WCAP-14788

# Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Wisconsin Electric Power Company Point Beach Units 1 & 2 (Fuel Upgrade & Uprate to 1656 Mwt - NSSS Power)

April 1999

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WESTINGHOUSE REVISED THERMAL DESIGN PROCEDURE

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(FUEL UPGRADE & UPRATE TO 1656 MWT - NSSS POWER)

## I. INTRODUCTION

An upgrade to the fuel product at uprated conditions of 1656 Mwt - NSSS power has been requested by Wisconsin Electric for both Units 1&2 at Point Beach Nuclear Power Station. The fuel product to satisfy the intended requirements is the Westinghouse 14 X 14 PERFORMANCE + 422 fuel assembly. To utilize this new fuel assembly at uprated conditions, a new accident analysis will be required in addition to recalculating and revising the Instrument Uncertainty Methodology. This report supersedes "ITDP Instrument Uncertainty Report" dated June 7, 1984 (84WE*-G-044).

Four operating parameter uncertainties are used in the uncertainty analysis of the Revised Thermal Design Procedure (RTDP). These parameters are Pressurizer Pressure, Primary Coolant Temperature ($T_{avg}$), Reactor Power, and Reactor Coolant System Flow. They are frequently monitored and several are used for control purposes. Reactor power is monitored by the performance of a secondary side heat balance (power calorimetric) at least every 24 hours. RCS flow is monitored by the performance of a calorimetric flow measurement at the beginning of each cycle. The RCS Cold Leg loop flow indicators are evaluated against the calorimetric flow measurement. Pressurizer pressure is a controlled parameter and the uncertainty reflects the control system. $T_{avg}$ is a controlled parameter via the temperature input to the rod control system, and the uncertainty reflects this control system. The RTDP[1] is used to predict the plant's DNBR design limit. The RTDP methodology considers the uncertainties in the system operating plant parameters, fuel fabrication and nuclear and thermal parameters and includes the use of various DNB correlations. Use of the RTDP methodology requires that variances in the plant operating parameters are justified. The purpose of the following evaluation is to define the specific Point Beach Units 1 & 2 Nuclear Plant instrument uncertainties for the four primary system operating parameters which are used to predict the plant safety analysis DNBR design limit via the RTDP, and to determine the starting points of certain plant parameters in some of the accident analyses.

Westinghouse has been involved with the development of several techniques to treat instrumentation uncertainties. An early version (for D. C. Cook 2 and Trojan) used the methodology outlined in WCAP-8567 "Improved Thermal Design Procedure", [2,3,4] which is based on the conservative assumption that the uncertainties can be described with uniform probability distributions. Another approach is based on the more realistic assumption that the uncertainties can be described with random, normal, two sided probability distributions.[5] This approach is used to substantiate the acceptability of the protection system setpoints for many Westinghouse plants, e.g., D. C. Cook 2[6], V. C. Summer, Wolf Creek, Millstone Unit 3 and others. The second approach is now utilized for the determination of all instrumentation uncertainties for the RTDP parameters and protection functions.

The determination of pressure, temperature, power and RCS flow uncertainties are applicable for the Point Beach Plant Units 1 & 2 for power levels up to 1656 Mwt - NSSS power, for 18 month fuel cycles + 25% per the plant Technical Specifications, and for a full power Tavg window of 558.1 to 574.0°F.

## II. METHODOLOGY

The methodology used to combine the error components for a channel is the square root of the sum of the squares of those groups of components which are statistically independent. Those errors that are dependent are combined arithmetically into independent groups, which are then systematically combined. The uncertainties used are considered to be random, two sided distributions. The sum of both sides is equal to the range for that parameter, e.g., Rack Drift is typically [ ]$^{+a,c}$, the range for this parameter is [ ]$^{+a,c}$. This technique has been utilized before as noted above, and has been endorsed by the NRC staff[7,8,9,10] and various industry standards[11,12].

The relationships between the error components and the channel instrument error allowance are variations of the basic Westinghouse Setpoint Methodology[13] and are defined as follows:

1. For precision parameter indication using Special Test Equipment or a digital voltmeter (DVM) at the input to the racks;

$$CSA = \{(SMTE + SCA)^2 + (SPE)^2 + (STE)^2 + (SMTE + SD)^2 + (SRA)^2 + (RDOUT)^2\}^{1/2} + BIAS \qquad \text{Eq. 1}$$

2. For parameter indication utilizing the plant process computer;

$$CSA = \{(SMTE + SCA)^2 + (SPE)^2 + (STE)^2 + (SMTE + SD)^2 + (SRA)^2 + (RMTE + RCA)^2 + (RTE)^2 + (RMTE + RD)^2 + (RMTE + A/D)^2\}^{1/2} + BIAS \qquad \text{Eq. 2}$$

3. For parameters with closed-loop automatic control systems, the calculation takes credit for [ ]$^{+a,c}$. There is a functional dependency between the transmitters/racks and the automatic control system/indicator when an uncertainty in the transmitters/racks is common to the automatic control system and the indicator. That is, an uncertainty in the high direction in the transmitter/ racks will result in a high uncertainty in the automatic control system and the indicator. To account for the functional dependency, a square root function is used for the transmitter/ racks/reference signal, and a square root function is used for the controller/indicators;

$$
\begin{aligned}
CSA = \{&(PMA^2(random) + (PEA)^2 \\
&+ (SMTE + SCA)^2 + (SPE)^2 + (STE)^2 + (SMTE + SD)^2 + (SRA)^2 \\
&+ (RMTE + RCA)^2 + (RTE)^2 + (RMTE + RD)^2 + (REF)^2\}^{1/2} \\
&+ \{(CMTE + CA)^2 + (RMTE + RCA)^2_{IND} + (RDOUT)^2_{IND}\}^{1/2} \\
&+ BIAS
\end{aligned}
\qquad \text{Eq. 3}
$$

where:

| | | |
|---|---|---|
| CSA | = | Channel Statistical Allowance |
| PMA | = | Process Measurement Accuracy |
| PEA | = | Primary Element Accuracy |
| SRA | = | Sensor Reference Accuracy |
| SCA | = | Sensor Calibration Accuracy |
| SMTE | = | Sensor Measurement and Test Equipment Accuracy |
| SPE | = | Sensor Pressure Effects |
| STE | = | Sensor Temperature Effects |
| SD | = | Sensor Drift |
| RCA | = | Rack Calibration Accuracy |
| RMTE | = | Rack Measurement and Test Equipment Accuracy |
| RTE | = | Rack Temperature Effects |
| RD | = | Rack Drift |
| RDOUT | = | Readout Device Accuracy |
| CA | = | Controller Allowance |
| CMTE | = | Controller Measurement and Test Equipment Acuracy |
| A/D | = | Analog to Digital Conversion Accuracy |
| REF | = | Reference signal for automatic control sytem |
| IND | = | Indicator. |

PMA and PEA terms are not included in equations 1 and 2 since the equations are to determine instrumentation uncertainties only. PMA and PEA terms are included in the determination of control system uncertainties.

The parameters above are defined in references 5 and 12 and are based on SAMA Standard PMC 20.1, 1973[14]. However, for ease in understanding they are paraphrased below:

| | |
|---|---|
| PMA | - non-instrument related measurement errors, e.g., temperature stratification of a fluid in a pipe. |
| PEA | - errors due to a metering device, e.g., elbow, venturi, orifice. |
| SRA | - reference (calibration) accuracy for a sensor/transmitter. |
| SCA | - calibration tolerance for a sensor/transmitter. |
| SMTE | - measurement and test equipment used to calibrate a sensor/transmitter. |
| SPE | - change in input-output relationship due to a change in static pressure |

fo. a differential pressure (d/p) cell.

STE - change in input-output relationship due to a change in ambient temperature for a sensor or transmitter.

SD - change in input-output relationship over a period of time at reference conditions for a sensor or transmitter.

RCA - calibration accuracy for all rack modules in loop or channel assuming the loop or channel is string calibrated, or tuned, to this accuracy.

RMTE - measurement and test equipment used to calibrate rack modules.

RTE - change in input-output relationship due to a change in ambient temperature for the rack modules.

RD - change in input-output relationship over a period of time at reference conditions for the rack modules.

RDOUT - the measurement accuracy of a special test local gauge, digital voltmeter or multimeter on it's most accurate applicable range for the parameter measured, or 1/2 the smallest increment on an indicator (readability).

CA - allowance of the controller rack module(s) that performs the comparison and calculates the difference between the controlled parameter and the reference signal.

CMTE - measurement and test equipment used to calibrate the controller rack module(s) that perform(s) the comparison between the controlled parameter and the reference signal.

A/D - allowance for conversion accuracy of an analog signal to a digital signal for process computer use.

REF - the reference signal uncertainty for a closed-loop automatic control system.

IND - indicator accuracies are used for these uncertainty calculations. Control board indicators are typically used.

BIAS - a one directional uncertainty for a sensor/transmitter or a process parameter with a known magnitude.

A more detailed explanation of the Westinghouse methodology noting the interaction of several parameters is provided in references 6 and 13.

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## III. INSTRUMENTATION UNCERTAINTIES

The instrumentation uncertainties will be discussed first for the two parameters which are controlled by automatic systems, Pressurizer Pressure, and $T_{avg}$ (through automatic rod control).

### 1. PRESSURIZER PRESSURE

Pressurizer pressure is controlled by a closed-loop automatic control system that compares the measured vapor space pressure to a reference value. This uncertainty calculation takes credit for the closed-loop control system design where [ ]$^{+a,c}$. The control channel uncertainties for the automatic control system include allowances for the pressure transmitters, the process racks, and the control system reference setpoint. The pressurizer pressure control system reference setpoint is generated by the setting of a variable potentiometer on the Main Control Board manual/automatic station. The reference setpoint (Pref) is adjusted and verified by the plant operators with the control board indicators. This uncertainty calculation also includes the control board indicators for verification of the automatic control system performance.

On Table 1, the electronics uncertainty for this function is [ ]$^{+a,c}$ with a [ ]$^{+a,c}$ bias corresponding to [ ]$^{+a,c}$ with a [ ]$^{+a,c}$ bias for the average of 3 control board indicators. In addition to the control system uncertainty, an allowance is made for pressure overshoot or undershoot due to the interaction and thermal inertia of the heaters and spray. An allowance of [ ]$^{+a,c}$ is made for this effect. The total control system uncertainty including indication is [ ]$^{+a,c}$ with a [ ]$^{+a,c}$ bias which results in a standard deviation of [ ]$^{+a,c}$ for a normal, two sided probability distribution.

TABLE 1
PRESSURIZER PRESSURE CONTROL SYSTEM UNCERTAINTY

All Values in % Span

| | | |
|---|---|---|
| REF | = | [ ]+a,c |
| PMA | = | |
| PEA | = | |
| SRA | = | |
| SCA | = | |
| SMTE | = | |
| STE | = | |
| SD | = | |
| BIAS | = | |
| RCA | = | |
| RMTE | = | |
| RTE | = | |
| RD | = | |
| $RCA_{IND}$ | = | |
| $RMTE_{IND}$ | = | |
| $RDOUT_{IND}$ | = | |
| CA | = | |
| CMTE | = | |

RANGE = 1700 - 2500 psig, SPAN = 800 psi
CHANNELS P-429, -430, -431 & -449

| | | |
|---|---|---|
| ELECTRONICS UNCERTAINTY | = | [ ]+a,c |
| PLUS | | |
| ELECTRONICS UNCERTAINTY | = | |
| PLUS | | |
| CONTROLLER UNCERTAINTY | = | |

* 15 psi setting tolerance around 2235 psig

2. Tavg

$T_{avg}$ is controlled by a system that compares the high $T_{avg}$ from the loops with a reference derived from the First Stage Turbine Impulse Chamber Pressure. $T_{avg}$ is the average of the narrow range $T_H$ and $T_C$ values. The high loop $T_{avg}$ is then used for rod control. Allowances are made (as noted on Table 2) for the RTDs, transmitter and the process racks/indicators and controller. The CSA for this function is dependent on the type of RTD, pressure transmitter, and the location of the RTDs, i.e., in the Hot and Cold Leg bypass manifolds. Based on one $T_H$ and one $T_C$ RTD per channel to calculate $T_{avg}$ and with the RTDs located in the hot and cold leg bypass manifolds, the CSA for the electronics is [ ]$^{+a,c}$. Assuming a normal, two sided probability distribution results in an electronics standard deviation ($s_1$) of [ ]$^{+a,c}$.

