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HOLTEC
INTERNATIONAL

DYNARACK SIMULATIONS TO DEMONSTRATE THE ABILITY TO PREDICT CLASSICAL NONLINEAR PHENOMENA

by

A.I. Soler

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SUMMARY OF REVISIONS

Holtec Report HI-91614

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EXECUTIVE SUMMARY

This report contains results of additional verification studies on Holtec computer code DYNARACK performed to demonstrate DYNARACK's capability to simulate certain unusual and arcane characteristics exhibited by some nonlinear systems. This report is a sequel to the original DYNARACK Q.A. validation report, and it was undertaken to establish DYNARACK's ability to simulate the so-called "jump" and subharmonic response phenomena associated with certain nonlinear systems. Towards this end, sample dynamic problems were selected from the literature and were analyzed on DYNARACK. These problems demonstrate DYNARACK's ability to capture both standard and special characterizations of nonlinear systems.

1.0 INTRODUCTION

This report provides verification that the Holtec dynamic simulation code DYNARACK* is able to predict certain classical nonlinear phenomena that have been found to exist in certain dynamic systems. In particular, we address here the simulation of Coulomb friction by piecewise linear springs, the development of subharmonic resonance, the establishment of limit cycles, and the prediction of nonlinear "jumps" in the solution depending on the direction of loading.

Four problems are addressed in this report which show that the computer code is capable of predicting the classical phenomena.

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* Holtec Proprietary Reports: User's Manual (HI-89343, Revision 0); Theory, (HI-87162, Revision 1 and HI-90439, Revision 0), and Verification (HI-87161, Revision 2.)

2.0 NONLINEAR DYNAMIC ANALYSIS PROBLEMS

2.1 Subharmonic Resonance

Consider the generic single degree-of-freedom system shown in Figure 2.1. A mass m is subjected to a driving excitation F(t) or a base excitation y(t). The mass is attached to the surrounding environment by a friction interface, by a gap element $F_g(t)$ with spring rate K_4, and by an elastic spring-damper system that can exhibit at most nonlinear cubic behavior (in Figure 2.1, δ_1 is the extension of the elastic spring). Figure 2.2 shows the behavior of the different elements qualitatively. All of these "spring" elements are coded in DYNARACK; the user need only input the information regarding the degrees-of-freedom that cause the spring extension, and the information concerning spring rate magnitude, etc.

In the initial problem, we assume F(t) = 0, and the coefficient of friction $\mu = 0$ (note that $|P_{LIM}| = \mu F_{LIM}$ where F_{LIM} is an input value or the load from an adjacent compression only stop element that represents the contact). We also assume in this problem that x_0 is large so that the gap element never acts. The nonlinear spring is assumed as ($K_3 = 0$)

$$F_s = -[K_1\ \delta_1 + K_2\ \delta_1^2]$$

and the input excitation is $\ddot{y}(t) = 10g \sin(2\pi ft)$.

We assume m = .1036 lb.sec.^{2}/in., C_1 = 0., K_1 = 90 lb./in., K_2 = 10 lb./in.^{2}, f = 9.4 HZ, and g = 386.4 in./sec. For a low amplitude linear excitation, the linear natural frequency is

$$f_o = \frac{1}{2\pi} \left(\frac{K_1}{m} \right)^{\frac{1}{2}} = 4.69 \text{ HZ}$$

If we integrate $\ddot{y}(t)$, and require that there be no rigid body base motion, then

$$\dot{y}(0) = \frac{10g}{2\pi f} = -65.4228 \text{ in./sec.} \qquad y(0) = 0$$

To ensure that the spring is initially unstretched, we assume the same initial conditions on the mass m.

Figure 2.3 shows the acceleration of the mass versus time. The subharmonic resonance is clearly visible in that there is a strong response at 4.7 HZ (half the frequency of the imposed driving excitation). Appendix A contains numerical results for the same problem done in Reference 6.6.

2.2 Sliding Friction and Dead Bands

Consider Figure 2.1 for the case $x_0 \rightarrow \infty$, $F_S = 0$, $C_1 = 0$, $F(t) = \beta \sin rt$. That is, we consider a mass resting on a frictional surface which generates a frictional resisting force $\pm R$ and is driven by an external sinusoidal force. Tou and Schutheiss* have given solutions for this situation. The interesting features of the motion are that if $R/\beta < .536$, the motion is roughly sinusoidal, but has discontinuities in acceleration. If $R/\beta > .536$, then the motion is sporadic, there being so-called dead bands

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* "Static and Sliding Friction in Feedback Systems, J. Tou and P.M. Schultheiss, Jour. Appl. Physics, Vol. 24, 9, September 1953, pp 1210-1217.

within which no motion occurs. When $R/\beta > 1$, no motion is possible except for an initial transient. Appendix B contains a copy of the reference. Here, we use DYNARACK to model the phenomena. The governing equation is

$$m \frac{d^2 x}{dt^2} = \beta \sin wt \pm R$$

We simulate the event for $m = \beta = 1$, and $R/\beta = \lambda = .3, .7$, and 1.01. The friction spring constant (Figure 2.2) is set at $K_f = 1 \times 10^7$ lb./in. to simulate an "infinite" slope. Figures 2.4 - 2.6 show the results for the three values of λ. It is clearly evident that DYNARACK is capable of reproducing the expected phenomena. In Figure 2.6, the small non-zero velocity components subsequent to the initial transient are due to the presence of the finite K_f.

