ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1 - DOCKET NO. 390

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-00-06

CALDON, INC. - ENGINEERING REPORT-160

SUPPLEMENT TO CALDON, INC. TOPICAL REPORT ER-80P

(NON-PROPRIETARY)

ER-160
Revision 0
May 2000

CALDON, INC.

ENGINEERING REPORT-160

Supplement to Topical Report ER-80P:
Basis for a Power Uprate
With the LEFM✓ ™ System

Revision 0

Prepared By: Jennifer Regan
Reviewed By: Herb Estrada, Jr.

Supplement to Caldon Topical Report ER-80P:
Basis For A 1.4% Power Uprate
With The LEFM✓

1. Purpose and Background

On May 3, 2000, the NRC approved a rule change amending 10CFR50 Appendix K to permit power increases based on improvements in accuracy of the instrumentation used to measure thermal power. These power increases, referred to as "Appendix K Uprates", are relatively small increases on the order of 1% to 1.7%, depending on the demonstrated instrument accuracy. The purpose of this supplement is to provide a basis for a 1.4% uprate using Caldon's LEFM✓ system to measure thermal power.

2. Probabilistic Basis for Power Uprate

A power uprate can be obtained based on improved accuracy of the instrumentation used to measure thermal power, in accordance with the Appendix K rule change described above. As shown in Table 1, the LEFM✓ measures thermal power to within ±0.6%. To assess the increase in thermal power rating appropriate to the use of the LEFM✓, this discussion will interpret the meaning of the data of Tables 1 and 2 on a probabilistic basis.

When they developed standards for the measurement of steam turbine heat rate in power plants, the ASME performed a series of Monte Carlo analyses which demonstrated that, if the uncertainty elements of a measurement system are calculated on a 2 standard deviation basis, the uncertainty in the overall measurement that results is characterized by a normal distribution with 2 standard deviations equal to the root sum square of appropriately weighted individual elements (Reference 1). This result held even when the uncertainties of individual elements were not normally distributed. For example, a particular element might be characterized by a "roulette wheel" (flat) distribution between defined uncertainty bounds. It was subject only to one condition: that no single element dominate the calculation of the overall uncertainty.

While it is not obvious, the tabulations in Tables 1 and 2 meet this condition. The profile factor uncertainty of the LEFM✓ in Table 1 appears dominant, but is, in fact, made up of four elements, none of which is dominant. Similarly, the instrumentation allowance in Table 2 appears dominant, but is in fact made up of numerous elements in several instruments. Therefore, the overall uncertainties described in Tables 1 and 2 are likely to be normally distributed. Furthermore, the sensitivity of the results to the nature of the elemental uncertainty distribution has been investigated as described in Reference 2. This investigation shows that the distribution of the total uncertainty is likely to be normal whether the contributors are each normally distributed or distributed in roulette wheel fashion.

Table 1 implies a distribution wherein one standard deviation of LEFM✓ uncertainty is about ±0.3% full power. As shown in Table 3, with this distribution there is essentially no chance (less than one in 3 million) that an operator using the LEFM✓ to determine thermal power will exceed a power level 1.5% above that to which he is controlling. Here the odds have been computed on the basis of 5 standard deviations (Appendix to this Supplement). Similarly, Table 2 implies a normal distribution

of nozzle-based uncertainty with one standard deviation of ±0.7%. As shown in Table 3, the odds of exceeding a power 3.5% above that indicated by the current instrumentation are similarly small. The one sigma value of 0.7% assumed for uncertainty of venturi-based power measurement is regarded by the NRC as representative of the low end of the scale for venturi-based uncertainty. Specifically, the NRC states, "Generally, the single loop uncertainty for thermal power appears to range from 1.8% to over 3% of power when using a venturi to measure feedwater flow based on a review of various Westinghouse PWR plants" (Reference 4).

Table 3. Probabilities and Odds Associated With Nozzle and LEFM Uncertainty Bounds

Number of Standard Deviations	Venturi Nozzle Bounds (±)	LEFM✓ Bounds (±)	Probability of Operation Within Bounds	Odds of Exceeding Bounds on the High Side
1	0.7%	0.3%	68%	1/6.3
2	1.4%	0.6%	95.4%	1/44
3	2.1%	0.9%	99.7%	1/741
4	2.8%	1.2%	99.994%	1/32,300
5	3.5%	1.5%	99.99994%	1/3.3 million

To clarify the basis for a power increase with use of the LEFM✓, the results of Table 3 are shown graphically in Figures 1 through 3. All three figures show power level (as a percent of the pre-uprate 100% power) along the "x" axis, and probability data along the "y" axis. All three figures illustrate both operation with the current instrumentation at the current 100% power level and operation with the LEFM✓ at a 1.4% power increase.

Figure 1 shows the probable operating ranges. As expected, the curves peak at the power level where operation is intended, and fall off symmetrically on either side of the peak. Of greater interest from the standpoint of operating safety is the probability that any given power level will be exceeded, as shown in Figure 2. As Figure 2 shows, the probability of exceeding a given power level is 100%, or a sure thing, just prior to the intended power level. The probability for each case equalizes at 102% power, which is the power level at which most plants' safety systems are analyzed for proper performance. Figure 3 presents the same data as Figure 2, but focuses in the vicinity of 102% power where the probability curves for the LEFM✓ and current instrumentation intersect. Though the intended operating point is higher for the LEFM✓ system due to the power increase, the probability of exceeding 102% power is the same for both instruments. In other words, the probability of exceeding the analyzed power level of 102% is the same for the current instrumentation operating at 100% as for the LEFM✓operating at 101.4%.

Figure 3 also shows another advantage of more accurate power measurements. As power measurement precision increases, the chance of a significant overpower incident decreases. For example, a plant equipped with flow nozzles, intending to operate at 100% of its licensed power, has about a 1 in 100 chance of exceeding 102.3%. On the other hand, the same plant, equipped with the LEFM✓, and intending to operate at 101.4% of its (previous) licensed power, has less than a 1 in 741 chance of exceeding 102.3%. (These odds are based on Table 3.)