However, this does not include the deadband of ±1.5°F for automatic control. The $T_{avg}$ controller accuracy is the combination of the instrumentation accuracy and the deadband. The probability distribution for the deadband has been determined to be [ ].$^{+a,c}$ The variance for the deadband uncertainty is then:

$$(s_2)^2 = [\quad]^{+a,c} = [\quad]^{+a,c}$$

where [ ]$^{+a,c}$. Combining the variance for instrumentation and deadband results in a controller variance of:

$$(s_T)^2 = (s_1)^2 + (s_2)^2 = [\quad]^{+a,c}$$

The controller $s_T$ = [ ]$^{+a,c}$ for a total random uncertainty of [ ]$^{+a,c}$.

An additional bias of [ ]$^{+a,c}$ for $T_{cold}$ streaming (in terms of Tavg) based on a conservative [ ]$^{+a,c}$ $T_{cold}$ streaming uncertainty is included in Table 2. An additional bias of [ ]$^{+a,c}$ for R/E linearization (in terms of Tavg) is included in Table 2. Therefore, the total uncertainty of the controller with the additional biases is [ ]$^{+a,c}$ random and [ ]$^{+a,c}$ bias.

TABLE 2

TAVG ROD CONTROL SYSTEM UNCERTAINTY

| | | % $T_{avg}$ Span | % Turbine Pressure Span (P-485,-486) |
|---|---|---|---|
| $PMA_1$ | = | [ | ] +a,c |
| $PMA_2$ | = | | |
| SRA | = | | |
| SCA | = | | |
| SMTE | = | | |
| STE | = | | |
| SD | = | | |
| $BIAS_1$ | = | | |
| $BIAS_2$ | = | | |
| R/E | = | | |
| R/E_MTE | = | | |
| RCA | = | | |
| RMTE | = | | |
| RTE | = | | |
| RD | = | | |
| $RCA_{IND}$ | = | | |
| $RMTE_{IND}$ | = | | |
| $RDOUT_{IND}$ | = | | |
| CA | = | | |
| CMTE | = | | |

# Hot Leg RTDs = 1/Channel #Cold Leg RTDs = 1/Channel

Tavg span = 100 °F (520-620°F)

R/E span = 150 °F (500-650°F for Hot and Cold Leg)

Turbine pressure span = 650psi (0-650 psig)

| | | |
|---|---|---|
| ELECTRONICS CSA | = | [ ] +a,c |
| ELECTRONICS SIGMA | = | |
| CONTROLLER SIGMA | = | |
| CONTROLLER CSA | = | |
| CONTROLLER BIAS | = | |

Note A: Module TM-401BB = 0.5% span

Note B: Module PM-485A = 0.5% span

Modules for Loop A1 similar for A2, B1 & B2.

Note C: Module TM-401EE = 0.5% span
Module TM-401D = 0.5% span
Module TM-401H = 0.5% span
Module TM-401M = 0.25% span
Module TM-401P = 0.82% span
Module TM-401N = 1.63% span
Module TM-401I = 0.5% span

## 3. RCS FLOW

Calorimetric RCS Flow Measurement Uncertainty (Using LEFM on the Feedwater Header)

RTDP and Point Beach's Technical Specifications require an RCS flow measurement with a high degree of accuracy. A total RCS flow measurement every fuel cycle, 18 months, is performed to verify RCS flow and to normalize the RCS flow instrument channels. Interim surveillances performed with the process computer ensure that the RCS flow is maintained within the assumed safety analysis values, i.e., Minimum Measured Flow (MMF). The 18 month RCS flow surveillance is satisfied by a secondary side power-based calorimetric RCS flow measurement. The calorimetric flow measurement is performed at the beginning of a cycle near full power operation.

Eighteen month instrument drift is used in this uncertainty analysis for hot and cold leg RTDs, and for feedwater pressure, steam pressure and pressurizer pressure transmitters.

A Leading Edge Flow Meter (LEFM) installed on the Feedwater header is used to determine total Feedwater flow. Feedwater temperature indication by the LEFM is compared to individual loop feedwater temperatures which are then adjusted if necessary.

The flow measurement is performed by determining the Steam Generator thermal output (corrected for the RCP heat input and the loop's share of primary system heat losses) and the enthalpy rise (Delta-h) of the primary coolant. Assuming that the primary and secondary sides are in equilibrium, the RCS total vessel flow is the sum of the individual primary loop flows, i.e.,

$$W_{RCS} = \Sigma_{i=1}^{N}(W_L)_i \qquad \text{Eq. 4}$$

The individual primary loop volumetric flows are determined by correcting the thermal output of the Steam Generator for Steam Generator blowdown (if not secured), subtracting the RCP heat addition, adding the loop's share of the primary side system losses, dividing by the primary side enthalpy rise and multiplying by the Cold Leg specific volume. The equation for this calculation is:

$$W_L = \frac{(A)\{Q_{SG} - Q_P + (Q_L/N)\}(V_C)}{(h_H - h_C)} \qquad \text{Eq. 5}$$

where;

| | | |
|---|---|---|
| $W_L$ | = | Loop Flow (gpm) |
| A | = | Constant conversion factor 0.1247 gpm/(ft$^3$/hr) |
| $Q_{SG}$ | = | Steam Generator thermal output (BTU/hr) |
| $Q_P$ | = | RCP heat addition (BTU/hr) |
| $Q_L$ | = | Primary system net heat losses (BTU/hr) |

$V_C$ = Specific volume of the Cold Leg at $T_C$ ($ft^3$/lb)

N = Number of primary side loops

$h_H$ = Hot Leg enthalpy (BTU/lb)

$h_C$ = Cold Leg enthalpy (BTU/lb).

The thermal output of the Steam Generator is determined by a secondary side calorimetric measurement, which is defined as:

$$Q_{SG} = (h_s - h_f)W_f \qquad \text{Eq. 6}$$

where;

$h_s$ = Steam enthalpy (BTU/lb)

$h_f$ = Feedwater enthalpy (BTU/lb)

$W_f$ = Feedwater flow (LEFM feedwater header flow divided by # loops)(lb/hr).

The Steam enthalpy is based on the measurement of Steam Generator outlet Steam pressure assuming saturated conditions. The Feedwater enthalpy is based on the measurement of Feedwater temperature and nominal Feedwater pressure. The Feedwater flow is determined by LEFM measurements.

RCP heat addition is determined by calculation, based on the best estimate of coolant flow, pump head, and pump hydraulic efficiency.

The primary system net heat losses are determined by calculation, considering the following system heat inputs (+) and heat losses (-):

Charging flow (+)
Letdown flow (-)
Seal injection flow (+)
RCP thermal barrier cooler heat removal (-)
Pressurizer spray flow (-)
Pressurizer surge line flow (+)
Component insulation heat losses (-)
Component support heat losses (-)
CRDM heat losses (-).

A single calculated sum for 100% RTP operation is used for these losses or heat inputs.

The Hot Leg and Cold Leg enthalpies are based on the measurement of the Hot Leg temperature, Cold Leg temperature and the nominal Pressurizer pressure. The Cold Leg specific volume is based on measurement of the Cold Leg temperature and nominal Pressurizer pressure.

The RCS flow measurement is thus based on the following plant measurements:

Steamline pressure ($P_s$)
Feedwater temperature ($T_f$)
Feedwater pressure ($P_f$)
Feedwater flow from LEFM
Hot Leg temperature ($T_H$)
Cold Leg temperature ($T_C$)
Pressurizer pressure ($P_p$)
Steam Generator blowdown flow (if not secured)

and on the following calculated values:

Feedwater density ($\rho_f$)
Feedwater enthalpy ($h_f$)
Steam enthalpy ($h_s$)
Moisture carryover (impacts $h_s$)
Primary system net heat losses ($Q_L$)
RCP heat addition ($Q_p$)
Hot Leg enthalpy ($h_H$)
Cold Leg enthalpy ($h_C$).

These measurements and calculations are presented schematically in Figure 1. The derivation of the measurement and flow uncertainties on Table 5 are noted below.

Secondary Side

The secondary side uncertainties are in four principal areas, Feedwater flow, Feedwater enthalpy, Steam enthalpy and net pump heat addition. These areas are specifically identified on Table 5.

For the measurement of Feedwater flow, the LEFM is located on the feedwater header and provides a total flow. The accuracy to which the total flow is determined is based on calculations performed by the manufacture of the LEFM.

Using the NBS/NRC Steam Tables it is possible to determine the sensitivities of various parameters to changes in Feedwater temperature and pressure. Table 3 notes the instrument uncertainties for the hardware used to perform the measurements. Table 4 lists the various sensitivities. As can be seen on Table 5, Feedwater temperature uncertainties have an impact on Feedwater density and Feedwater enthalpy. Feedwater pressure uncertainties impact Feedwater density and Feedwater enthalpy.

Using the NBS/NRC Steam Tables, it is possible to determine the sensitivity of Steam enthalpy to changes in Steam pressure and Steam quality. Table 3 notes the uncertainty in Steam pressure and Table 4 provides the sensitivity. For Steam quality, the Steam Tables were used to determine the sensitivity at a moisture content of [ ]+a,c. This value is noted on Table 4.

The net pump heat addition uncertainty is derived from the combination of the primary system net heat losses and pump heat addition and are summarized for a two loop plant as follows:

| | |
|---|---|
| System heat losses | - 2.0 MWt |
| Component conduction and convection losses | - 1.4 MWt |
| Pump heat adder | + 9.4 MWt |
| Net Heat input to RCS | + 6.0 MWt |

The uncertainty on system heat losses, which is essentially all due to charging and letdown flows, has been estimated to be [ ]+a,c of the calculated value. Since direct measurements are not possible, the uncertainty on component conduction and convection losses has been assumed to be [ ]+a,c of the calculated value. Reactor coolant pump hydraulics are known to a relatively high confidence level, supported by system hydraulics tests performed at Prairie Island Unit 2 and by input power measurements from several other plants. Therefore, the uncertainty for the pump heat addition is estimated to be [ ]+a,c of the best estimate value. Considering these parameters as one quantity, which is designated the net pump heat addition uncertainty, the combined uncertainties are less than [ ]+a,c of the total, which is [ ]+a,c of core power.

Primary Side

The primary side uncertainties are in three principal areas, hot leg enthalpy, cold leg enthalpy and cold leg specific volume. These are specifically noted on Table 5. Three primary side parameters are actually measured, $T_H$, $T_C$ and Pressurizer pressure. Hot Leg enthalpy is influenced by $T_H$, Pressurizer pressure and Hot Leg temperature streaming. The uncertainties for the instrumentation are noted on Table 3 and the sensitivities are provided on Table 4. The hot leg streaming is split into random and systematic components. For Point Beach Units 1 & 2 where the RTDs are located in bypass manifolds, the hot leg temperature streaming uncertainty components are [ ]+a,c random and [ ]+a,c systematic.

The cold leg enthalpy and specific volume uncertainties are impacted by $T_C$ and Pressurizer pressure. Table 3 notes the $T_C$ instrument uncertainty and Table 4 provides the sensitivities.

Parameter dependent effects are identified on Table 5. Westinghouse has determined the dependent sets in the calculation and the direction of interaction, i.e., whether components in a dependent set are additive or subtractive with respect to a conservative calculation of RCS flow. The same work was performed for the instrument bias values. As a result, the calculation

explicitly accounts for dependent effects and biases with credit taken for sign (or direction of impact).

Using Table 5, the 2 loop uncertainty equation (with biases) is as follows:

[ ]+a,c

Based on the number of loops; number, type and measurement method of RTDs, and the vessel Delta-T, the flow uncertainty is:

| # of loops | flow uncertainty (% flow) |
|---|---|
| 2 | [ ]+a,c |

TABLE 3

FLOW CALORIMETRIC INSTRUMENTATION UNCERTAINTIES
% SPAN

| | FW TEMP | FW PRES | FW FLOW | STM PRESS | TH | TC | PRZ PRESS |
|---|---|---|---|---|---|---|---|
| LEFM = | [redacted] | | | | | | [redacted] +a,c |
| SRA = | | | | | | | |
| SCA = | | | | | | | |
| SMTE = | | | | | | | |
| SPE = | | | | | | | |
| STE = | | | | | | | |
| SD = | | | | | | | |
| BIAS = | | | | | | | |
| R/E = | | | | | | | |
| RMTE = | | | | | | | |
| RTE = | | | | | | | |
| RD = | | | | | | | |
| RDOUT = | | | | | | | |
| CSA = | | | | | | | |
| # OF INSTRUMENTS USED | 1/Loop | 1/Loop | 1 | 1/Loop | 2/Loop | 2/Loop | 4 |
| | °F | psi | % Flow | psi | °F | °F | psi |
| INST SPAN = | (1) | 1500(2) | 100(3) | 1400(4) | 150(5) | 150(5) | 800(6) |
| INST UNC. (RANDOM) = | [redacted] | | | | | | [redacted] +a,c |
| INST UNC. (BIAS) = | | | | | | | |
| NOMINAL = | 440.7 °F | 800-875 psia | 100 % Flow | 700-775 psia | 590.2-605.5 °F | 526.0-542.5 °F | 2250 psia |

(1) Special test instrumentation TE-3111 and an Resistance Thermometer bridge are used for this measurement.
(2) Pressure (P-2245) is measured with a digital voltmeter at the input of the process instrumentation.
(3) Flow (F-3110) is measured with an LEFM on the feedwater header.
(4) Pressure (P-468, -469, -478, -479, -482, -483) is measured with a digital voltmeter at the input of the process instrumentation.
(5) Temperature is measured with a digital voltmeter at the output of the R/E process instrumentation modules.
(6) Pressure (P-429, -430, -431, -449) is measured with a digital voltmeter at the input of the process instrumentation.

