C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the Donald C. Cook Nuclear Plant, Unit No. 1, at steady state reactor core power levels not to exceed 3304 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 273, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Less than Four Loop Operation

The licensee shall not operate the reactor at power levels above P-7 (as defined in Table 3.3-1 of Specification 3.3.1.1 of Appendix A to this license) with less than four reactor coolant loops in operation until (a) safety analyses for less than four loop operation have been submitted, and (b) approval for less than four loop operation at power levels above P-7 has been granted by the Commission by amendment of this license.

(4) Indiana Michigan Power Company shall implement and maintain, in effect, all provisions of the approved Fire Protection Program as described in the Updated Final Safety Analysis Report for the facility and as approved in the SERs dated December 12, 1977, July 31, 1979, January 30, 1981, February 7, 1983, November 22, 1983, December 23, 1983, March 16, 1984, August 27, 1985, June 30, 1986, January 28, 1987, May 26, 1987, June 16, 1988, June 17, 1988, June 7, 1989, February 1, 1990, February 9, 1990, March 26, 1990, April 26, 1990, March 31, 1993, April 8, 1993, December 14, 1994, January 24, 1995, April 19, 1995, June 8, 1995, and March 11, 1996, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

1.0 DEFINITIONS

DEFINED TERMS -

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3304 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1 1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electric power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature $\Delta T \leq \Delta T_o \; [K_1 - K_2 \left[\dfrac{1+\tau_1 s}{1+\tau_2 s}\right] (T-T') + K_3 (P-P') - f_1 (\Delta I)]$

Where:	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
	T	=	Average temperature, °F
	T'	=	Indicated T_{avg} at RATED THERMAL POWER (≤ 574.0 °F)
	P	=	Pressurizer pressure, psig
	P'	=	Indicated RCS nominal operating pressure (2235 psig or 2085 psig)
	$\dfrac{1+\tau_1 s}{1+\tau_2 s}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	=	Time constants utilized in the lead-lag controller for T_{avg} τ_1= 22 secs. τ_2 = 4 secs.
	S	=	Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \left[K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 (T - T'') - f_2(\Delta I)\right]$

Where:	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
	T	=	Average temperature, °F
	T''	=	Indicated T_{avg} at RATED THERMAL POWER ($\leq$ 562.1 °F)
	K_4	=	1.083
	K_5	=	0.0177/°F for increasing average temperature and 0 for decreasing average temperature
	K_6	=	0.0015 for $T > T''$; $K_6 = 0$ for $T \leq T''$
	$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
	τ_3	=	Time constants utilized in the rate lag controller for T_{avg} $\tau_3 =$ 10 secs.
	S	=	Laplace transform operator
	$f_2(\Delta I)$	=	0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 3.4 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.5 percent ΔT span.