There are two assumptions critical to the preceding discussion of thermal power margin. First, the necessity of an uncertainty distribution that is normal has been discussed and, based on the ASME studies and the Appendix, is satisfied. The second is that Tables 1 and 2 *actually describe* the performance of the instruments in service. Verification that the LEFM systems are operating within their design bounds is provided continuously, as mentioned above and discussed in detail in Reference 2. But there is no comparable on-line assurance that current nozzle-based instrumentation

is operating within its design bounds. This is the basis for the conclusion that power increases with LEFM systems increase safety.

3. Benefits of On-Line Verification

To illustrate the benefits of on-line verification, Figure 4 shows the results of a survey of sustained overpower events reported in Licensee Event Reports from 1981 through 1999 (Reference 3). The 61 identified events have been categorized by cause in order to examine whether they would have been preventable with the on-line verification capabilities of LEFM systems. Figure 4 illustrates that the LEFM systems with on-line verification would have prevented all significant sustained overpower events. Looking at the extremes, five cases have been reported in Licensee Event Reports where steady state overpower has occurred in an amount not consistent with the probability predictions implied by Table 3; i.e., operation at 2% or more beyond the licensed power level. The causes for these events are summarized in Table 4.

Table 4. Sustained Overpower Events Above 102% and Their Causes

LER Number	Reported Power Excursion	Reported Duration	Reported Cause of Event
82-002	2.7%	46 days	Differential pressure transmitter found out of tolerance.
87-069	2.1%	2 days	Procedural - nuclear instruments interval and deadband error allowed beyond limit.
88-035	2%-3%	10 days	Hole in venturi pressure tap.
91-012	2.09%	5 years	Core power calculation error; improper density compensation.
94-002	2.6%	8 months	Perimeter bypass flow of venturi feed nozzles.

In three of these cases, the sustained overpower event was the result of the instrumentation system (transmitters or nozzles) failing to operate as designed. The other two cases were due to procedural errors and improper density compensation. The common link in all of these cases is that there was no indication of a problem until an independent means of measurement or calculation was employed. There is currently no indication available to the operators for the accuracy of the thermal power measurement. All of these case would have been prevented by use of LEFM systems, because LEFM systems incorporate on-line verification features and real-time control room displays that prevent occurrences of subtle failure by providing operators continuous information about the measurement, and about the accuracy of the measurement.

It is the LEFM's ability to confirm on-line that it is performing within its accuracy bounds, as well as its high accuracy, that justifies a power uprate with its use. In addition to providing for a power uprate, LEFM systems will assure that the probability of exceeding the analyzed power level (i.e., 1.02 times the current licensed rating) by as little as 0.5% is negligibly small.

4. Using the LEFM✓ to Control Thermal Power

With the existing instrumentation, for each feedwater flow measurement, the differential pressure transmitters provide an output proportional to the differential pressure across the flow nozzle.

Resistance thermometers (or thermocouples) measure the feedwater temperature. Typically, these outputs are supplied to the plant computer where the density and enthalpy are calculated with the aid of synthesized ASME steam tables. The thermal power is then calculated, also by the plant computer.

It is anticipated that a licensee will make use of LEFM mass flow and temperature measurements by directly substituting the LEFM indications for the nozzle-based mass flow indication and the RTD temperature indications in the plant computer. The plant computer would then calculate enthalpy and thermal power as it does now. As an alternative, the calorimetric power can be manually calculated, using LEFM indications and following a prescribed procedure.

While this discussion is focused on operation at full power, it should be noted that LEFM systems provide accurate flow and temperature indications from synchronization to full power. The LEFM✓ may be used for thermal power determinations following synchronization at 10% to 15% power (when feedwater heating commences) and up to full power, with an accuracy better than the present instrumentation.

In order to maintain control of thermal power at 100 percent power, a real-time display of thermal power as calculated using the LEFM will be available in the main control room for the reactor operator's use. The operator will use this display to maintain reactor power at or below the licensed thermal rating, with a tolerance in accordance with current plant practice. The thermal power display will also present, in the same location as the thermal power value, a clear indication of the validity of the thermal power measurement as determined by LEFM diagnostics. For example, an audible alarm will annunciate to the operators when the LEFM is not operating within its design basis accuracy. This indication will be provided by the LEFM's on-line verification system, which is discussed in detail in Reference 2.

5. References

1. ANSI/ASME Power Test Code PTC 19.1 – 1985, Part 1 Measurement Uncertainty, Reaffirmed 1990.
2. Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓ System", Rev. 0.
3. Regan, J.,"Operation Near 100% Rated Thermal Power: Historical Licensee Event Reports", Proceedings of the 1999 ANS Winter Meeting, November 1999.
4. NRC SER dated March 8 1999, "Safety Evaluation by the Office of Nuclear Reactor Regulation Topical Report ER-80P, 'Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM System', Comanche Peak Steam Electric Station, Units 1 and 2 Docket Nos. 50-445 and 50-446"

Figure 1. Probable Operating Ranges for the LEFMCheck System at Increased Power Levels