TABLE 4

FLOW CALORIMETRIC SENSITIVITIES

+a,c

| | |
|---|---|
| FEEDWATER FLOW | |
| LEFM | = |
| DENSITY | |
| TEMPERATURE | = |
| PRESSURE | = |
| FEEDWATER ENTHALPY | |
| TEMPERATURE | = |
| PRESSURE | = |
| $h_s$ | = |
| $h_f$ | = |
| Dh (SG) | = |
| STEAM ENTHALPY | |
| PRESSURE | = |
| MOISTURE | = |
| HOT LEG ENTHALPY | |
| TEMPERATURE | = |
| PRESSURE | = |
| $h_H$ | = |
| $h_C$ | = |
| Dh (VESS) | = |
| COLD LEG ENTHALPY | |
| TEMPERATURE | = |
| PRESSURE | = |
| COLD LEG SPECIFIC VOLUME | |
| TEMPERATURE | = |
| PRESSURE | = |

TABLE 5

CALORIMETRIC RCS FLOW MEASUREMENT UNCERTAINTY

| COMPONENT | INSTRUMENT UNCERTAINTY | FLOW UNCERTAINTY +a,c |
|---|---|---|
| FEEDWATER FLOW | | |
| LEFM | | |
| DENSITY | | |
| TEMPERATURE | | |
| PRESSURE | | |
| FEEDWATER ENTHALPY | | |
| TEMPERATURE | | |
| PRESSURE | | |
| STEAM ENTHALPY | | |
| PRESSURE | | |
| MOISTURE | | |
| NET PUMP HEAT ADDITION | | |
| HOT LEG ENTHALPY | | |
| TEMPERATURE | | |
| STREAMING, RANDOM | | |
| STREAMING, SYSTEMATIC | | |
| PRESSURE | | |
| COLD LEG ENTHALPY | | |
| TEMPERATURE | | |
| PRESSURE | | |
| COLD LEG SPECIFIC VOLUME | | |
| TEMPERATURE | | |
| PRESSURE | | |

*, **, +, ++ INDICATES SETS OF DEPENDENT PARAMETERS

TABLE 5 (CONTINUED)

CALORIMETRIC RCS FLOW MEASUREMENT UNCERTAINTY

| COMPONENT | FLOW UNCERTAINTY |
|---|---|
| | +a,c |
| BIAS VALUES | |
| FEEDWATER PRESSURE | [ ] |
| DENSITY | |
| ENTHALPY | |
| STEAM PRESSURE | |
| ENTHALPY | |
| PRESSURIZER PRESSURE | |
| ENTHALPY - HOT LEG | |
| ENTHALPY - COLD LEG | |
| SPECIFIC VOLUME - COLD LEG | |
| COLD LEG ENTHALPY | |
| TEMPERATURE | |
| COLD LEG SPECIFIC VOLUME | |
| TEMPERATURE | |
| FLOW BIAS TOTAL VALUE | |
| | +a,c |
| 2 LOOP UNCERTAINTY (WITHOUT BIAS VALUES) | [ ] |
| 2 LOOP UNCERTAINTY (WITH BIAS VALUES) | |

Loop RCS Flow Uncertainty (Using Plant Computer)

As noted earlier, the calorimetric RCS flow measurement is used as the reference for normalizing the loop RCS flow measurement from the cold leg elbow tap transmitters. Since the cold leg elbow tap transmitters feed the plant computer, it is a simple matter to perform an RCS flow surveillance. Table 6 notes the instrument uncertainties for determining flow by using the loop RCS flow channels and the plant computer, assuming one loop RCS flow channel per reactor coolant loop. The d/p transmitter uncertainties are converted to percent flow using the following conversion factor:

$$\% \text{ flow} = (\text{d/p uncertainty})(1/2)(\text{FLOWmax} / \text{FLOWnominal})^2$$

where FLOWmax is the maximum value of the loop RCS flow channel. The loop RCS flow uncertainty is then combined with the calorimetric RCS flow measurement uncertainty. This combination of uncertainties results in the following total flow uncertainty:

| # of loops | flow uncertainty ( % flow ) |
|---|---|
| 2 | ± 2.3 |

The corresponding value used in RTDP is:

| # of loops | standard deviation ( % flow ) |
|---|---|
| 2 | [ ] +a,c |

TABLE 6

LOOP RCS FLOW UNCERTAINTY
PLANT COMPUTER

INSTRUMENT UNCERTAINTIES
1 LOOP RCS FLOW CHANNEL PER REACTOR COOLANT LOOP
(F-411, -412, -413, -414, -415, -416)

| | % d/p SPAN | % Flow |
|---|---|---|
| PMA = | [ | ] +a,c |
| PEA = | | |
| SRA = | | |
| SCA = | | |
| SMTE = | | |
| SPE = | | |
| STE = | | |
| SD = | | |
| BIAS = | | |
| RCA = | | |
| RMTE = | | |
| RTE = | | |
| RD = | | |
| A/D = | | |
| A/D_MTE= | | |
| FLOW CALORIMETRIC BIAS = | | |
| FLOW CALORIMETRIC = | | |
| INSTRUMENT SPAN = | ] | |

| | | | |
|---|---|---|---|
| SINGLE LOOP ELBOW TAP FLOW UNCERTAINTY | = | [ | ] +a,c |
| 2 LOOP RCS FLOW UNCERTAINTY (WITHOUT BIAS VALUES) | = | | |
| 2 LOOP RCS FLOW UNCERTAINTY (WITH BIAS VALUES) | = | 2.3 | |

Note A: Module FM-411, -412, -413, -414, -415, -416 = 0.5% span
* Zero values due to normalization to calorimetric RCS flow measurement

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## 4. REACTOR POWER

The plant performs a primary/secondary side heat balance at least every 24 hours when power is above 15% Rated Thermal Power. This heat balance is used to verify that the plant is operating within the limits of the Operating License and to adjust the Power Range Neutron Flux channels when the difference between the Power Range Neutron Flux channels and the heat balance is greater than required by the plant Technical Specifications.

Assuming that the primary and secondary sides are in equilibrium; the core power is determined by summing the thermal output of the steam generators, correcting the total secondary power for Steam Generator blowdown (if not secured), subtracting the RCP heat addition, adding the primary side system losses, and dividing by the core Btu/hr at rated full power. The equation for this calculation is:

$$RP = \frac{\{\sum_{i=1}^{N}\{Q_{SG} - Q_P + (Q_L/N)\}_i\}(100)}{H} \qquad \text{Eq. 7}$$

where;

| | |
|---|---|
| RP | = Core power ( % RTP ) |
| N | = Number of primary side loops |
| $Q_{SG}$ | = Steam generator thermal output (BTU / hr ) as defined in Eq. 6 |
| $Q_P$ | = RCP heat addition (BTU / hr ) as defined in Eq. 5 |
| $Q_L$ | = Primary system net heat losses (BTU / hr ) as defined in Eq. 5 |
| H | = Rated core power (BTU / hr). |

For the purposes of this uncertainty analysis (and based on H noted above) it is assumed that the plant is at 100% RTP when the measurement is taken. Measurements performed at lower power levels will result in different uncertainty values. However, operation at lower power levels results in increased margin to DNB far in excess of any margin losses due to increased measurement uncertainty.

The feedwater flow in equation 6 is determined by multiple measurements and the following calculation:

$$W_f = (K)(F_a)\{(p_f)(d/p)\}^{1/2} \qquad \text{Eq. 8}$$

where:

| | |
|---|---|
| $W_f$ | = Feedwater loop flow (lb/hr) |
| K | = Feedwater venturi flow coefficient |
| $F_a$ | = Feedwater venturi correction for thermal expansion |
| $p_f$ | = Feedwater density (lb/ft$^3$) |
| d/p | = Feedwater venturi pressure drop (inches $H_2O$). |

The feedwater venturi flow coefficient is the product of a number of constants including as-built dimensions of the venturi and calibration tests performed by the vendor. The thermal expansion correction is based on the coefficient of expansion of the venturi material and the difference between feedwater temperature and calibration temperature. Feedwater density is based on the measurement of feedwater temperature and feedwater pressure. The venturi pressure drop is obtained from the output of the differential pressure transmitter connected to the venturi.

The power measurement is thus based on the following plant measurements:

Steamline pressure ($P_s$)
Feedwater temperature ($T_f$)
Feedwater pressure ($P_f$)
Feedwater venturi differential pressure (d/p)
Steam generator blowdown (if not secured);

and on the following calculated values:

Feedwater venturi flow coefficients (K)
Feedwater venturi thermal expansion correction ($F_a$)
Feedwater density ($p_f$)
Feedwater enthalpy ($h_f$)
Steam enthalpy ($h_s$)
Moisture carryover (impacts $h_s$)
Primary system net heat losses ($Q_L$)
RCP heat addition ($Q_p$)

Secondary Side

The secondary side power calorimetric equations and effects are the same as those noted for the calorimetric RCS flow measurement (secondary side portion), equation 6. The measurements and calculations are presented schematically on Figure 2.

For the measurement of feedwater flow, each feedwater venturi is calibrated by the vendor in a hydraulics laboratory under controlled conditions to an accuracy of [ ]$^{+a,c}$. The calibration data which substantiates this accuracy is provided to the plant by the vendor. An additional uncertainty factor of [ ]$^{+a,c}$ is included for installation effects, resulting in a conservative overall flow coefficient (K) uncertainty of [ ]$^{+a,c}$. Since the calculated steam generator thermal output is proportional to feedwater flow, the flow coefficient uncertainty is expressed as [ ]$^{+a,c}$. It should be noted that no allowance is made for feedwater venturi fouling. The effect of fouling results in an indicated power higher than actual, which is conservative.

The uncertainty applied to the feedwater venturi thermal expansion correction ($F_a$) is based on the uncertainties of the measured feedwater temperature and the coefficient of thermal expansion for the venturi material, 304 stainless steel. For this material, a change of $\pm$ 1.0 °F in the nominal feedwater temperature range changes $F_a$ by [ ]$^{+a,c}$ and the steam generator thermal output by the same amount.

Based on data introduced into the ASME Code, the uncertainty in $F_a$ for 304 stainless steel is $\pm$ 5 %. This results in an additional uncertainty of [ ]$^{+a,c}$ in power.

Using the NBSNRC Steam Tables it is possible to determine the sensitivities of various parameters to changes in feedwater temperature and pressure. Table 7 notes the instrument uncertainties for the hardware used to perform the measurements. Table 8 lists the various sensitivities. As can be seen on Table 8, feedwater temperature uncertainties have an impact on venturi $F_a$, feedwater density and feedwater enthalpy. Feedwater pressure uncertainties impact feedwater density and feedwater enthalpy.

Feedwater venturi d/p uncertainties are converted to % feedwater flow using the following conversion factor:

$$\% \text{ flow} = (\text{d/p uncertainty})(1/2)(\text{FLOWmax/FLOWnominal})^2.$$

The feedwater flow transmitter span (FLOWmax) is 120.0% of nominal flow.

Since it is necessary to make this determination daily, the plant computer is used for the calorimetric power measurement. As noted in Table 9, Westinghouse has determined the dependent sets in the calculation and the direction of interaction. This is the same as that performed for the calorimetric RCS flow measurement, but applicable only to power. The same was performed for the bias values.