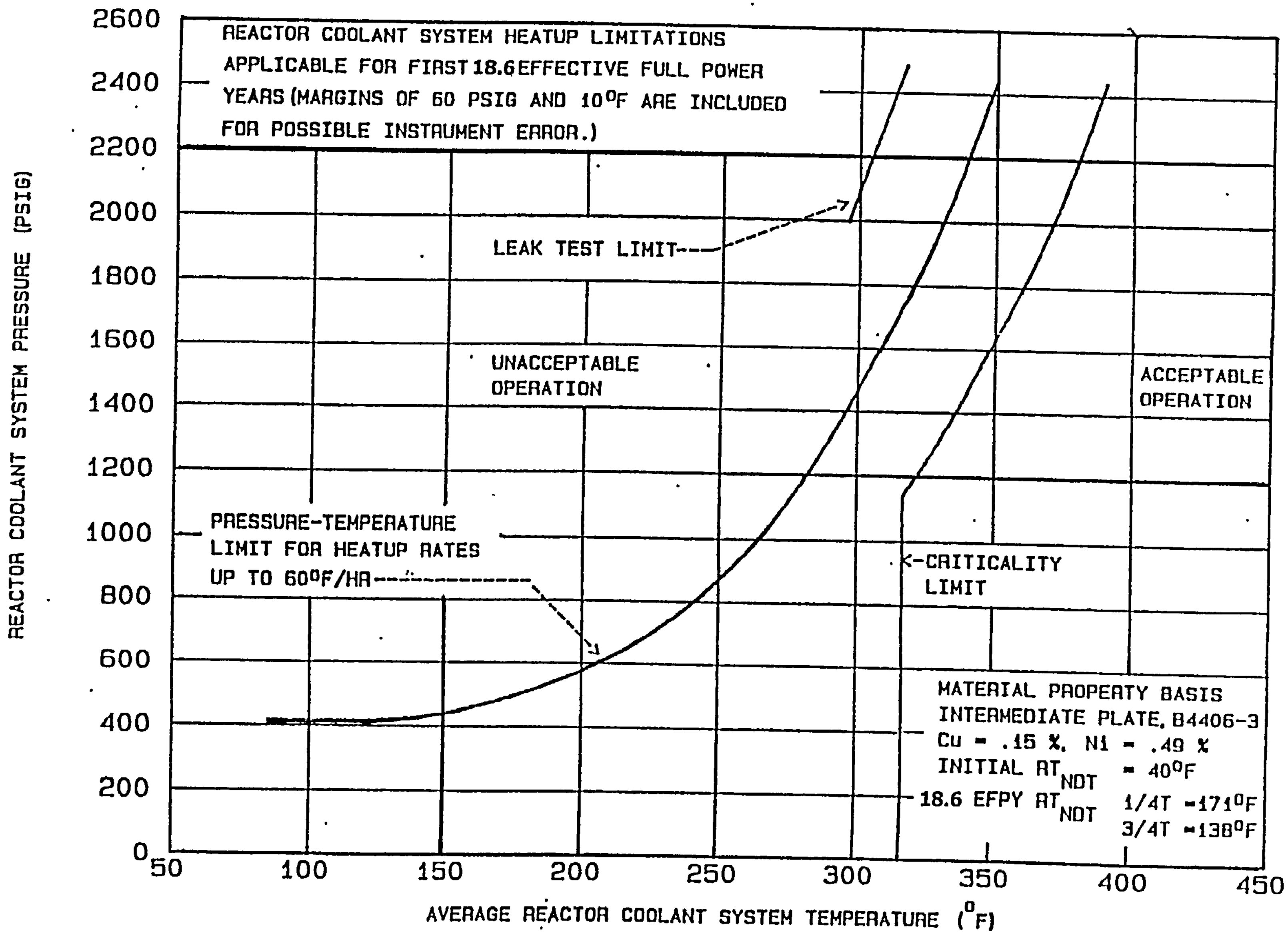

FIGURE 3.4-2

REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS VERSUS 60° F/HR RATE
CRITICALITY LIMIT AND HYDROSTATIC TEST LIMIT