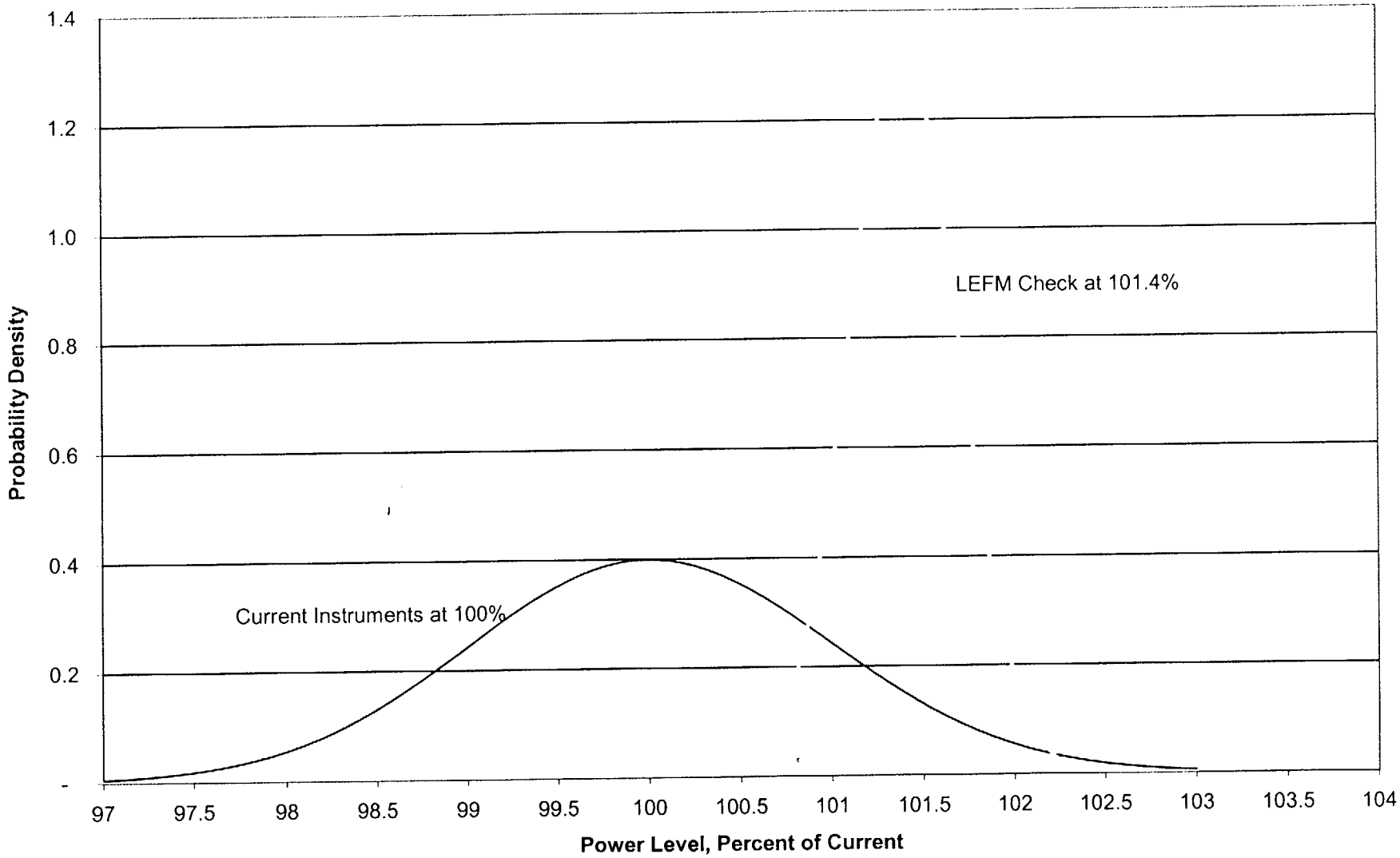


Figure 2. Probability of Exceeding Power Levels With the LEFMCheck System and Increased Power

