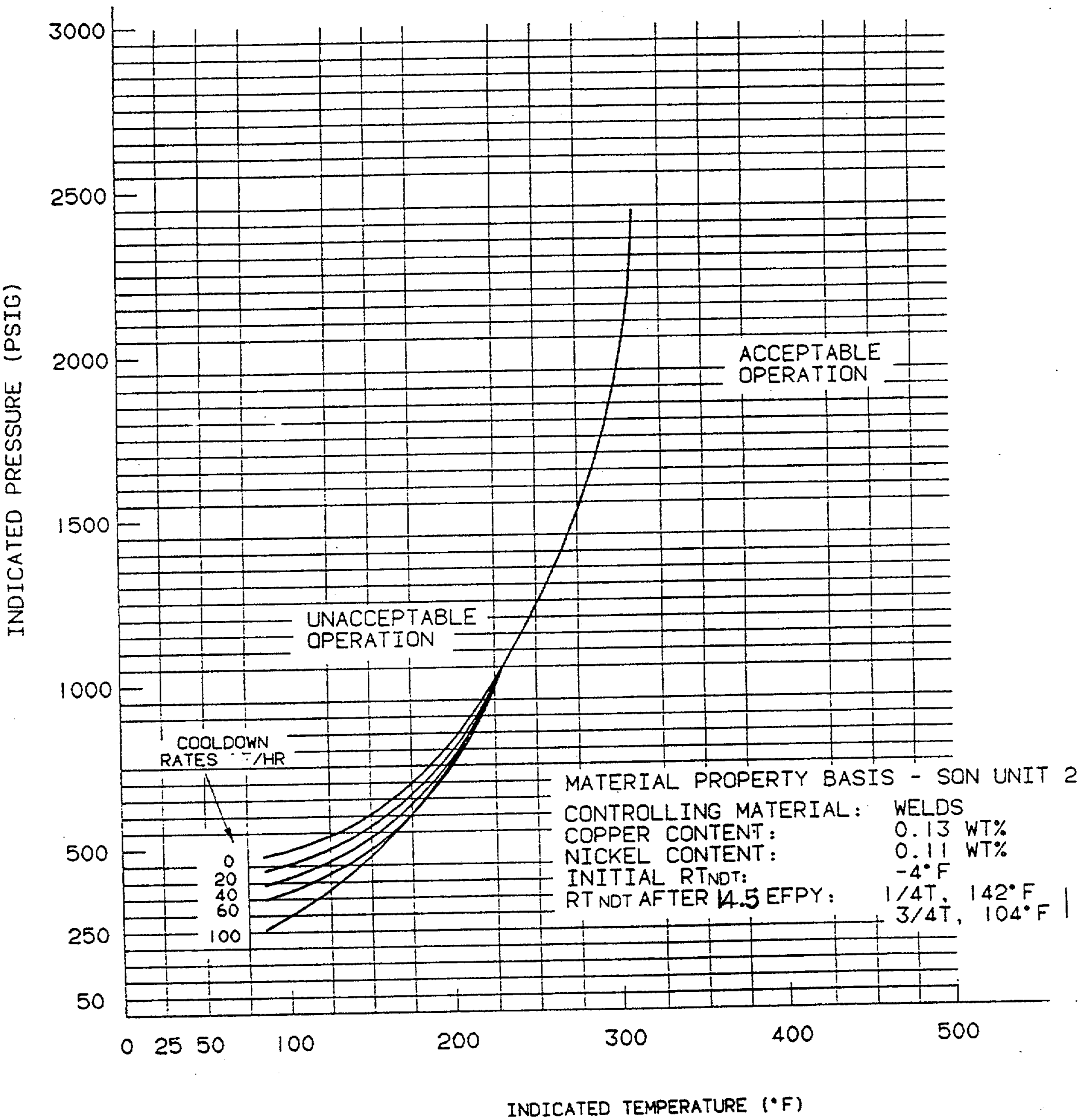

FIGURE 3.4-3 SEQUOYAH UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE UP TO 14.5 EFPY

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	62
2	45
3	28

PRESSURE/TEMPERATURE LIMITS (Continued)

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

10 CFR 50, Appendix G, addresses metal temperature of the closure head flange and vessel regions. Appendix G states that the minimum metal temperature of the closure flange region should be at least 120 degrees Fahrenheit (F) higher than the limiting RT_{NDT} for this region when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (561 pounds per square inch gauge (psig) for Westinghouse Electric Corporation plants). For SQN, Unit 2, the minimum temperature of the closure flange and vessel flange regions is 117 degrees F since the limiting initial RT_{NDT} for the closure head flange is -13 degrees F (see Table B 3/4.4-1). These numbers (561 psig and 117 degrees F) include a margin for instrumentation error of 10 degrees F and 60 psig. The SQN Unit 2 heat up and cooldown curves shown in Figures 3.4-2 and 3.4-3 are not impacted by this regulation.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, and ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G to 10 CFR 50 and Appendix G of the 1986 ASME Boiler and Pressure Vessel Code, Section III, Division 1 and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT} at the end of 16 effective full power years of service life. The 16 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence of the material in question, has been predicted using Regulatory Guide 1.99, Revision 2 and a peak surface fluence of 0.864 x 10^{19} n/cm^2 for 16 effective full power years (Reference WCAP 12971, "Heatup and Cooldown Limit Curves for Normal Operation," June 1991. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 16 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments. The heatup and cooldown limits in WCAP-12971 were based on a core thermal power of 3411 MWt. The curves have been evaluated in WCAP-15725 to be effective for operation through the end of 14.5 EFPY for the uprated core thermal power of 3455 MWt.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1194 psig) of the system design pressure during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

In Mode 1 above 28% RTP, the number of MSSVs per steam generator required to be operable must be according to Table 3.7-1 in the accompanying LCO. At or below 28% RTP in Modes 1, 2, and 3, only two MSSVs per steam generator are required to be operable.

In Modes 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in Modes 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be operable in these modes.

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 1.6×10^7 lbs/hr at 1170 psig which is 106.4 percent of the total secondary steam flow of 1.514×10^7 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

To calculate this setpoint, the governing equation is the relationship q = mΔh, where q is the heat input from the primary side, m is the steam flow rate and Δh is the heat of vaporization at the steam relief pressure (assuming no subcooled feedwater). Thus, an algorithm for use in defining the revised Technical Specification table setpoint values would be:

$$\text{Hi } \Phi = (100/Q)\,\frac{(W_s h_{fg} N)}{K}$$

where:

Hi Φ = Safety Analysis power range high neutron flux setpoint, percent