Using the power uncertainty values noted on Table 9, the 2 loop uncertainty (with bias values) equation is as follows:

[ ]$^{+a,c}$

Based on the number of loops and the instrument uncertainties for the four parameters, the uncertainty for the secondary side power calorimetric measurement is:

| # of loops | power uncertainty (% RTP) |
|---|---|
| 2 | [ ] +a,c |

TABLE 7

POWER CALORIMETRIC INSTRUMENTATION UNCERTAINTIES

(% SPAN)

| | FW TEMP | FW PRES | FW D/P | STM PRESS |
|---|---|---|---|---|
| SRA = | [ | | | ] +a,c |
| SCA = | | | | |
| SMTE = | | | | |
| SPE = | | | | |
| STE = | | | | |
| SD = | | | | |
| BIAS = | | | | |
| RCA = | | | | |
| RMTE = | | | | |
| RTE = | | | | |
| RD = | | | | |
| A/D = | | | | |
| CSA = | | | | |
| # OF INSTRUMENTS USED | 1/Loop | 1/Loop | 1/Loop | 1/Loop |
| | °F | psi | % d/p | psi |
| INST SPAN = | 150 | 1600 | 120% Flow | 1400 |
| INST UNC. (RANDOM) = | [ | | | ] +a,c |
| INST UNC. (BIAS) = | | | | |
| NOMINAL = | 440.7 °F | 800-875 psia | 100 % Flow | 700-775 psia |

(a) Included in RCA

Feedwater temperature measurement is from channels T-2104 and -2105
Feedwater pressure measurement is from channels P-2289 and -2290
Feedwater flow measurement is from channels F-466, -467, -476 and -477
Steam pressure measurement is from channels P-468, -469, -478, -479, -482 and -483.

TABLE 8

POWER CALORIMETRIC SENSITIVITIES

+a,c

| | | |
|---|---|---|
| FEEDWATER FLOW | | |
| $F_a$ | | |
| TEMPERATURE | = | |
| MATERIAL | = | |
| DENSITY | | |
| TEMPERATURE | = | |
| PRESSURE | = | |
| DELTA P | = | |
| FEEDWATER ENTHALPY | | |
| TEMPERATURE | = | |
| PRESSURE | = | |
| $h_s$ | = | |
| $h_f$ | = | |
| Dh (SG) | = | |
| STEAM ENTHALPY | | |
| PRESSURE | = | |
| MOISTURE | = | |

TABLE 9

SECONDARY SIDE POWER CALORIMETRIC MEASUREMENT UNCERTAINTY

| COMPONENT | INSTRUMENT UNCERTAINTY | POWER UNCERTAINTY |
|---|---|---|
| FEEDWATER FLOW | | +a,c |
| VENTURI | | |
| THERMAL EXPANSION COEFFICIENT | | |
| TEMPERATURE | | |
| MATERIAL | | |
| DENSITY | | |
| TEMPERATURE | | |
| PRESSURE | | |
| DELTA P | | |
| FEEDWATER ENTHALPY | | |
| TEMPERATURE | | |
| PRESSURE | | |
| STEAM ENTHALPY | | |
| PRESSURE | | |
| MOISTURE | | |
| NET PUMP HEAT ADDITION | | |
| BIAS VALUES | | |
| FEEDWATER DELTA P | | |
| FEEDWATER PRESSURE | DENSITY | |
| | ENTHALPY | |
| STEAM PRESSURE | ENTHALPY | |
| POWER BIAS TOTAL VALUE | | |
| SINGLE LOOP UNCERTAINTY (WITHOUT BIAS) | | |
| 2 LOOP UNCERTAINTY (WITHOUT BIAS) | | |
| 2 LOOP UNCERTAINTY (WITH BIAS VALUES) | | |

*, * *, INDICATES SETS OF DEPENDENT PARAMETERS

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## IV. RESULTS/CONCLUSIONS

The preceding sections provide the methodology to account for pressure, temperature, power and RCS flow uncertainties for the RTDP analysis. The uncertainty calculations have been performed for Point Beach Units 1 & 2 with the plant specific instrumentation and calibration procedures. The following table summarizes the results and the uncertainties that are used in the Point Beach 1 & 2 safety analysis.

| Parameter | Calculated Uncertainty | Uncertainty Used in Safety Analysis |
|---|---|---|
| Pressurizer Pressure | ±31.1 psi (random)<br>0.0 psi (bias) | ±50.0 psi (random) |
| Tavg | ±4.4 °F (random)<br>-1.2 °F (bias) | ±6.0 °F (random)<br>(includes bias) |
| Power | ±1.9% RTP (random) | ±2.0% RTP (random) |
| RCS Flow<br>(plant computer)<br><br>(calorimetric measurement) | ±2.08% flow (random)<br>+0.26% flow (bias)<br>±1.9% flow (random)<br>+0.26% flow (bias) | ±2.4% flow (random)<br>(includes bias) |

## REFERENCES


1. Westinghouse WCAP-11397-P-A, "Revised Thermal Design Procedure", dated April 1989.

2. Westinghouse letter NS-CE-1583, C. Eicheldinger to J. F. Stolz, NRC, dated 10/25/77.

3. Westinghouse letter NS-PLC-5111, T. M. Anderson to E. Case, NRC, dated 5/30/78.

4. Westinghouse letter NS-TMA-1837, T. M. Anderson to S. Varga, NRC, dated 6/23/78.

5. Westinghouse letter NS-EPR-2577, E. P. Rahe Jr. to C. H. Berlinger, NRC, dated 3/31/82.

6. Westinghouse Letter NS-TMA-1835, T. M. Anderson to E. Case, NRC, dated 6/22/78.

7. NRC letter, S. A. Varga to J. Dolan, Indiana and Michigan Electric Company, dated 2/12/81.

8. NUREG-0717 Supplement No. 4, Safety Evaluation Report related to the operation of Virgil C. Summer Nuclear Station Unit No. 1, Docket 50-395, August, 1982.

9. Regulatory Guide 1.105 Rev. 2, "Instrument Setpoints for Safety-Related Systems", dated 2/86.

10. NUREG/CR-3659 (PNL-4973), "A Mathematical Model for Assessing the Uncertainties of Instrumentation Measurements for Power and Flow of PWR Reactors", 2/85.

11. ANSI/ANS Standard 58.4-1979, "Criteria for Technical Specifications for Nuclear Power Stations".

12. ISA Standard S67.04, 1994, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants".

13. Tuley, C. R., Williams T. P., "The Significance of Verifying the SAMA PMC 20.1-1973 Defined Reference Accuracy for the Westinghouse Setpoint Methodology", Instrumentation, Controls, and Automation in the Power Industry, June 1992, Vol.35, pp. 497-508.

14. Scientific Apparatus Manufacturers Association, Standard PMC 20.1, 1973, "Process Measurement and Control Terminology".

15. WCAP-11397-P-A, "Revised Thermal Design Procedure", dated April 1989.

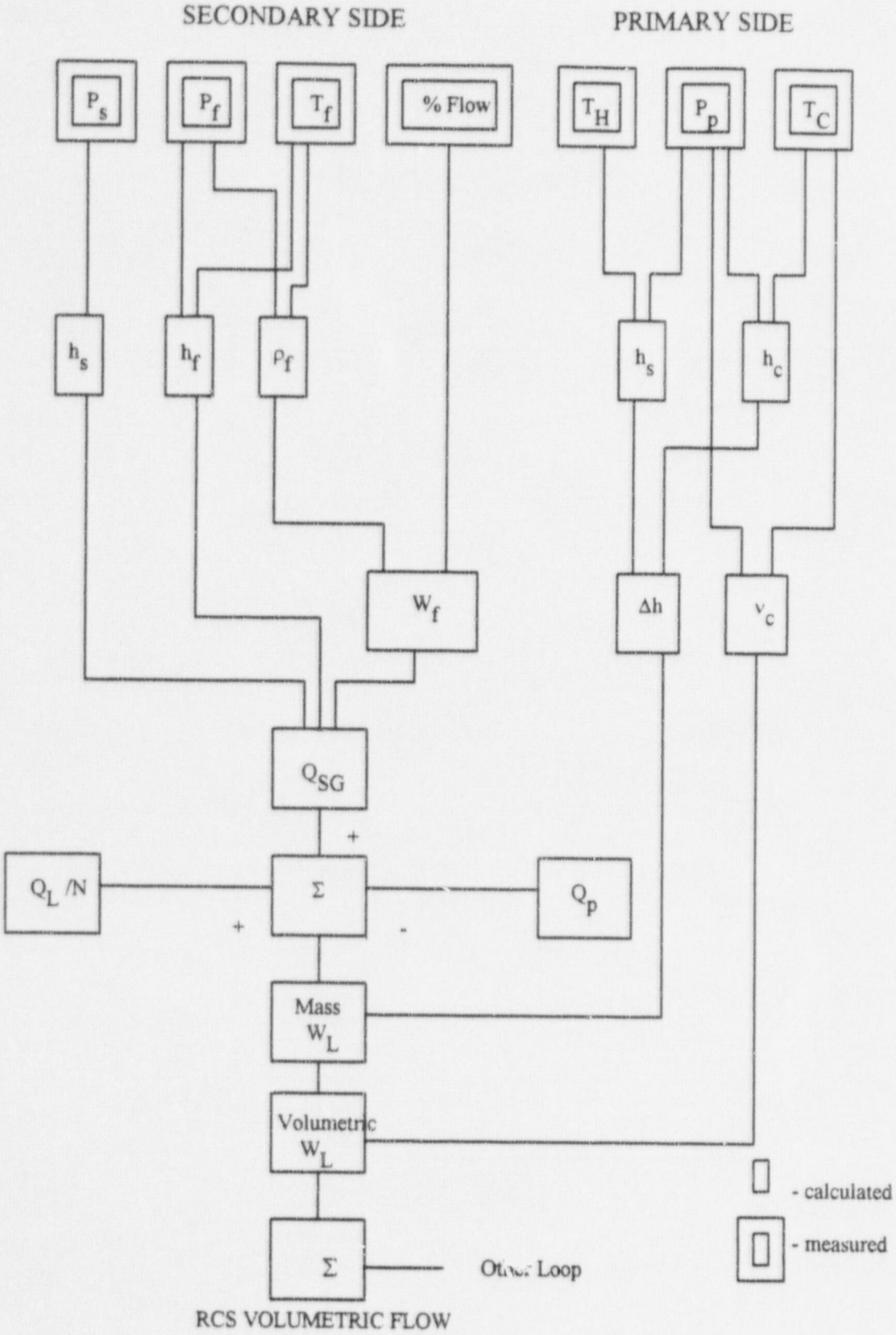



Figure 1
Calorimetric RCS Flow Measurement (Using LEFM)

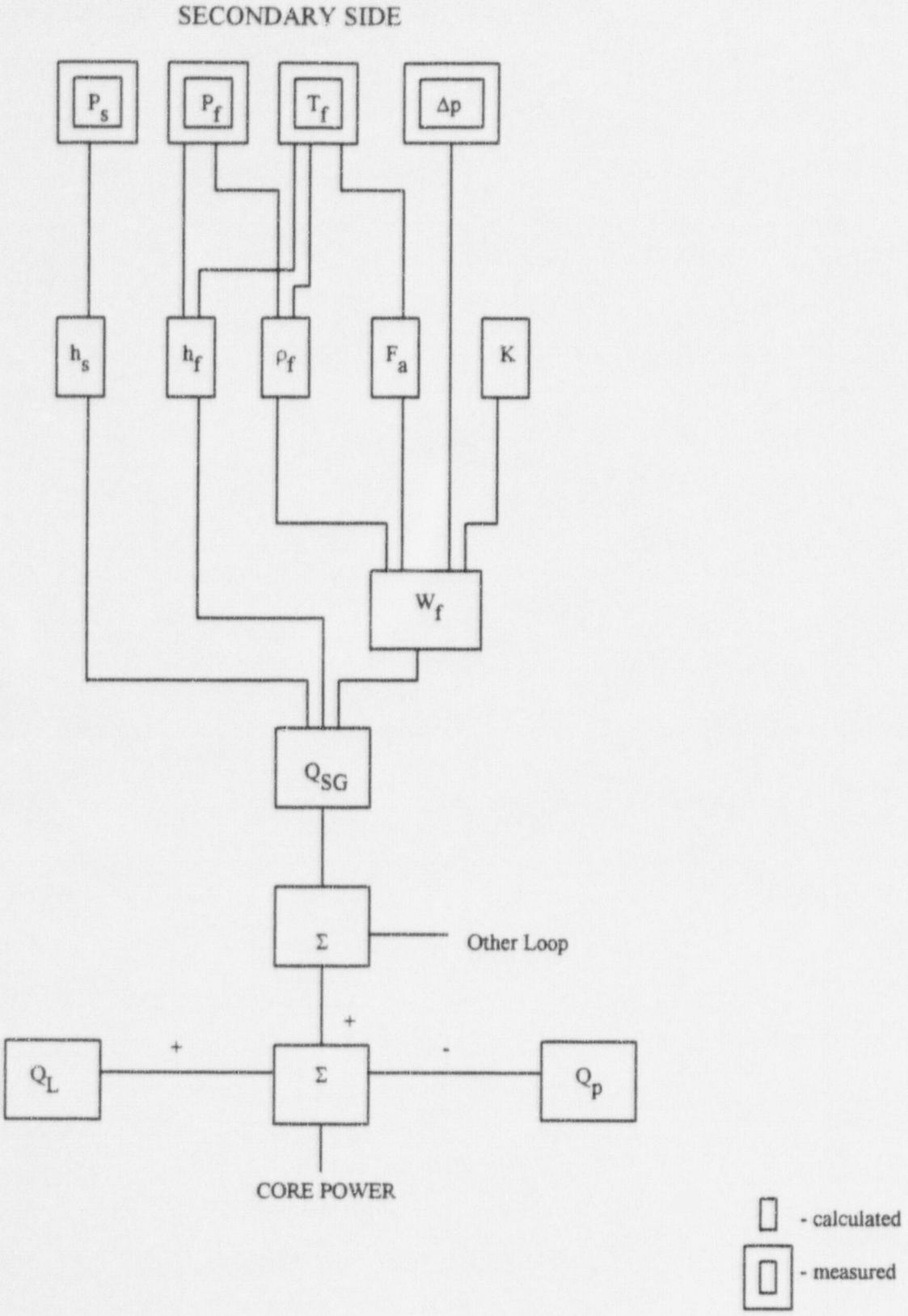



Figure 2
Calorimetric Power Measurement (Using Feedwater Venturi)

**Attachment 6 – Marked up Technical Specifications for implementation of the 422V+ fuel assemblies at Point Beach Nuclear Plant Units 1 and 2**

Included in this attachment are the marked-up Technical Specifications indicating the proposed changes necessary for incorporation of the 422V+ fuel assemblies at PBNP. These changes are discussed in detail in Attachment 1 of this submittal. Also included in this attachment is a clean copy of the Technical Specification pages with the changes incorporated.