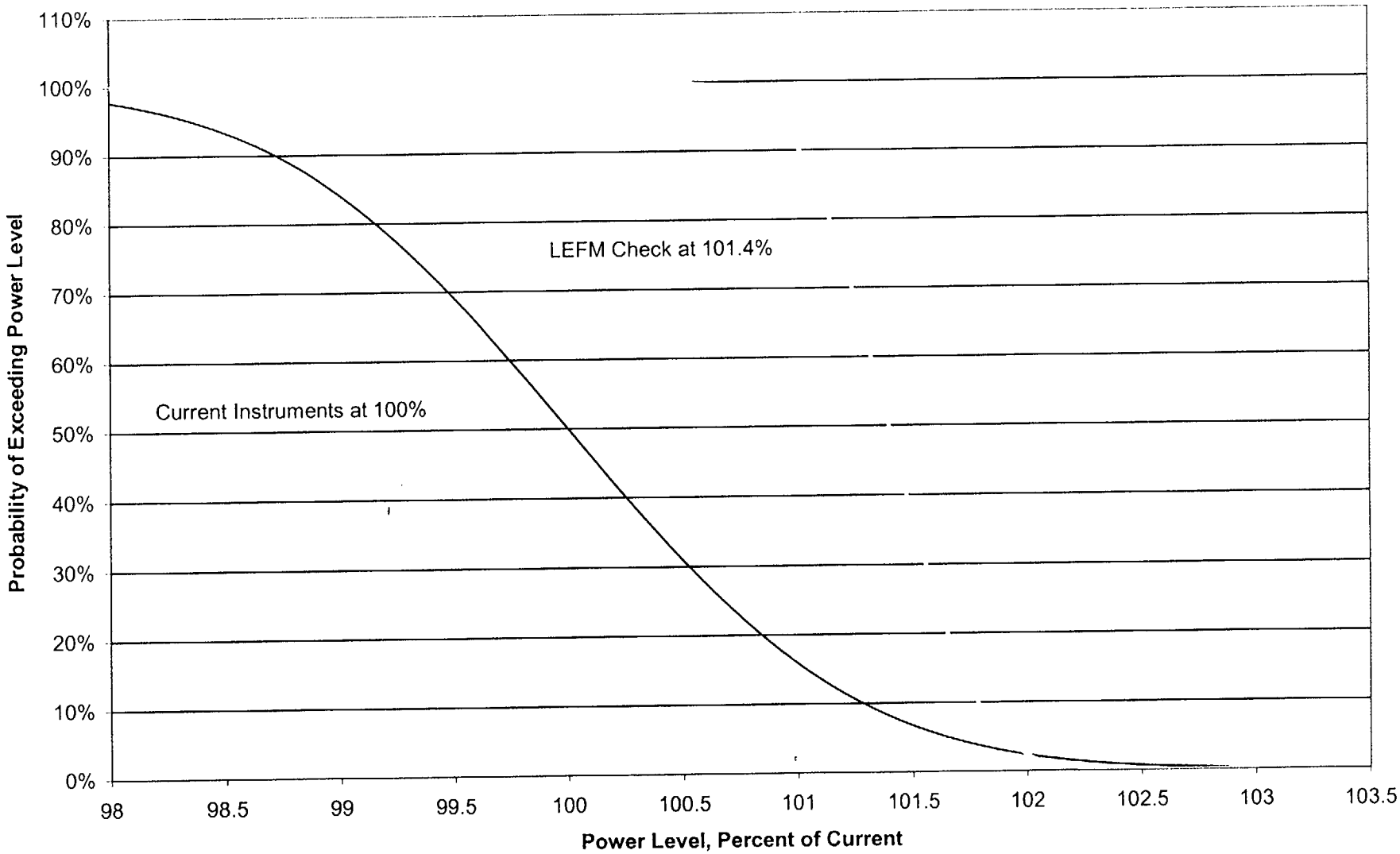
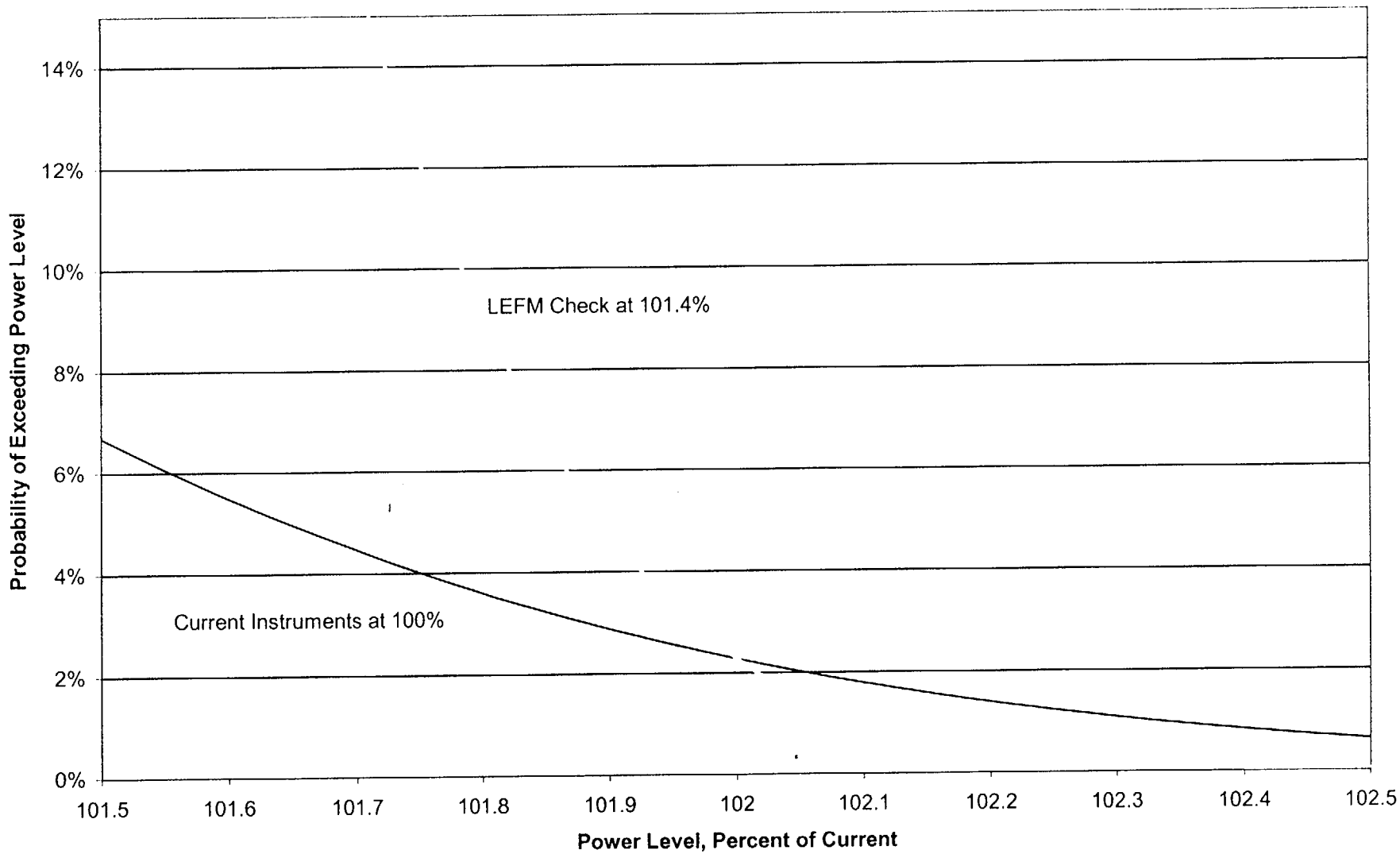


