Attachment 1

Unit 2	Revision
Page 2-2	Replace
Page 2-5	Replace
Page 2-8	Replace
Page 2-9	Replace
Page 3/4 2-15	Replace

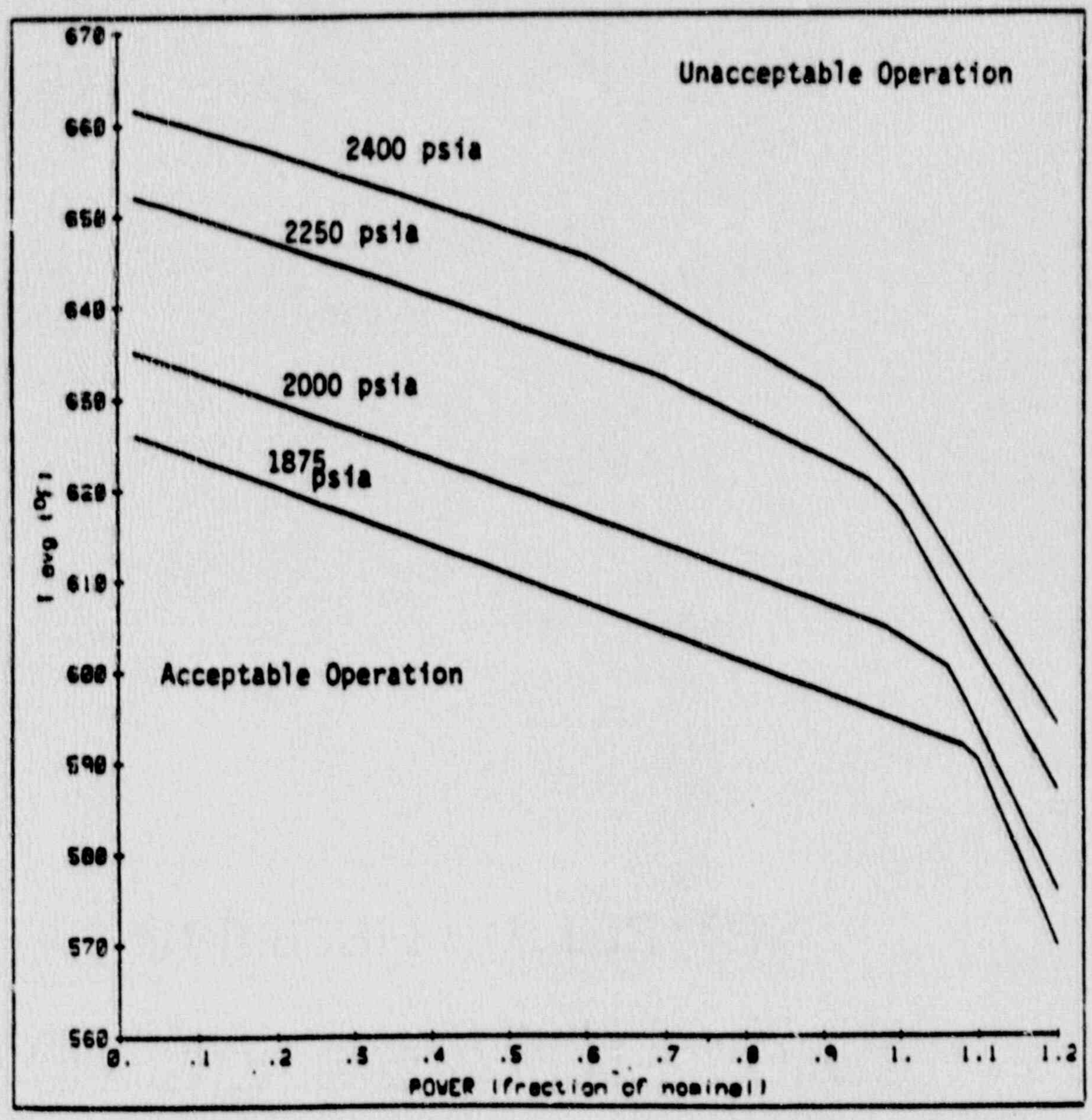

Figure 2.1-1 Reactor Core Safety Limit

Three Loops in Operation

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive R[illegible]	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 second	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 second
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 second	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 second
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	$\geq$ 1865 psig	$\geq$ 1855 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% of design flow per loop*

*Design flow is 87,200 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION

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

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

$T' \leq$ 577.2°F (Maximum Reference T_{avg} at RATED THERMAL POWER)

P = Pressurizer pressure, psig

P' = 2235 psig (Nominal RCS operating pressure)

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = 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 = 30 secs, τ_2 = 4 secs.

S = Laplace transform operator, sec^{-1}.

Operation with 3 loops	Operation with 2 loops
K_1 = 1.18	K_1 = (values blank pending
K_2 = 0.0154	K_2 = NRC approval of
K_3 = 0.000635	K_3 = 2 loop operation)

and f_1 (ΔI) is a 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 such that:

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION continued

(i) for $q_t - q_b$ between -35 percent and +9 percent, $f_1 (\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -35 percent, the ΔT trip setpoint shall be automatically reduced by 1.37 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds +9 percent, the ΔT trip setpoint shall be automatically reduced by 1.75 percent of its value at RATED THERMAL POWER.