## 15.2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

### 15.2.1 SAFETY LIMIT, REACTOR CORE

Applicability:

Applies to the limiting combinations of thermal power, reactor coolant system pressure, and coolant temperature during operation.

Objective:

To maintain the integrity of the fuel cladding.

Specification:

or Figure 15.2.1-2 as applicable

1. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 15.2.1-1 for Units 1 and 2.* The safety limit is exceeded if the point defined by the combination of reactor coolant system average temperature and power level is at any time above the appropriate pressure line.

Basis:

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excess cladding temperature because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore thermal power and Reactor Coolant temperature and pressure have been related to DNB.

* Figure 15.2.1-1 applies to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, Figure 15.2.1-2 applies to Unit 1.

This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

and 14x14 upgraded OFA fuel assemblies

The DNB design basis is as follows: there must be at least a 95 percent probability at a 95 percent confidence level that DNB will not occur during steady state operation, normal operational transients, and anticipated transients and is an appropriate margin to DNB for all operating conditions.

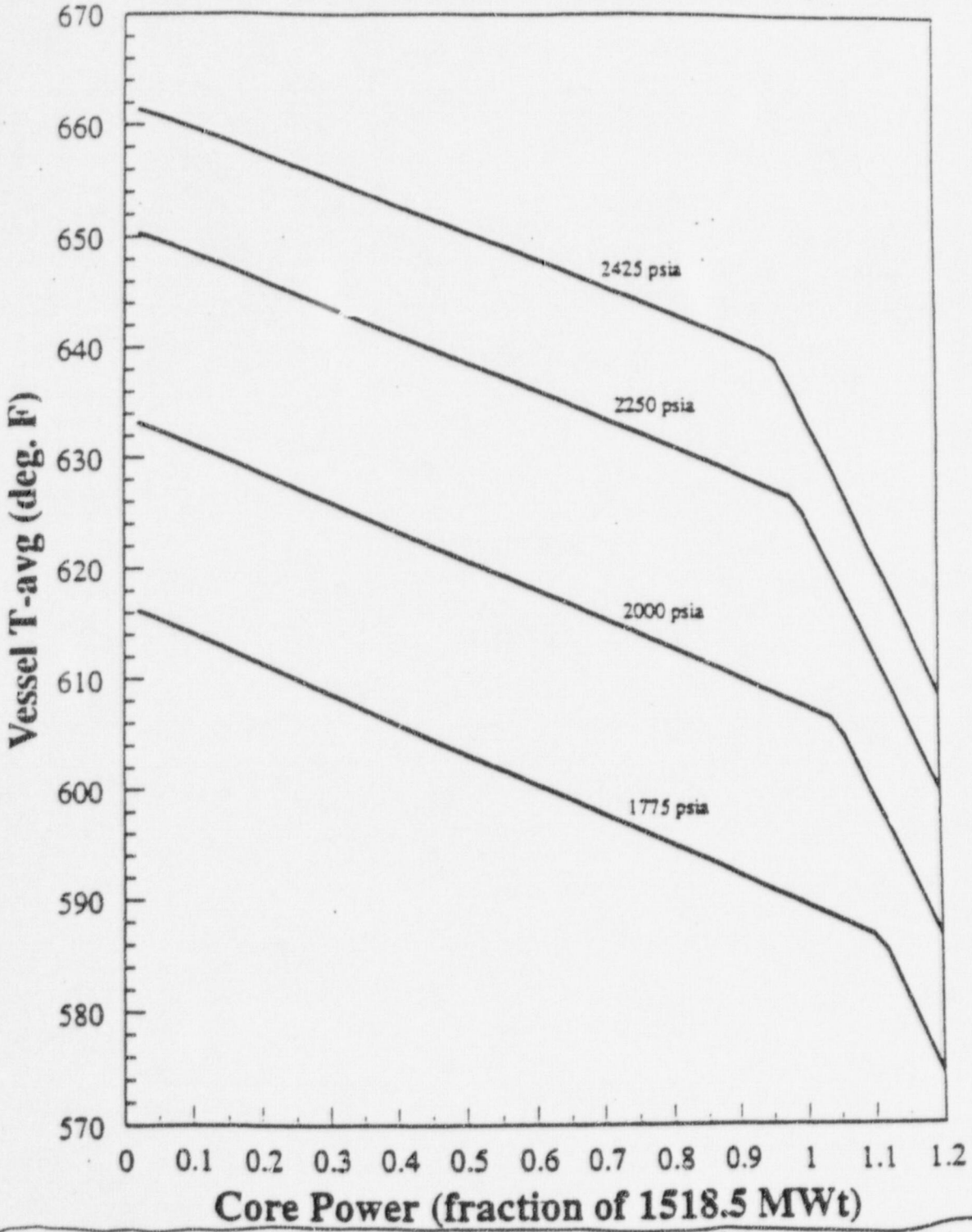
with any combination

family of in is to

The curves of Figure 15.2.1-1 and 15.2.1-2 are applicable for a core of 14 x 14 OFA. The curves also apply to the reinsertion of previously-depleted 14 x 14 standard fuel assemblies into an OFA core. The use of these assemblies is justified by a cycle-specific reload analysis. The WRB-1 correlation is used to generate these curves. Uncertainties in plant parameters and DNB correlation predictions are statistically convoluted to obtain a DNBR uncertainty factor. This DNBR uncertainty factor establishes a value of design limit DNBR. This value of design limit DNBR is shown to be met in plant safety analyses, using values of input parameters considered at their nominal values.

The family of curves in Figure 15.2.1-2 is applicable to any combination of 422V+ fuel assemblies, burned 14x14 OFA fuel assemblies and burned 14x14 upgraded OFA fuel assemblies, or a full core of 422V+ fuel assemblies.

Figure 15.2.1-1 *
POINT BEACH NUCLEAR PLANT UNITS 1 AND 2*
REACTOR CORE SAFETY LIMITS



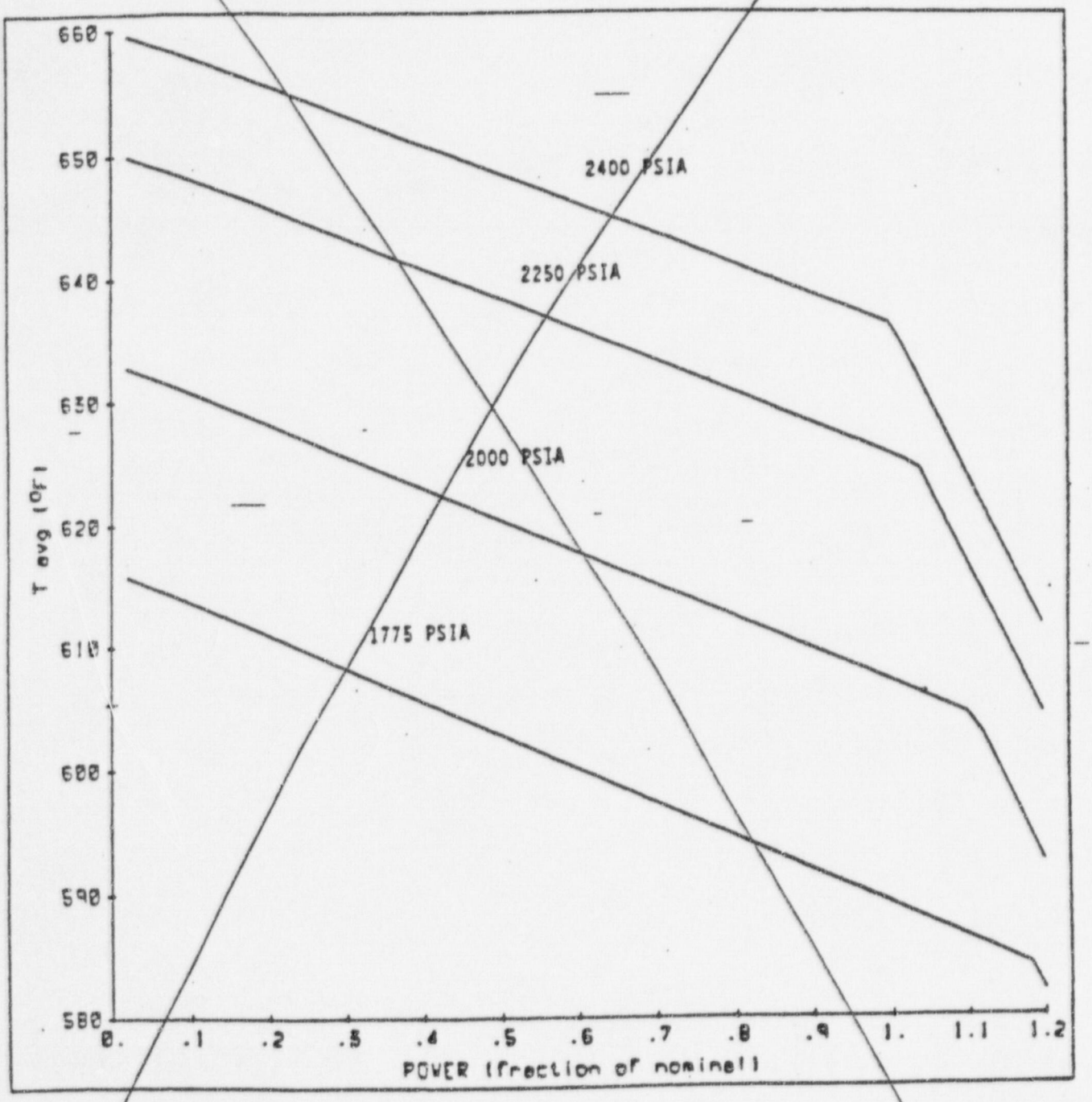

* This figure applies to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, Figure 15.2.1-2 applies to Unit 1.