Figure 3. Probability of Exceeding Power Level in the Vicinity of 102% Power



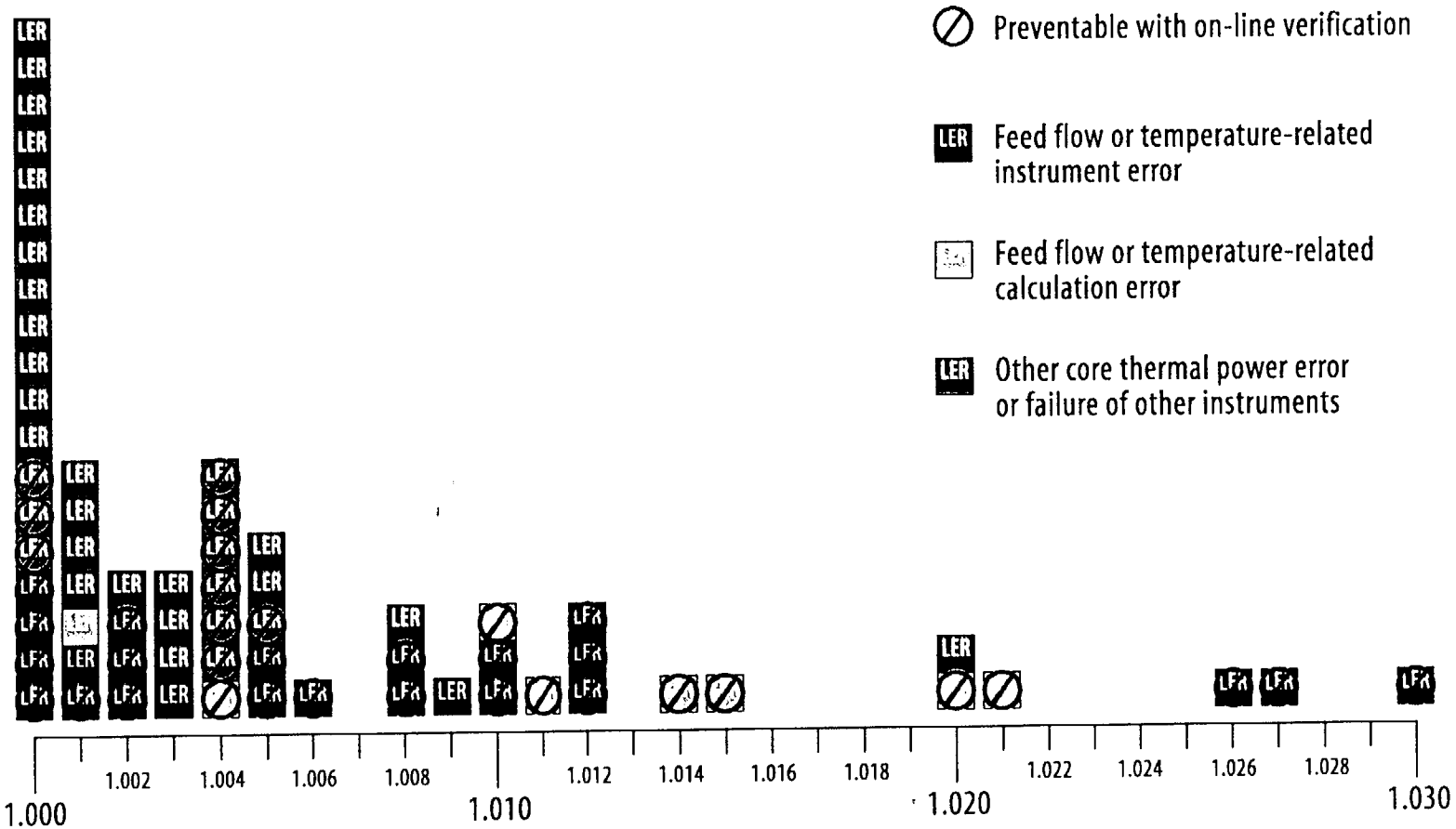


Figure 4. Results of LER Survey 1982 – 1999

Appendix is Proprietary to Caldon in its Entirety

ENCLOSURE 8

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1 - DOCKET NO. 390

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-00-06

MARKED PAGES

I. AFFECTED PAGE LIST

Operating License
Page 3

Technical Specification:
1.1-5
2.0-2
3.7-1
3.7-3
5.0-32

Technical Specification Bases:
B 3.7-3
B 3.7-35

II. MARKED PAGES

ATTACHED

(4) TVA, 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, instrument calibration, or other activity associated with radioactive apparatus or components; and

(5) TVA, 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 facility.

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

TVA is authorized to operate the facility at reactor core power levels not in excess of ~~3411~~ megawatts thermal.

3459

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 23, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Safety Parameter Display System (SPDS) (Section 18.2 of SER Supplements 5 and 15)

Prior to startup following the first refueling outage, TVA shall accomplish the necessary activities, provide acceptable responses, and implement all proposed corrective actions related to having the Watts Bar Unit 1 SPDS operational.

(4) Vehicle Bomb Control Program (Section 13.6.9 of SSER 20)

During the period of the exemption granted in paragraph 2.D.(3) of this license, in implementing the power ascension phase of the approved initial test program, TVA shall not exceed 50% power until the requirements of 10 CFR 73.55(c)(7) and (8) are fully implemented. TVA shall submit a letter under oath or affirmation when the requirements of 73.55(c)(7) and (8) have been fully implemented.

1.1 Definitions

PHYSICS TESTS (continued)	a. Described in Chapter 14, Initial Test Program of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT	The PTLR is the unit specific document that provides the RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.9.6. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)."
QUADRANT POWER TILT RATIO (QPTR)	QPTR 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)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3411~~ 3459 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that 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.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or

(continued)