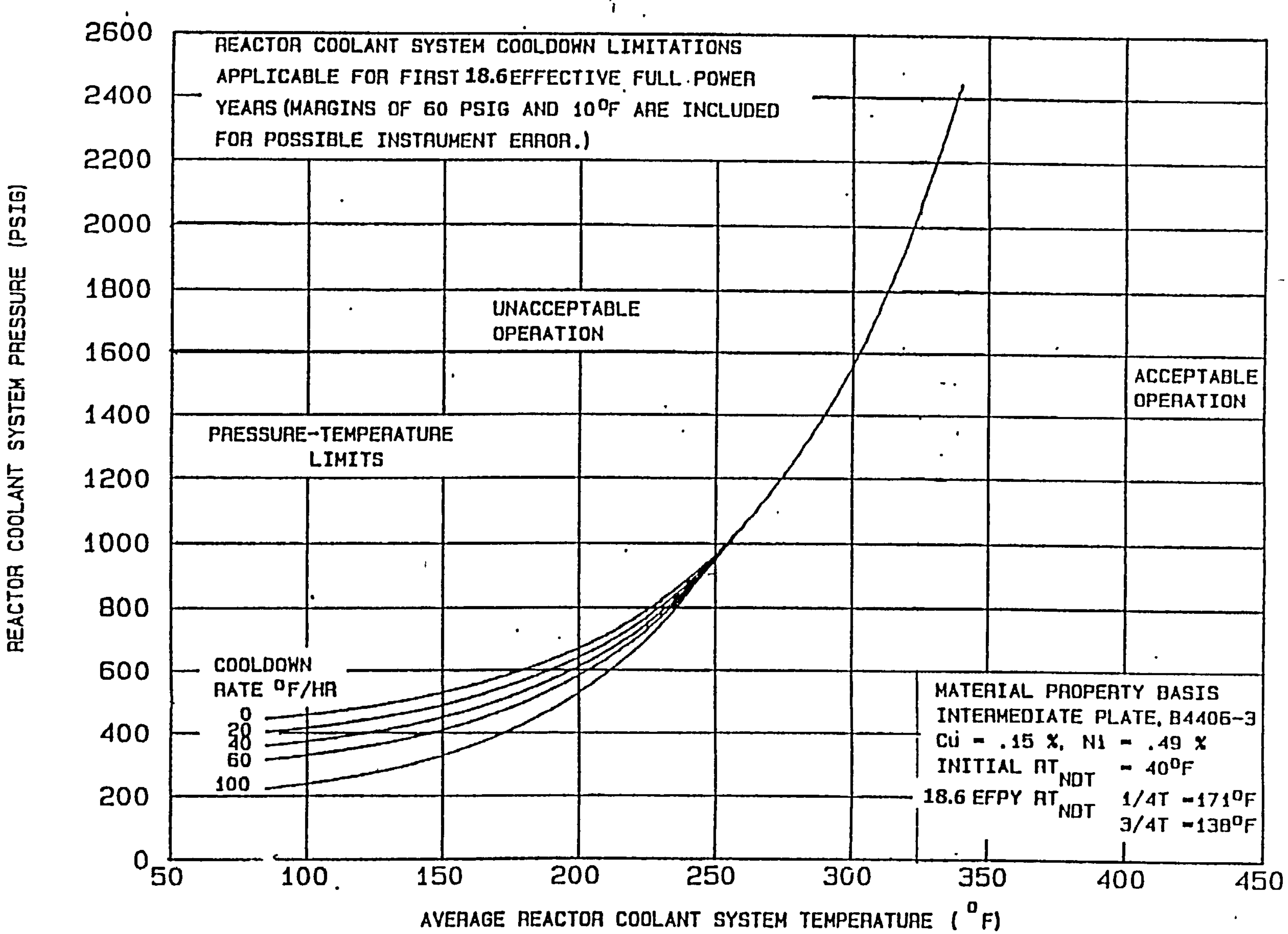

FIGURE 3.4-3

REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS VERSUS COOLDOWN RATES

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	63.8
2	45.5
3	27.4

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.4 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

An ID or OD one-quarter thickness surface flaw is postulated at the location in the vessel which is found to be the limiting case. There are several factors which influence the postulated location. The thermal induced bending stress during heatup is compressive on the inner surface while tensile on the outer surface of the vessel wall. During cooldown the bending stress profile is reversed. In addition, the material toughness is dependent upon irradiation and temperature and therefore the fluence profile through the reactor vessel wall, the rate of heatup and also the rate of cooldown influence the postulated flaw location.

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 18.6 EFPY.

Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, and the copper and nickel content of the material must be predicted. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include the adjusted RT_{NDT} at the end of 18.6 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The 18.6 EFPY heatup and cooldown curves were developed based on the following:

1. The intermediate shell plate, B4406-3, being the limiting material with a copper and nickel content of .15% and .49%, respectively.

2. The fluence values documented in I&M letter AEP:NRC:2900-02.

3. Figure 1, NRC Regulatory Guide 1.99, Revision 2

The shift in RT_{NDT} of the reactor vessel material has been established by removing and evaluating the material surveillance capsules installed near the inside wall of the reactor vessel in accordance with the removal schedule in Table 4.4-5. Per this schedule, Capsule U is the last capsule to be removed until Capsule S is to be removed after 32 EFPY (EOL). Capsule V, W, and Z will remain in the reactor vessel, and will be removed to address industry reactor embrittlement concerns, if required

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, or of one PORV and the RHR safety valve, ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 152°F. Either PORV or RHR safety valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a charging pump and its injection into a water solid RCS. Therefore, any one of the three blocked open PORVs constituted an acceptable RCS vent to preclude APPLICABILITY of Specification 3.4.9.3.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The safety valve is OPERABLE with a lift setting of ±3% about the nominal value. However, the safety valve shall be reset to the nominal value ±1% whenever found outside the ±1% tolerance. The total relieving capacity for all valves on all of the steam lines is 17,153,800 lbs/hr which is approximately 118 percent of the total secondary steam flow of 14,540,000 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per operable steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

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

$$Hi\Phi = (100/Q)\frac{(4\, w_s h_{fg})}{K}$$

where:

$Hi\Phi$ = Safety Analysis power range high neutron flux setpoint in percent

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat) in Mwt

K = Conversion factor, 947.82 $\frac{\text{(Btu/Sec)}}{\text{Mwt}}$

w_s = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec. For example, if the maximum number of inoperable MSSVs on any one steam generator is one, then w_s should be a summation of the capacity of the operable MSSVs at the highest operable MSSV operating pressure, excluding the highest capacity MSSV. If the maximum number of inoperable MSSVs per steam generator is three, then w_s should be a summation of the capacity of the operable MSSVs at the highest operable MSSV operating pressure, excluding the three highest capacity MSSVs.

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate in Btu/lbm

4 = Number of loops in plant

The values calculated from this algorithm are then adjusted lower for use in Technical Specification 3.7.1.1 to account for instrument and channel uncertainties by 9%. This reduces the maximum plant operating power level so that it is lower than the reactor protection system setpoint by an appropriate operating margin.