* This figure applies to core reloads with any combination of OFA and upgraded CFA fuel assemblies

Replace with Insert 1

Figure 15.2.1-2*
REACTOR CORE SAFETY LIMITS
POINT BEACH UNIT 1

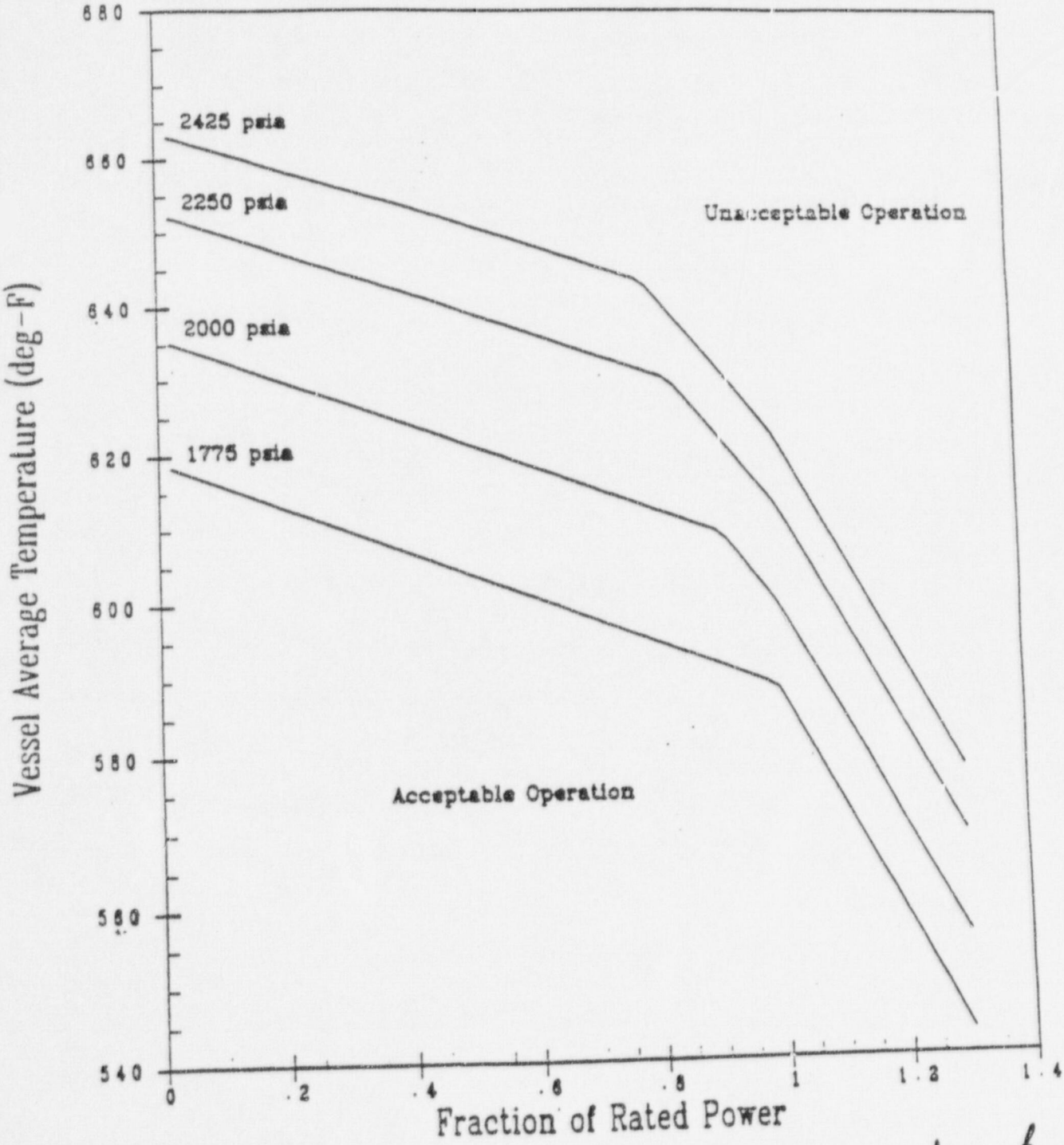



* This figure applies to Unit 1 prior to U1R24. Following U1R24, Figure 15.2.1-1 applies to Unit 1.

Insert 1

Figure 15.2.1-2 *

Point Beach Nuclear Plants Units 1 and 2
Reactor Core Safety Limit
Two Loops in Operation




* This figure applies to core reloads with any combination of 422V+ fuel assemblies, burned OFA and burned upgraded OFA fuel assemblies, or a full core of 422V+ fuel assemblies

## 15.2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability:

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, pressurizer level, and permissives related to reactor protection. _____

Objective:
To provide for automatic protective action in the event that the principal process variables approach a safety limit.

Specification:

1. Protective instrumentation for reactor trip settings shall be as follows:

   A. Startup protection

      (1) High flux, source range - within span of source range instrumentation.

      (2) High flux, intermediate range - ≤40% of rated power.

      (3) High flux, power range (low setpoint) - ≤25% of rated power.

   B. Core limit protection

      (1) High flux, power range (high setpoint) - ≤108% of rated power.

      (2) High pressurizer pressure* - ≤2385 psig for operation at 2250 psia primary system pressure
      ≤2210 psig for operation at 2000 psia primary system pressure

      and cores not containing 422V+ fuel assemblies

* These values apply to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, the high pressurizer pressure reactor trip setpoint for Unit 1 is ≤2385 psig.

(3) Low pressurizer pressure* - ≥1905 psig for operation at 2250 psia primary system pressure
≥1800 psig for operation at 2000 psia primary system pressure

(4) Overtemperature $\Delta T \left(\frac{1}{1+\tau_3 S}\right)$ replace with Insert 2a through 2c

$$\leq \Delta T_o \left(K_1 - K_2\left(T\left(\frac{1}{1+\tau_4 S}\right) - T'\right)\left(\frac{1+\tau_1 S}{1+\tau_2 S}\right) + K_3(P - P') - f(\Delta I)\right)$$

where (values are applicable to operation at both 2000 psia and 2250 psia unless otherwise indicated)

| | | |
|---|---|---|
| $\Delta T_o$ | = | indicated ΔT at rated power, °F |
| T | = | average temperature, °F |
| T' | ≤ | 572.9°F** |
| P | = | pressurizer pressure, psig |
| P' | = | 2235 psig (2250 psia operation only) |
| P' | = | 1985 psig (2000 psia operation only)** |
| $K_1$ | ≤ | 1.19 (2250 psia operation only) |
| $K_1$ | ≤ | 1.14 (2000 psia operation only)** |
| $K_2$ | = | 0.025 (2250 psia operation only) |
| $K_2$ | = | 0.022 (2000 psia operation only)** |
| $K_3$ | = | 0.0013 (2250 psia operation only) |
| $K_3$ | = | 0.001 (2000 psia operation only)** |
| $\tau_1$ | = | 25 sec |
| $\tau_2$ | = | 3 sec |
| $\tau_3$ | = | 2 sec for Rosemont or equivalent RTD |
| | = | 0 sec for Sostman or equivalent RTD |
| $\tau_4$ | = | 2 sec for Rosemont or equivalent RTD |
| | = | 0 sec for Sostman or equivalent RTD |

and f(ΔI) is an even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where $q_t$ and $q_b$ are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, such that:

(a) for $q_t - q_b$ within -17, +5 percent, f(ΔI) = 0.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +5 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power.

* These values apply to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, the low pressurizer pressure reactor trip setpoint for Unit 1 is ≥1790 psig.

** These values apply to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, the values are: T' ≤ 573.9°F, P' = 2235 psig, $K_1 \leq 1.30$, $K_2 = 0.0200$, and $K_3 = 0.000791$.

replace with Insert 2a through 2c

(c) for each percent that the magnitude of $q_t - q_b$ exceeds -17 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power.

(5) Overpower $\Delta T \left(\frac{1}{1+\tau_3 S}\right)$

$$\leq \Delta T_o\left[K_4 - K_5\left(\frac{\tau_5 S}{\tau_5 S+1}\right)\left(\frac{1}{1+\tau_4 S}\right)T - K_6\left[T\left(\frac{1}{1+\tau_4 S}\right) - T'\right]\right]$$

where (values are applicable to operation at both 2000 psia and 2250 psia)

| | | |
|---|---|---|
| $\Delta T_o$ | = | indicated ΔT at rated power, °F |
| T | = | average temperature, °F |
| T' | ≤ | 572.9°F* |
| $K_4$ | ≤ | 1.09 of rated power* |
| $K_5$ | = | 0.0262 for increasing T |
| | = | 0.0 for decreasing T |
| $K_6$ | = | 0.00123 for T ≥ T' |
| | = | 0.0 for T < T' |
| $\tau_5$ | = | 10 sec |
| $\tau_3$ | = | 2 sec for Rosemont or equivalent RTD |
| | | 0 sec for Sostman or equivalent RTD |
| $\tau_4$ | = | 2 sec for Rosemont or equivalent RTD |
| | | 0 sec for Sostman or equivalent RTD |

(6) Undervoltage - ≥ 3120 V.

(7) Indicated reactor coolant flow per loop - ≥90 percent of normal indicated loop flow

(8) Reactor coolant pump motor breaker open

(a) Low frequency set point ≥55.0 HZ

(b) Low voltage set point ≥ 3120 V.

* These values apply to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, the values for Unit 1 are: T' ≤ 573.9°F and $K_4$ ≤ 1.089 of rated power.

INSERT 2a

(3)

Low pressurizer pressure* - ≥1905 psig for operation at 2250 psia primary system pressure
- ≥1800 psig for operation at 2000 psia primary system pressure and cores not containing 422V+ fuel assemblies

~~* These values apply to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, the low pressurizer pressure reactor trip setpoint for Unit 1 is ≥1790 psig.~~

TS 15.2.3.1.B(4)

Overtemperature

$$\Delta T \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_o \left(K_1 - K_2\left(T\left(\frac{1}{1+\tau_4 S}\right) - T'\right)\left(\frac{1+\tau_1 S}{1+\tau_2 S}\right) + K_3(P - P') - f(\Delta I)\right)$$

where (values are applicable to operation at both 2000 psia and 2250 psia unless otherwise indicated)

| | | |
|---|---|---|
| $\Delta T_o$ | = | indicated ΔT at rated power, °F |
| T | = | average temperature, °F |
| T' | ≤ | 569.0°F (for cores containing 422V+ fuel assemblies) |
| T' | ≤ | 572.9°F~~**~~ (for cores not containing 422V+ fuel assemblies) |
| P | = | pressurizer pressure, psig |
| P' | = | 2235 psig (for 2250 psia operation ~~only~~) |
| P' | = | 1985 psig (for 2000 psia operation and cores not containing 422V+ fuel assemblies ~~only)**~~ |
| $K_1$ | ≤ | 1.16 (for 2250 psia operation and cores containing 422V+ fuel assemblies) |
| $K_1$ | ≤ | 1.19 (for 2250 psia operation and cores not containing 422V+ fuel assemblies ~~only~~) |
| $K_1$ | ≤ | 1.14 (for 2000 psia operation and cores not containing 422V+ fuel assemblies ~~only)**~~ |
| $K_2$ | = | 0.0149 (for 2250 psia operation and cores containing 422V+ fuel assemblies) |
| $K_2$ | = | 0.025 (for 2250 psia operation and cores not containing 422V+ fuel assemblies ~~only~~) |
| $K_2$ | = | 0.022 (for 2000 psia operation and cores not containing 422V+ fuel assemblies ~~only)**~~ |
| $K_3$ | = | 0.00072 (for 2250 psia operation and cores containing 422V+ fuel assemblies) |
| $K_3$ | = | 0.0013 (for 2250 psia operation and cores not containing 422V+ fuel assemblies ~~only~~) |
| $K_3$ | = | 0.001 (for 2000 psia operation and cores not containing 422V+ fuel assemblies ~~only)**~~ |
| $\tau_1$ | = | 25 sec |
| $\tau_2$ | = | 3 sec |
| $\tau_3$ | = | 2 sec for Rosemont or equivalent RTD |

Insert 2b

| | | |
|---|---|---|
| | = | 0 sec for Sostman or equivalent RTD |
| $\tau_4$ | = | 2 sec for Rosemont or equivalent RTD |
| | = | 0 sec for Sostman or equivalent RTD |

and f(ΔI) is an even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where $q_t$ and $q_b$ are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, such that:

(a) for $q_t - q_b$ within –17 , +5 percent, f(ΔI) = 0 for cores not containing 422V+ fuel assemblies; for $q_t - q_b$ within –12, +5 percent, f(ΔI) = 0 for cores containing 422V+ fuel assemblies.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +5 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power for cores not containing 422V+ fuel assemblies and reduced by an equivalent of 2.12 percent of rated power for cores containing 422V+ fuel assemblies.

(c) for cores not containing 422V+ fuel assemblies, for each percent that the magnitude of $q_t - q_b$ exceeds –17 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power; for cores containing 422V+ fuel assemblies, for each percent that the magnitude of $q_t - q_b$ exceeds –12 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power.