REPLACE WITH THE ATTACHED FIGURE

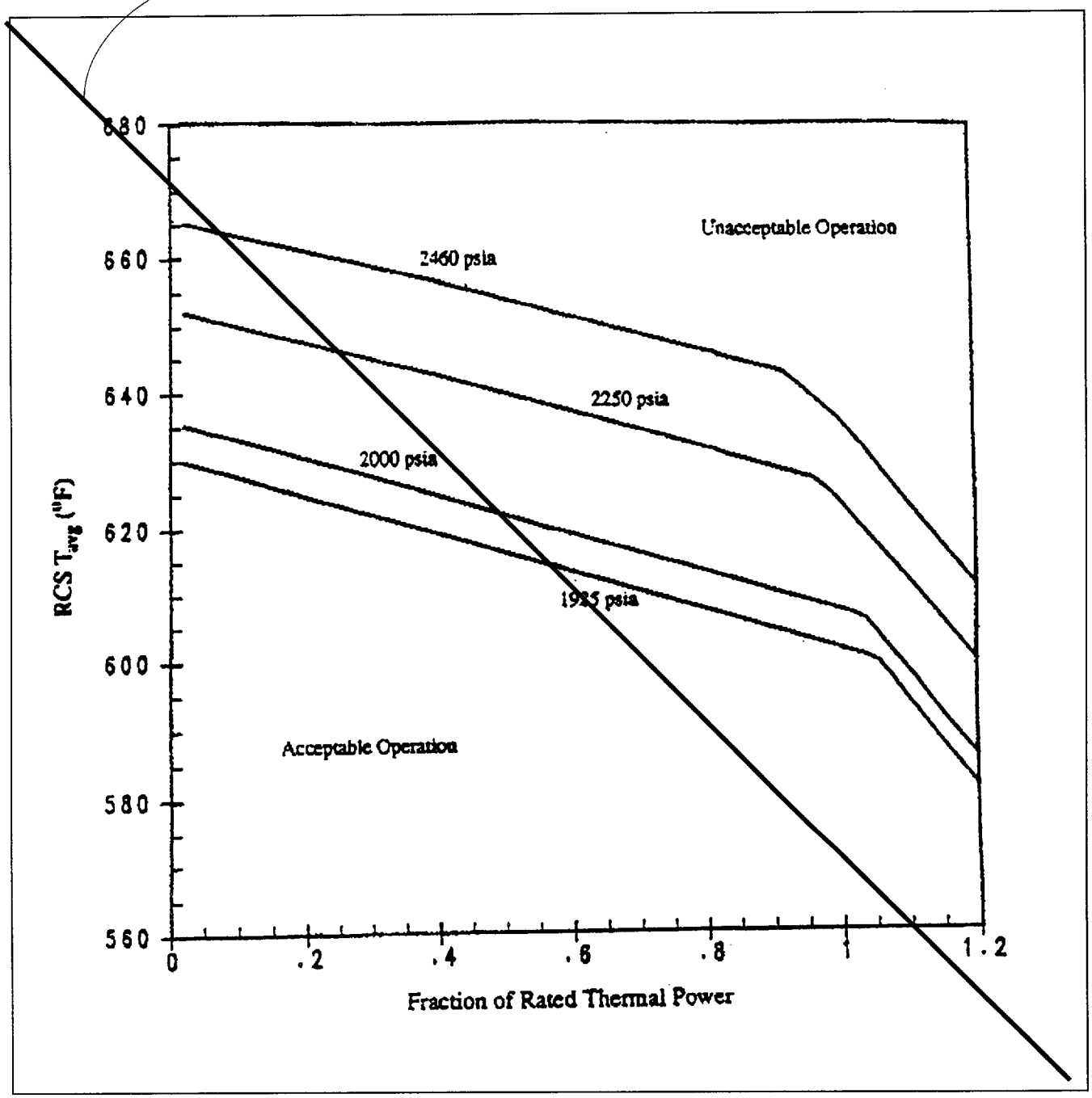


Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

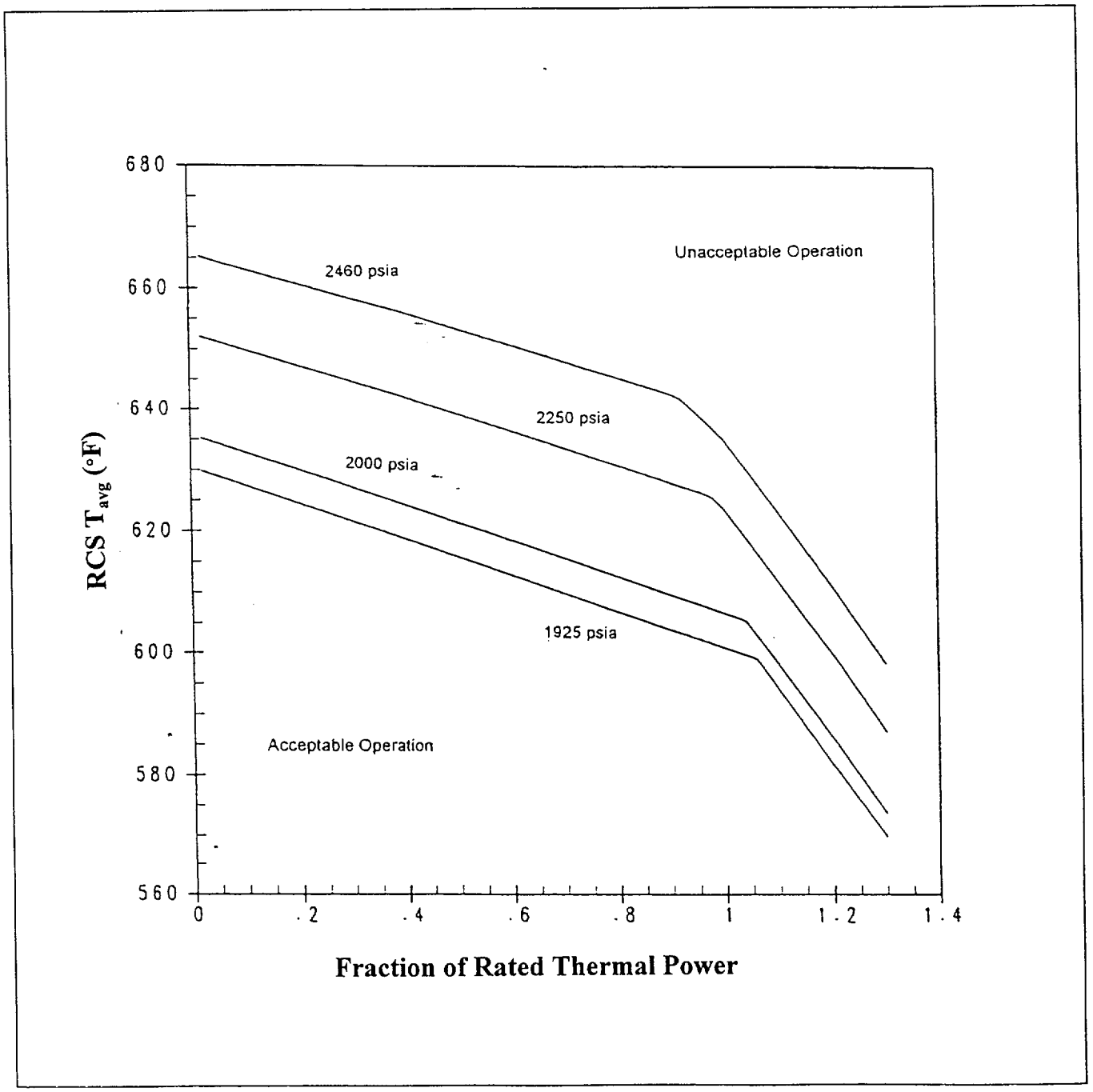


Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Five MSSVs per steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more steam generators with one MSSV inoperable.	A.1	Reduce THERMAL POWER to ≤ ~~59 %~~ 58% RTP.	4 hours
B. One or more steam generators with two or more MSSVs inoperable.	B.1	Reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs. AND ---------NOTE-------- Only required in MODE 1 ----------	4 hours
	B.2	Reduce the Power Range Neutron Flux - High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	36 hours
C. Required Action and associated Completion Time not met. OR One or more steam generators with ≥ 4 MSSVs inoperable.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 4.	12 hours

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Maximum Allowable Power

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (% RTP)
3	≤ ~~42~~ 41
2	≤ ~~26~~ 25

5.9 Reporting Requirements (continued)

5.9.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to the initial and each reload cycle, or prior to any remaining portion of a cycle, and shall be documented in the COLR for the following:

LCO 3.1.4 Moderator Temperature Coefficient
LCO 3.1.6 Shutdown Bank Insertion Limit
LCO 3.1.7 Control Bank Insertion Limits
LCO 3.2.1 Heat Flux Hot Channel Factor
LCO 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor
LCO 3.2.3 Axial Flux Difference
LCO 3.9.1 Boron Concentration

INSERT A

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

Delete

1. WCAP-9272-P-A, WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary). (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalphy Rise Hot Channel Factor, 3.2.3 - Axial Flux Difference, and 3.9.1 - Boron Concentration.