Note 2: Overpower $\Delta T \leq \Delta T_o [K_4 - K_5 \frac{\tau_3 S}{1+\tau_3 S} T - K_6 (T-T") - f_2(\Delta I)]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

$T"$ = Reference T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 577.2°F)

K_4 = 1.08

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature

K_6 = 0.00109/°F 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

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 3 Loops in Operation	2 Loops in Operation
Reactor Coolant System T_{avg}	$\leq$ 581.2°F	**
Pressurizer Pressure	$\geq$ 2220 psia*	**
Reactor Coolant System Total Flow Rate	$\geq$ 261,600 gpm	**

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

** Values blank pending NRC approval of 2 loop operation.

Attachment 2

Attachment 3

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Pursuant to the requirements in 10CFR50.92, each application for amendment to an operating license must be reviewed to determine if the modification involves a significant hazard. The amendment as defined in WCAP-12659, describing the "Alabama Power J. M. Farley Unit 2 Increased Steam Generator Tube Plugging and Reduced Thermal Design Flow Licensing Report" has been reviewed and deemed not to involve a significant hazard based on the following evaluation.

The proposed amendment involves an increase in equivalent plugging limits from the current licensed value of 10% uniform plugging to a new licensed value of 15% average with a 20% peak in any one steam generator. Also included is a decrease of approximately 1.5% in Reactor Coolant System (RCS) total flowrate from the current licensed value of 265,500 gpm to a new licensed value of 261,600 gpm. As discussed in WCAP-12659, a comprehensive evaluation of the effects of the increased tube plugging and reduced RCS flowrate has been completed, and no adverse safety implications have been identified.

Neither the increased tube plugging nor the reduced RCS flowrate involve a significant increase in the probability or consequences of any accident previously evaluated. The LOCA and non-LOCA accidents were reviewed in WCAP-12659 verifying that the effects of increased tube plugging and reduced RCS flowrate do not invalidate the current analyses of record and that all design basis conclusions are still met. Several non-LOCA accidents [main feedwater pipe rupture, uncontrolled bank withdrawal from subcritical, partial loss of flow, single reactor coolant pump locked rotor, steam generator tube rupture (doses)] were reanalyzed using the revised conditions associated with increased tube plugging and reduced RCS flowrate, and acceptable results were obtained. For those non-LOCA accidents for which evaluations were performed, acceptable results were obtained by use of existing sensitivity studies, existing margins, or allocation of generic DNB margin. In all cases, current licensing criteria of record is met. In addition, effects of asymmetrical flow distributions have been evaluated and found acceptable. The most limiting Large Break LOCA analysis (C_d=0.4) was reanalyzed for the new configuration, and the analysis demonstrated a calculated PCT less than the Appendix K limit of 2200°F. Evaluations of Small Break LOCA, LOCA hydraulic forcing functions, post-LOCA long-term core cooling, and hot leg switchover to prevent boron precipitation were performed, and all current conclusions for the J. M. Farley Unit 2 remain valid.

Evaluations of MSLB and LOCA mass and energy releases concluded the present mass and energy releases are applicable and the containment responses remain valid and all licensing conclusions remain valid.

Neither the increased tube plugging nor the reduction in RCS total flowrate creates the possibility of a new or different kind of accident from any accident previously evaluated. These effects were reviewed in detail in WCAP-12659 for any adverse effects on RCS components. Major Nuclear Steam Supply System components (i.e., The reactor vessel, internals, loop piping, reactor coolant pump, pressurizer, and the CRDMS) were reviewed, and no adverse safety effects were found. No new single failures were found. No new accident initiators were found. Since increased tube plugging can physically alter the steam generator, the possible effects on the steam generator were evaluated in detail. The results of thermal and hydraulic evaluation, U-bend tube vibration assessment, and the structural evaluation have concluded that the current components of the J. M. Farley Unit 2 steam generators satisfy the requirements of the ASME B&PV, Section III for the increased tube plugging and reduced RCS flowrate. No significant reduction in a margin of safety is involved with the increased tube plugging and reduced RCS flowrate since all current acceptance criteria continue to be met.

The evaluation of the effect of these variables on non-LOCA and LOCA transients has verified that plant operation will be maintained within the bounds of safe, analyzed conditions as defined in the FSAR with the revised technical specifications, and the conclusions presented in the FSAR remain valid. Safe operation with revised core limits, a revised K1 of 1.18 and an approximate 1.5% reduction in thermal design flow has been demonstrated.The impact on K4 was small and no change to K4 was required since adequate margin exists in the Technical Specification value. In addition, component evaluations have been performed to demonstrate compliance to current acceptance criteria. As such, no reduction in a margin of safety as defined in the basis for any technical specification is required for operation of J. M. Farley Unit 2 with increased steam generator tube plugging and reduced RCS flowrate