~~** These values apply to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, the values are: T' ≤ 573.9°F, P' = 2235 psig, $K_1$ ≤ 1.30, $K_2$ = 0.0200, and $K_3$ = 0.000791.~~

TS 15.2.3.1.B(5)
Overpower

$$\Delta T\left(\frac{1}{1+\tau_3 S}\right) \le \Delta T_o\left[K_4 - K_5\left(\frac{\tau_5 S}{\tau_5 S+1}\right)\left(\frac{1}{1+\tau_4 S}\right)T - K_6\left[T\left(\frac{1}{1+\tau_4 S}\right) - T'\right]\right]$$

where (values are applicable to operation at both 2000 psia and 2250 psia)

| | | |
|---|---|---|
| $\Delta T_o$ | = | indicated ΔT at rated power, °F |
| T | = | average temperature, °F |
| T' | ≤ | 569.0°F (for cores containing 422V+ fuel assemblies) |
| T' | ≤ | 572.9°F~~*~~ (for cores not containing 422V+ fuel assemblies) |
| $K_4$ | ≤ | 1.10 of rated power (for cores containing 422V+ fuel assemblies) |
| $K_4$ | ≤ | 1.09 of rated power~~*~~ (for cores not containing 422V+ fuel assemblies) |
| $K_5$ | = | 0.0262 for increasing T |
| | = | 0.0 for decreasing T |
| $K_6$ | = | 0.00103 for T ≥ T' (for cores containing 422V+ fuel assemblies) |

Insert 2c

$K_6$ = 0.00123 for $T \geq T'$ (for cores not containing 422V+ fuel assemblies)
= 0.0 for $T < T'$

$\tau_5$ = 10 sec

$\tau_3$ = 2 sec for Rosemont or equivalent RTD
0 sec for Sostman or equivalent RTD

$\tau_4$ = 2 sec for Rosemont or equivalent RTD
0 sec for Sostman or equivalent RTD

(6) Undervoltage - ≥3120V

(7) Indicated reactor coolant flow per loop ≥90 percent of normal indicated loop flow

(8) Reactor coolant pump motor breaker open
- (a) Low frequency set point ≥55.0 HZ
- (b) Low voltage set point ≥3120V

~~* These values apply to Unit 2 following U2R22 and to Unit 1 following U1R24. Prior to U1R24, the values for Unit 1 are: $T' \leq 573.9°F$ and $K_4 \leq 1.089$ of rated power.~~

C.

Other reactor trips:

(1) High pressurizer water level - ≤95% of span

(2) Low-low steam generator water level -
≥20% of narrow range instrument span
≥5% of narrow range instrument span (Unit 1)*

(3) Steam-Feedwater Flow Mismatch Trip - ≤1.0 x $10^6$ lb/hr

(4) Turbine Trip (Not a protection circuit)

(5) Safety Injection Signal

(6) Manual Trip

* This setting limit applies to Unit 1 until the narrow range lower tap is changed to the lower position consistent with Unit 2.

Basis

The source range high flux reactor trip prevents a startup accident from subcritical conditions from proceeding into the power range. Any setpoint within its range would prevent an excursion from proceeding to the point at which significant thermal power is generated.[1]

The high flux low power reactor trip provides redundant protection in the power range for a power excursion beginning from low power. This trip insures that a more restrictive trip point is used for this case than for an excursion beginning from near full power.[1]

The overpower nuclear flux reactor trip protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure circuitry. The prescribed setpoint, with allowance for errors, is consistent with the trip point assumed in the accident analysis.[3]

118%

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 108% of design power density, and includes corrections for change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.[2]

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds)[5], and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial

power distribution, the reactor trip limit, with allowance for errors[2], is always below the core safety limit as shown on Figures 15.2.1-1 and 15.2.1-2. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced[6][7].

(for cores containing 422V+ fuel)

for OFA and upgraded OFA fuel

The overpower, overtemperature and pressurizer pressure system setpoints include the effect of reduced system pressure operation (including the effects of fuel densification). The setpoints will not exceed the core safety limits as shown in Figures 15.2.1-1 and 15.2.1-2.

(for OFA and upgraded OFA fuel only cores)

The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip setting is lower than the set pressure for the safety valves (2485 psig) such that the reactor is tripped before the safety valves actuate. The low pressurizer pressure reactor trip trips the reactor in the unlikely event of a loss-of-coolant accident[4].

The low flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis[8]. The low loop flow signal is caused by a condition of less than 90 percent flow as measured by the loop flow instrumentation. The loss of power signal is caused by the reactor coolant pump breaker opening

The setpoints for 422V+ fuel do not include the effect of reduced system pressure operation, therefore, cores containing 422V+ fuel must be operated at 2250 psia

as actuated by either high current, low supply voltage or low electrical frequency, or by a manual control switch. The significant feature of the breaker trip is the frequency setpoint, 55.0 HZ, which assures a trip signal before the pump inertia is reduced to an unacceptable value. The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. The specified setpoint allows adequate operating instrument error[2] and transient overshoot in level before the reactor trips.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified setpoint assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system.[9]

Numerous reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed setpoint above which these trips are unblocked assures their availability in the power range where needed. Specifications 15.2.3.2.A(1) and 15.2.3.2.C have ±1% tolerance to allow for a 2% deadband of the P10 bistable which is used to set the limit of both items. The difference between the nominal and maximum allowed value (or minimum allowed value) is to account for "as measured" rack drift effects.

Sustained operation is not permitted with only one reactor coolant pump. If a pump is lost while operating below 50 percent power, an orderly shutdown is allowed. The power-to-flow ratio will be maintained equal to or less than unity, which ensures that the minimum DNB ratio increases at lower flow because the maximum enthalpy rise does not increase above the maximum enthalpy rise which occurs during full power and full flow operation.


References

(1) FSAR 14.1.1
(2) FSAR, Page 14-5
(3) FSAR 14.2.6
(4) FSAR 14.3.1
(5) FSAR 14.1.2
(6) FSAR 7.2, 7.7
(7) FSAR 3.2.1
(8) FSAR 14.1.9
(9) FSAR 14.1.11


14.0

14.1.10 and

Replace with Insert 3

G. OPERATIONAL LIMITATIONS

The following DNB related parameters shall be maintained within the limits shown during Rated Power operation:

1. $T_{avg}$ shall be maintained ≥557°F and ≤573.9°F.

2. Reactor Coolant System (RCS) pressurizer pressure shall be maintained:

   ≥2205 psig during operation at 2250 psia, or
   ≥1955 psig during operation at 2000 psia.

3. Reactor Coolant System raw measured Total Flow Rate shall be maintained ≥181,800 gpm.

Basis:

The reactor coolant system total flow rate of 181,800 gpm is based on an assumed measurement uncertainty of 2.1 percent over thermal design flow (178,000 gpm). The raw measured flow is based upon the use of normalized elbow tap differential pressure which is calibrated against a precision flow calorimetric at the beginning of each cycle.

Insert 3

G. Operational Limitations

The following DNB related parameters shall be maintained within the limits shown during rated power operation:

1. $T_{avg}$ shall be maintained ≥558.1°F and ≤574.0°F for cores containing 422V+ fuel assemblies. $T_{avg}$ shall be maintained ≥557°F and ≤573.9°F for cores not containing 422V+ fuel assemblies.
2. Reactor Coolant System (RCS) pressurizer pressure shall be maintained:
   ≥2205 psig during operation at 2250 psia, or
   ≥1955 psig during operation at 2000 psia for cores not containing 422V+ fuel assemblies.
3. Reactor Coolant System raw measured Total Flow Rate shall be maintained
   ≥182,400 gpm for cores containing 422V+ fuel assemblies, or ≥181,800 gpm for cores not containing 422V+ fuel assemblies.

TS 15.3.1 Basis:

The reactor coolant system total flow rate of 182,400 gpm for cores containing 422V+ fuel assemblies is based on an assumed measurement uncertainty of 2.4 percent over thermal design flow (178,000 gpm). The reactor coolant system total flow rate of 181,800 gpm for cores not containing 422V+ fuel assemblies is based on an assumed measurement uncertainty of 2.1 percent over thermal design flow (178,000 gpm). The raw measured flow is based upon the use of normalized elbow tap differential pressure which is calibrated against a precision flow calorimetric at the beginning of each cycle.

TABLE 15.3.5-1
(PAGE 1 OF 2)
ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

| NO. | FUNCTIONAL UNIT | CHANNEL | SETTING LIMIT |
|---|---|---|---|
| 1 | High Containment Pressure (Hi) | Safety Injection* | ≤ 6 psig |
| 2 | High Containment Pressure (Hi-Hi) | a. Containment Spray<br>b. Steam Line Isolation of Both Lines | ≤ 30 psig<br>≤ 20 psig |
| 3 | Pressurizer Low Pressure | Safety Injection* | ≥ 1715 psig |
| 4 | Low Steam Line Pressure | Safety Injection*<br>Lead Time Constant<br>Lag Time Constant | ≥ 500 psig<br>> 12 seconds<br>≤ 2 seconds |
| 5 | High Steam Flow in a Steam Line Coincident with Safety Injection and Low $T_{AVG}$ | Steam Line Isolation of Affected Line | ≤ d/p corresponding to $0.66 \times 10^6$ lb/hr at 1005 psig<br>≥ 540°F |
| 6 | High-high Steam Flow in a Steam Line Coincident with Safety Injection | Steam Line Isolation of Affected Line | ≤ d/p corresponding to $4 \times 10^6$ lb/hr at 806 psig |
| 7 | Low-low Steam Generator Water Level | Auxiliary Feedwater Initiation | ≥20% of narrow range instrument<br>≥5% of narrow range instrument (Unit 1)** |
| 8 | Undervoltage on 4 KV Busses | Auxiliary Feedwater Initiation | ≥ 3120 V |

* Initiates also containment isolation, feedwater line isolation and starting of all containment fans.
** This setting limit applies to Unit 1 until the narrow range lower tap is changed to the lower position consistent with Unit 2.
d/p means differential pressure

AND

b. Within two hours fully withdraw the shutdown banks.

c. If the above actions and associated completion times are not met, be in hot shutdown within the following six hours.

2. When the reactor is critical, the control banks shall be inserted no further than the limits shown by the lines on Figure 15.3.10-1. If this condition is not met, perform the following actions:

a. Within one hour verify that the shutdown margin exceeds the applicable value as shown in Figure 15.3.10-2; OR within one hour restore the shutdown margin by boration;

AND

b. Within two hours restore the control banks to within limits.

c. If the above actions and associated completion times are not met, be in hot shutdown within the following six hours.

E. POWER DISTRIBUTION LIMITS

1. Hot Channel Factors

a. The hot channel factors defined in the basis shall meet the following limits:

| For OFA and Upgraded OFA Fuel | | For 422V+ Fuel |
|---|---|---|
| $F_Q(Z) \leq \frac{(2.50)}{P} \times K(Z)$ | for $P > 0.5$ | $F_Q(Z) \leq \frac{(2.60)}{P} \times K(Z)$ |
| $F_Q(Z) \leq 5.00 \times K(Z)$ | for $P \leq 0.5$ | $F_Q(Z) \leq 5.20 \times K(Z)$ |
| $F^N_{\Delta H} < 1.70 \times [1 + 0.3\,(1-P)]$ | | $F^N_{\Delta H} < 1.77 \times [1 + 0.3(1-P)]$ |

Where P is the fraction of full power at which the core is operating, K(Z) is the function in Figure 15.3.10-3 or Figure 15.3.10-3a, as applicable, and Z is the core height location of $F_Q$.

b. If $F_Q(Z)$ exceeds the limit of Specification 15.3.10.E.1.a, within fifteen minutes reduce thermal power until $F_Q(Z)$ limits are satisfied;

(1) After thermal power has been reduced in accordance with Specification 15.3.10.E.1.b, perform the following actions:

Power Distribution — Insert Attachment A

During power operation, the global power distribution is limited by TS 15.3.10.E.2, "Axial Flux Difference," and TS 15.3.10.E.3, "Quadrant Power Tilt," which are directly and continuously measured process variables. These specifications, along with TS 15.3.10.D, "Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

The purpose of the limits on the values of $F_Q(Z)$, the height dependent heat flux hot channel factor, is to limit the local peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution. $F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions.

The purpose of the limits on $F^N_{\Delta H}$, the nuclear enthalpy rise hot channel factor, is to ensure that the fuel design criteria are not exceded and the accident analysis assumptions remain valid. The design limits on local and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along a fuel rod to the average fuel rod power. Imposed limits pertain to the maximum $F^N_{\Delta H}$ in the core, that is the fuel rod with the highest integrated power. It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat flux is obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least monthly. However, during power operation, the global power distribution is monitored by TS 15.3.10.E.2, "Axial Flux Difference," and TS 15.3.10.E.3, "Quadrant Power Tilt," which address directly and continuously measured process variables.

Attachment A

As a result of the increased peaking factors allowed by the new 422V+ fuel, a new column was added to TS 15.3.10.E.1.a. The full power $F^N_{\Delta H}$ peaking factor design limit (radial peaking factor) for 422V+ fuel will increase to 1.77 from the 1.70 value for the OFA fuel. The maximum $F_Q(Z)$ peaking factor limit (total peaking factor) for 422V+ fuel will increase to 2.60 from the 2.50 value for the OFA fuel. The OFA fuel design will retain the current $F^N_{\Delta H}$ and $F_Q(Z)$ peaking factors of 1.70 and 2.50, respectively. In addition, the K(Z) envelope for the new 422V+ fuel was modified and a new TS figure 15.3.10-3a was developed and inserted in the Technical Specifications. The K(Z) envelope in TS Figure 15.3.10-3 remains for the OFA fuel.