2a. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," March 1998 (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

2b. WCAP-10054-P-A, "Small Break ECCS Evaluation Model Using NOTRUMP Code," August 1985, Addendum 2, Rev.1: "Addendum to the Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model, July 1997, (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

3. WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor (W(Z) Surveillance Requirements For F(Q) Methodology) and 3.2.3 - Axial Flux Difference (Relaxed Axial Offset Control).)

(continued)

5.9 Reporting Requirements (continued)

5.9.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT," April 1995. (W Proprietary). (Methodology for Specification 3.2.1 - Heat FTux Hot Channel Factor).

5. WCAP-15088-P, Rev 1, "Safety Evaluation Supporting A More Negative EOL Moderator Temperature Coefficient Technical Specification for the Watts Bar Nuclear Plant," July 1999. (W Proprietary), as approved by the NRC staff's Safety Evaluation accompanying the issuance of Amendment No. 20 (Methodology for Specification 3.1.4 Moderator Temperature Coefficient.).

INSERT B

(continued)

INSERTS

INSERT A

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102 percent of rated thermal power is specified in a previously approved method, 100.6 percent of rated thermal power may be used only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM) as described in document number 6 listed below. When feedwater flow measurements from the LEFM are unavailable, the originally approved initial power level of 102 percent of rated thermal power (3411 MWt) shall be used.

The approved analytical methods are specifically those described in the following documents:

INSERT B

6. Caldon, Inc. Engineering Report-80P, "Improving Thermal Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997; and Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™," Revision 0, May 2000; as approved by the NRC staff's Safety Evaluation accompanying the issuance of Amendment No. .

BASES (continued)

LCO

The accident analysis requires that five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at ~~102%~~ ~~RTP. The~~ 100.6% LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2 and the DBA analysis.

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances to relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB, or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressuration.

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.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

With one or more MSSVs inoperable, action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements.

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

power. Single failures that also affect this event include the following:

a. Failure of the diesel generator powering the motor driven AFW pump to the unaffected steam generators (requiring additional steam to drive the remaining AFW pump turbine); and

b. Failure of the steam driven AFW pump (requiring a longer time for cooldown using only one motor driven AFW pump).

These are not usually the limiting failures in terms of consequences for these events.

A nonlimiting event considered in CST inventory determinations is a break in either the main feedwater bypass line or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action. This loss of condensate inventory is partially compensated for by the retention of steam generator inventory.

Because the CST is the preferred source of feedwater and is relied on almost exclusively for accidents and transients, the CST satisfies Criterion 3 of the NRC Policy Statement.

LCO

As the preferred water source to satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for 2 hours following a reactor trip from 100.6% ~~102%~~ RTP, and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine, or before isolating AFW to a broken line.

The CST level required is equivalent to a usable volume of ≥ 200,000 gallons, which is based on holding the unit in MODE 3 for 2 hours, followed by a cooldown to RHR entry conditions at 50°F/hour. This basis is established in Reference 4 and exceeds the volume required by the accident analysis.

(continued)

ENCLOSURE 9

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1 - DOCKET NO. 390

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-00-06

REVISED PAGES

I. AFFECTED PAGE LIST

Operating License
Page 3

Technical Specification:
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2.0-2
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5.0-32
5.0-32a (No changes - Provided for page integrity)
5.0-32b

Technical Specification Bases:
B 3.7-3
B 3.7-35

II. REVISED PAGES

ATTACHED

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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

TVA is authorized to operate the facility at reactor core power levels not in excess of 3459 megawatts thermal.

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Prior to startup following the first refueling outage, TVA shall accomplish the necessary activities, provide acceptable responses, and implement all proposed corrective actions related to having the Watts Bar Unit 1 SPDS operational.

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During the period of the exemption granted in paragraph 2.D.(3) of this license, in implementing the power ascension phase of the approved initial test program, TVA shall not exceed 50% power until the requirements of 10 CFR 73.55(c)(7) and (8) are fully implemented. TVA shall submit a letter under oath or affirmation when the requirements of 73.55(c)(7) and (8) have been fully implemented.

1.1 Definitions

PHYSICS TESTS (continued)	a. Described in Chapter 14, Initial Test Program of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT	The PTLR is the unit specific document that provides the RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.9.6. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)."
QUADRANT POWER TILT RATIO (QPTR)	QPTR 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)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3459 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that 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.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or

(continued)

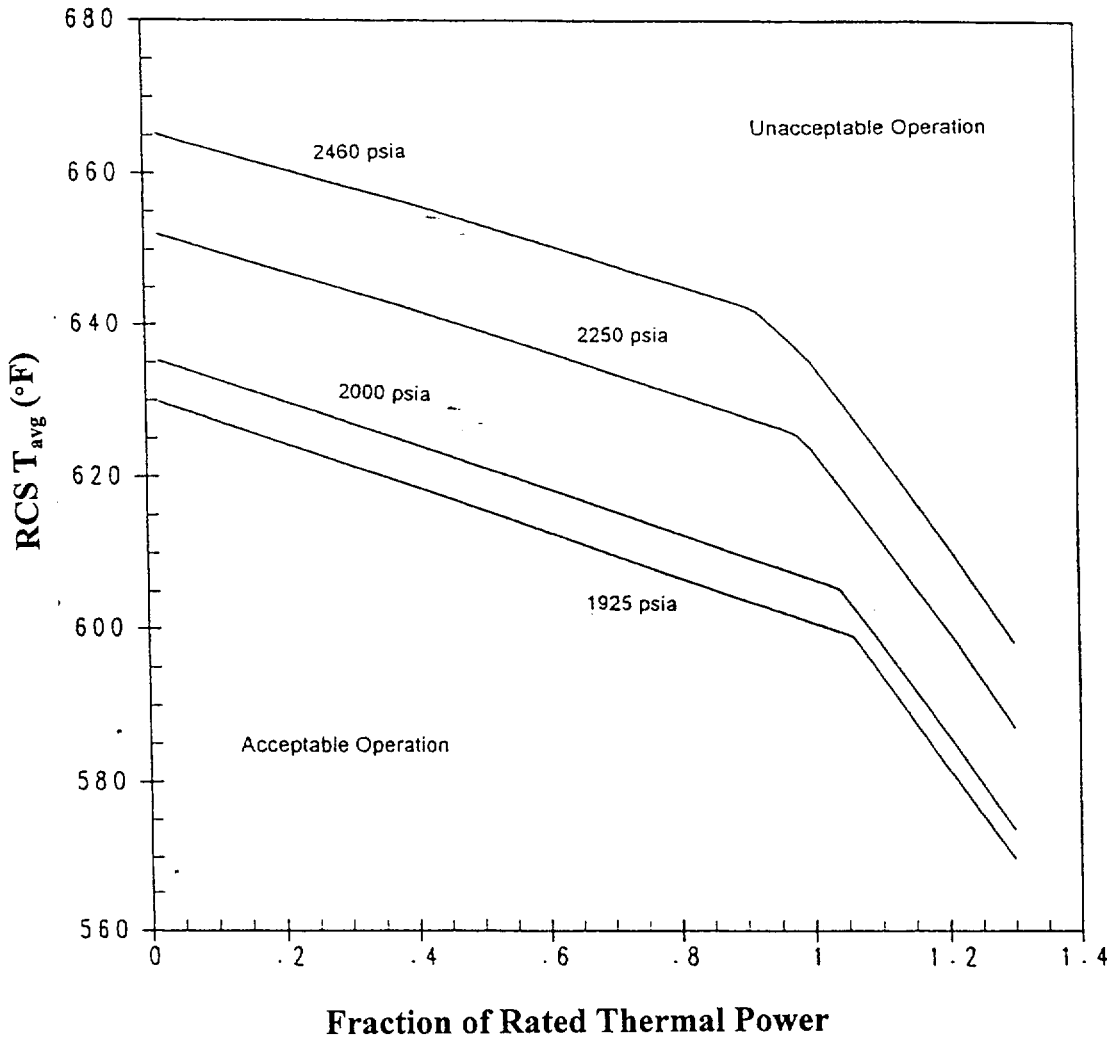

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Five MSSVs per steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more steam generators with one MSSV inoperable.	A.1	Reduce THERMAL POWER to ≤ 58 % RTP.	4 hours
B. One or more steam generators with two or more MSSVs inoperable.	B.1	Reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs. AND ----------NOTE--------- Only required in MODE 1 ----------	4 hours
	B.2	Reduce the Power Range Neutron Flux - High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	36 hours
C. Required Action and associated Completion Time not met. OR One or more steam generators with ≥ 4 MSSVs inoperable.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 4.	12 hours

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Maximum Allowable Power

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (% RTP)
3	≤ 41
2	≤ 25

5.9 Reporting Requirements (continued)

5.9.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to the initial and each reload cycle, or prior to any remaining portion of a cycle, and shall be documented in the COLR for the following:

LCO 3.1.4 Moderator Temperature Coefficient
LCO 3.1.6 Shutdown Bank Insertion Limit
LCO 3.1.7 Control Bank Insertion Limits
LCO 3.2.1 Heat Flux Hot Channel Factor
LCO 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor
LCO 3.2.3 Axial Flux Difference
LCO 3.9.1 Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102 percent of rated thermal power is specified in a previously approved method, 100.6 percent of rated thermal power may be used only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM) as described in document number 6 listed below. When feedwater flow measurements from the LEFM are unavailable, the originally approved initial power level of 102 percent of rated thermal power (3411 MWt) shall be used.

The approved analytical methods are specifically those described in the following documents:

1. WCAP-9272-P-A, WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary). (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalphy Rise Hot Channel Factor, 3.2.3 - Axial Flux Difference, and 3.9.1 - Boron Concentration.

(continued)

5.9 Reporting Requirements (continued)

5.9.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

2a. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," March 1998 (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

b. WCAP-10054-P-A, "Small Break ECCS Evaluation Model Using NOTRUMP Code," August 1985, Addendum 2, Rev. 1: "Addendum to the Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997, (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

3. WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).(Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor (W(Z) Surveillance Requirements For F(Q) Methodology) and 3.2.3 - Axial Flux Difference (Relaxed Axial Offset Control).)

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT," April 1995. (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).

5. WCAP-15088-P, Rev 1, "Safety Evaluation Supporting A More Negative EOL Moderator Temperature Coefficient Technical Specification for the Watts Bar Nuclear Plant," July 1999. (W Proprietary), as approved by the NRC staff's Safety Evaluation accompanying the issuance of Amendment No. 20 (Methodology for Specification 3.1.4 Moderator Temperature Coefficient.).

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5.9 Reporting Requirements (continued)

5.9.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. Caldon, Inc. Engineering Report-80P, "Improving Thermal Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997; and Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™," Revision 0, May 2000; as approved by the NRC staff's Safety Evaluation accompanying the issuance of Amendment No. .

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BASES (continued)

LCO

The accident analysis requires that five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 100.6% RTP. The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2 and the DBA analysis.

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances to relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB, or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressuration.

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.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

With one or more MSSVs inoperable, action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements.

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

power. Single failures that also affect this event include the following:

a. Failure of the diesel generator powering the motor driven AFW pump to the unaffected steam generators (requiring additional steam to drive the remaining AFW pump turbine); and

b. Failure of the steam driven AFW pump (requiring a longer time for cooldown using only one motor driven AFW pump).

These are not usually the limiting failures in terms of consequences for these events.

A nonlimiting event considered in CST inventory determinations is a break in either the main feedwater bypass line or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action. This loss of condensate inventory is partially compensated for by the retention of steam generator inventory.

Because the CST is the preferred source of feedwater and is relied on almost exclusively for accidents and transients, the CST satisfies Criterion 3 of the NRC Policy Statement.

LCO

As the preferred water source to satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for 2 hours following a reactor trip from 100.6% RTP, and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine, or before isolating AFW to a broken line.

The CST level required is equivalent to a usable volume of ≥ 200,000 gallons, which is based on holding the unit in MODE 3 for 2 hours, followed by a cooldown to RHR entry conditions at 50°F/hour. This basis is established in Reference 4 and exceeds the volume required by the accident analysis.

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