For OFA and Upgraded OFA Fuel and Figure 15.3.10-3a for 422V+ Fuel

It has been determined that, provided the following conditions are observed, the hot channel factor limits will be met:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 24 steps from the bank demand position, when the bank demand position is between 30 steps and 215 steps. A misalignment of 36 steps is allowed when the bank position is less than or equal to 30 steps, or, when the bank position is greater than or equal to 215 steps, due to the small worth and consequential effects of an individual rod misalignment.

2. Control rod banks are sequenced with overlapping banks as described in Figure 15.3.10-1.

3. Control bank insertion limits are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation of $F^N_{\Delta H}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above four conditions are observed, these hot channel factor limits are met. In Specification 15.3.10.E.1.a, $F_Q$ is arbitrarily limited for $p \leq 0.5$.

The An upper bound envelope of 2.50 (Fq defined in 15.3.10.E) times the normalized peaking factor axial dependence of Figure 15.3.10-3 consistent with the Technical Specifications on power distribution control as given in Section 15.3.10 was used in the large and small break LOCA analyses. The envelope was determined based on allowable power density distributions at full power restricted to axial flux difference (ΔI) values consistent with those in Specification 15.3.10.E.2.

The results of the analyses based on this upper bound envelope indicate a peak clad temperature of less than the 2200°F limit. When an $F_Q$ measurement is taken, both experimental error and manufacturing tolerance must be taken into account. Five percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. In the design limit of $F^N_{\Delta H}$, there is eight percent allowance for uncertainties which means that normal operation of the core is expected to result in a design $F^N_{\Delta H} \leq 1.70/1.08$. The logic behind the larger uncertainty in this case is as follows: For OFA and Upgraded OFA Fuel and 1.77/1.08 for 422V+ Fuel.

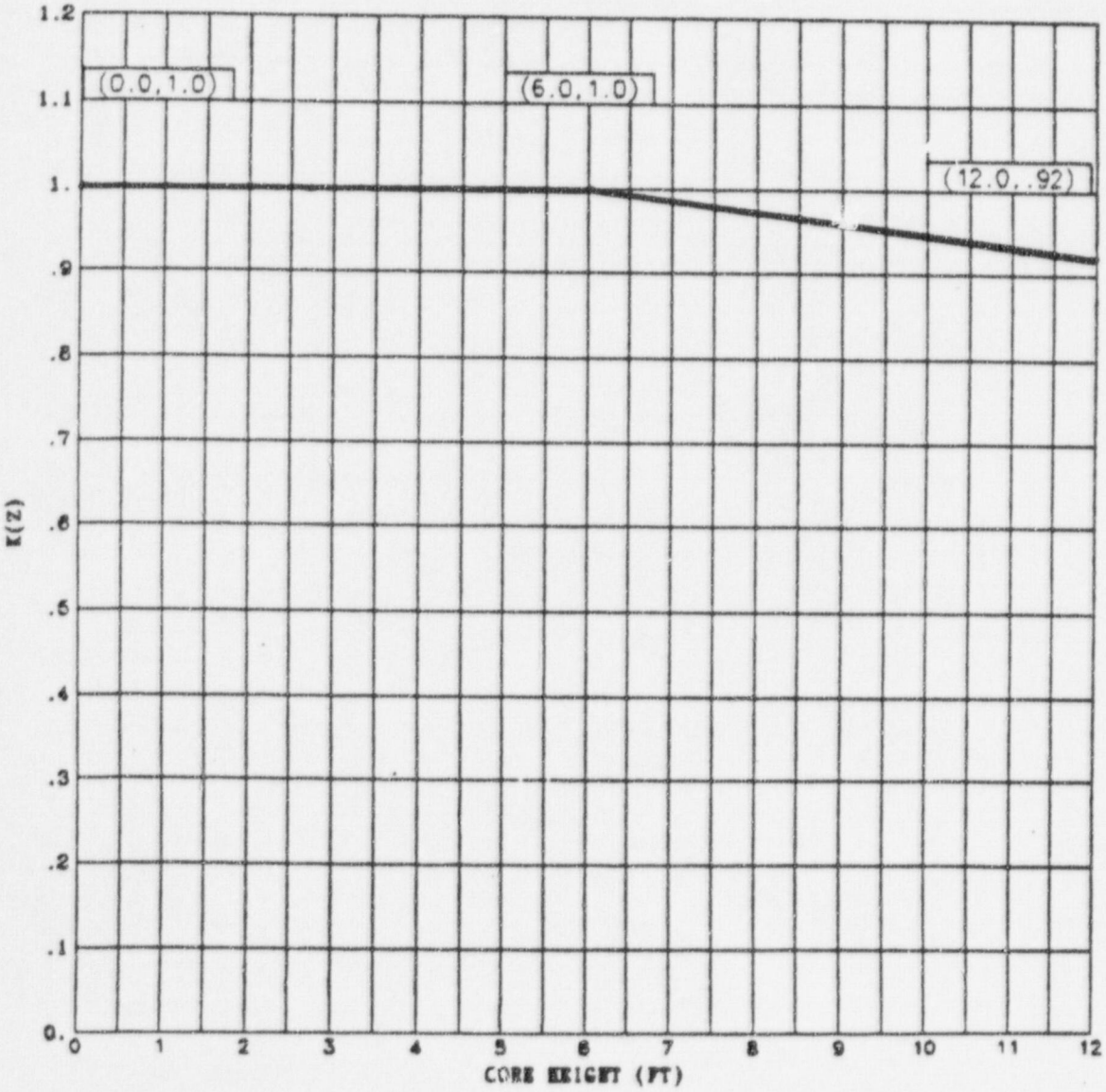
(a) Normal perturbations in the radial power shape (i.e., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting $F_Q$.

FIGURE 15.3.10-3

POINT BEACH UNITS 1 AND 2

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE

FOR OFA AND UPGRADED OFA FUEL

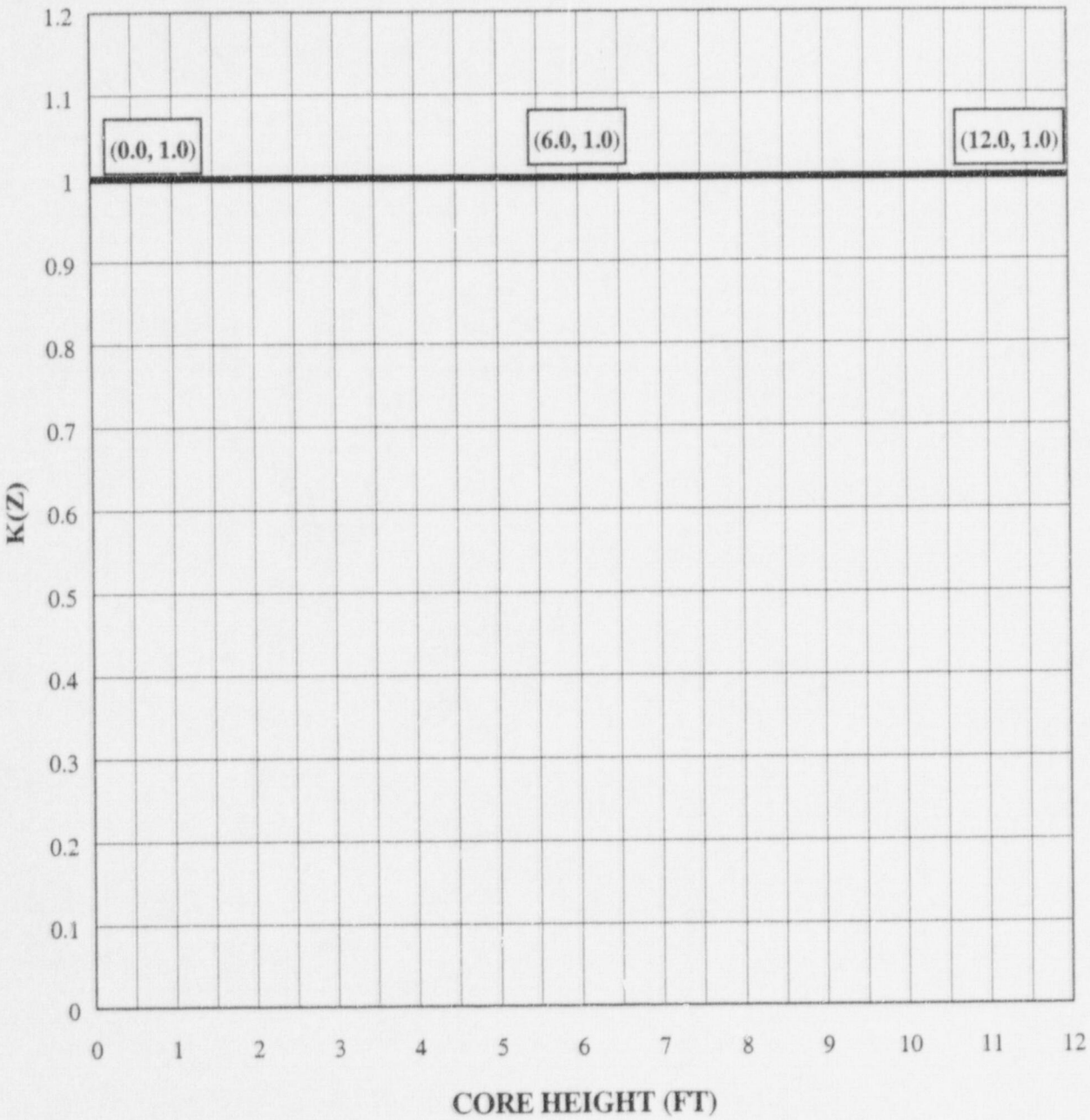



Unit 1 - Amendment No. 120 — May 8, 1989

Unit 2 - Amendment No. 123 — November 1, 1989

FIGURE 15.3.10-3A

POINT BEACH UNITS 1 AND 2
HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE FOR 422V+ FUEL



15.5.3 REACTOR

Applicability

Applies to the reactor core, Reactor Coolant System, and Emergency Core Cooling Systems.

Objective

To define those design features which are essential in providing for safe system operation.

Specifications

A. Reactor Core

1. General

The uranium fuel is in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 121 fuel assemblies. Each fuel assembly nominally contains 179 fuel rods[1]. Where safety limits are not violated, limited substitutions of fuel rods by filler rods consisting of Zircaloy 4 or stainless steel, or by vacancies, may be made to replace damaged fuel rods if justified by cycle specific reload analysis.

2. Core

A reactor core is a core loading pattern containing any combination of 14x14 OFA and 14x14 upgraded OFA fuel assemblies. The core may also contain previously depleted 14x14 standard fuel assemblies. The use of previously depleted 14x14 standard fuel assemblies will be justified by a cycle specific reload analysis.

15.5.4 FUEL STORAGE

Applicability

Applies to the capacity and storage arrays of new and spent fuel.

Objective

To define those aspects of fuel storage relating to prevention of criticality in fuel storage areas.

Specification

1. The new fuel storage and spent fuel pool structures are designed to withstand the anticipated earthquake loadings as Class I structures. The spent fuel pool has a stainless steel liner to ensure against loss of water.

2. The new and spent fuel storage racks are designed so that it is impossible to store assemblies in other than the prescribed storage locations. The fuel is stored vertically in an array with sufficient center-to-center distance between assemblies to assure $K_{eff}<0.95$ with the storage pool filled with unborated water and with the fuel loading in the assemblies limited to 5.0 w/o U-235,with or without axial blanket loadings. Each assembly with a fuel loading greater than 4.6 w/o U-235 must contain Integral Fuel Burnable Absorber (IFBA) rods in accordance with Figure 15.5.4-1 for the spent fuel pool. or have a reference infinite multiplication factor, $K_{\infty}$, less than or equal to 1.49364, which includes a 1% ΔK reactivity bias. An inspection area shall allow rotation of fuel assemblies for visual inspection, but shall not be used for storage.

3. The spent fuel storage pool shall be filled with borated water at a concentration of at least 2100 1800 ppm boron whenever there are spent fuel assemblies in the storage pool.

4. Except for the two storage locations adjacent to the designated slot for the spent fuel storage rack neutron absorbing material surveillance specimen irradiation, spent fuel assembly storage locations immediately adjacent to the spent fuel pool perimeter or divider walls shall not be occupied by fuel assemblies which have been subcritical for less than one year.

Fresh fuel assemblies with the maximum enrichment of up to 5.0 weight percent U235 and a minimum of 32 1.25X IFBA rods can utilize all available new fuel vault storage cells

**Part A of Attachment 5 - "Westinghouse Revised Thermal Design Procedure (RTDP) Instrument Uncertainty Methodology for WE Point Beach Units 1 and 2 (Fuel Upgrade and Uprate to 1656 Mwt NSSS Power)" - WCAP-14787 (Proprietary)**