2.3 Jump Phenomena

Consider the differential equation

$$m \ddot{x} + b \dot{x} + cx + dx^3 = E \sin \nu t$$

We let $x_1 = (d/c)^{1/2} x$; $t_1 = (c/m)^{1/2} t$

Then the differential equation becomes

$$\frac{d^2 x_1}{dt_1^2} + \delta \frac{dx_1}{dt_1} + x_1 + x_1^3 = E_1 \sin \nu_1 t$$

where x_1, t_1, E_1, ν_1, δ are dimensionless and

$$\delta = b/c\ (c/m)^{1/2} \quad E_1 = E/C\ (d/c)^{1/2} \quad \nu_1 = \nu(m/c)^{1/2}$$

[illegible]

An approximate first order asymptotic solution is obtained analytically by Bogliubov and Mitropolsky.*

For an assumed oscillatory solution

$$x_1 = a \cos (\nu_1 t + \theta) \quad a = a(t); \; \theta = \theta(t)$$

the approximate solution for the ν-a resonance curve is

$$w_c^2 = (1 + \frac{3}{8} a^2)^2$$

$$\nu_1 = \{w_c^2(a) \pm \left[\frac{E_1^2}{a^2} - \delta^2\right]^{1/2} \}^{1/2}$$

For the parameters $\delta = .2$, $E_1 = 1$, the resonance curve can be constructed using using the above approximate analytical solution. A typical result is shown in Appendix C, and tabular results, sufficient to plot the resonance curve, are also given in that appendix.

Note that the solution to the linear non-dimensional equation (neglect the x_1^3 term) predicts the peak amplitude $|x_1|_{max} = 5.0$ at a frequency of .99 rad/sec. It has been shown that a system with a hardening spring has a resonance curve whose central spine is tilted to the right as shown in Figure C-1 in Appendix C. The resonance curve is obtained for a given amplitude of excitation. In practice, the resonance curve shows areas of instability. If the excitation frequency rises, the response follows the resonance curve to a certain point, and then drops abruptly to a smaller amplitude. Similarly, if the excitation frequency is decreased through the resonance region, the response will pass along the

* Asymptotic Methods in the Theory of Non Linear Oscillations", by N.N. Bogolinbov and Y.A. Mitropolsky (Translation by Hindustan Publishing Corp., 1961), pp 244-245.

lower curve, to the inflection point, and then abruptly increase to the higher branch. It is found in actual shaker tests that the onset of the instability is sensitive to the sweep rate of the shaker. We attempt here to demonstrate the ability of DYNARACK to predict a jump phenomena. We simulate the resonance curve by doing a time history analysis with $E_1 = 1$, $\delta = .2$, and υ_1 slowly increased from a value less than 1.0 rad/sec. The sweep rate is set low enough to be able to ascertain the maximum response amplitude. Figures 2.7 and 2.8 show the result of the DYNARACK simulation. Figure 2.8 also contains the two branches of the resonance curve plotted from Table C-1 in Appendix C. Both figures show that the simulation code is able to model the jump instability; however, the results show that the onset of instability is very sensitive to the time step size. Figure 2.7 represents a run for 5000 seconds with step size .004 sec. while Figure 2.8 shows the curve obtained with a step size of .0001 sec. We see that the amplitude at the onset of instability is relatively insensitive to the step size but the frequency at which the instability occurs is sensitive to the step size. It is apparent that to exactly follow the resonance curve (itself an approximate solution), an extremely fine step size is called for. We did not attempt any modeling of the resonance with a decreasing forcing frequency.

2.4 LIMIT CYCLES

We consider the problem studied in Shaw and Holmes.* to investigate the ability of DYNARACK to predict the existence of stable limit cycles. In the Shaw and Holmes paper, the model

* S.W. Shaw and P.J. Holmes, "A Periodically Forced Piecewise Linear Oscillator", Journ. of Sound and Vibration, Vol. 90, 1983, pp 129-155.

described in Figure 2.1 is investigated under the assumptions that friction is absent, the spring F_S is linear, and the excitation is by a harmonic force. The non-dimensionalized equation can be written as

$$\ddot{x} + 2\alpha\dot{x} + H(x) = \beta \cos w t$$

where
$$H(x) = \begin{cases} x & x < x_o \\ \bar{w}^2 x + (1 - \bar{w}^2)x_o & x \geq x_o \end{cases}$$

and $\bar{w}^2 = (K_1 + K_4)/K_1$

Figure 2.9, reproduced from Shaw and Holmes shows different cases of the above equation. The case $x_o = 0$ is considered so the problem is essentially that of a piecewise linear spring having a different spring constant in tension and in compression.

The results from DYNARACK, for the same problems of Figure 2.9, are shown in Figures 2.10 - 2.14. The computer simulation started from a specified initial condition and covered a sufficient number of cycles so that a limit cycle could be established. Figures 2.10 and 2.11 show the match with two problems shown in Figure 2.9. The simulation predicts the one or two stable limit cycles. Figures 2.12 - 2.14 show results of the similar analysis for the third case considered. Figures 2.12 and 2.13 show the effect of different starting conditions. It is clear that only the stable period one orbit is being tracked by the numerical solution. Figure 2.14 is a plot of only the last few hundred time steps and confirms the tendency toward the stable period one orbit. The period three orbit appears unstable which seems to contradict the conclusion in Shaw and Holmes. However, if one reads the text in Figure 2.9, Shaw and Holmes appear to conclude that an <u>unstable</u> period three also exists at the same parameter values. We can only conclude that it is extremely fortuitous if the numerical solution can simulate a stable orbit when an unstable orbit with the same period exists concurrently.

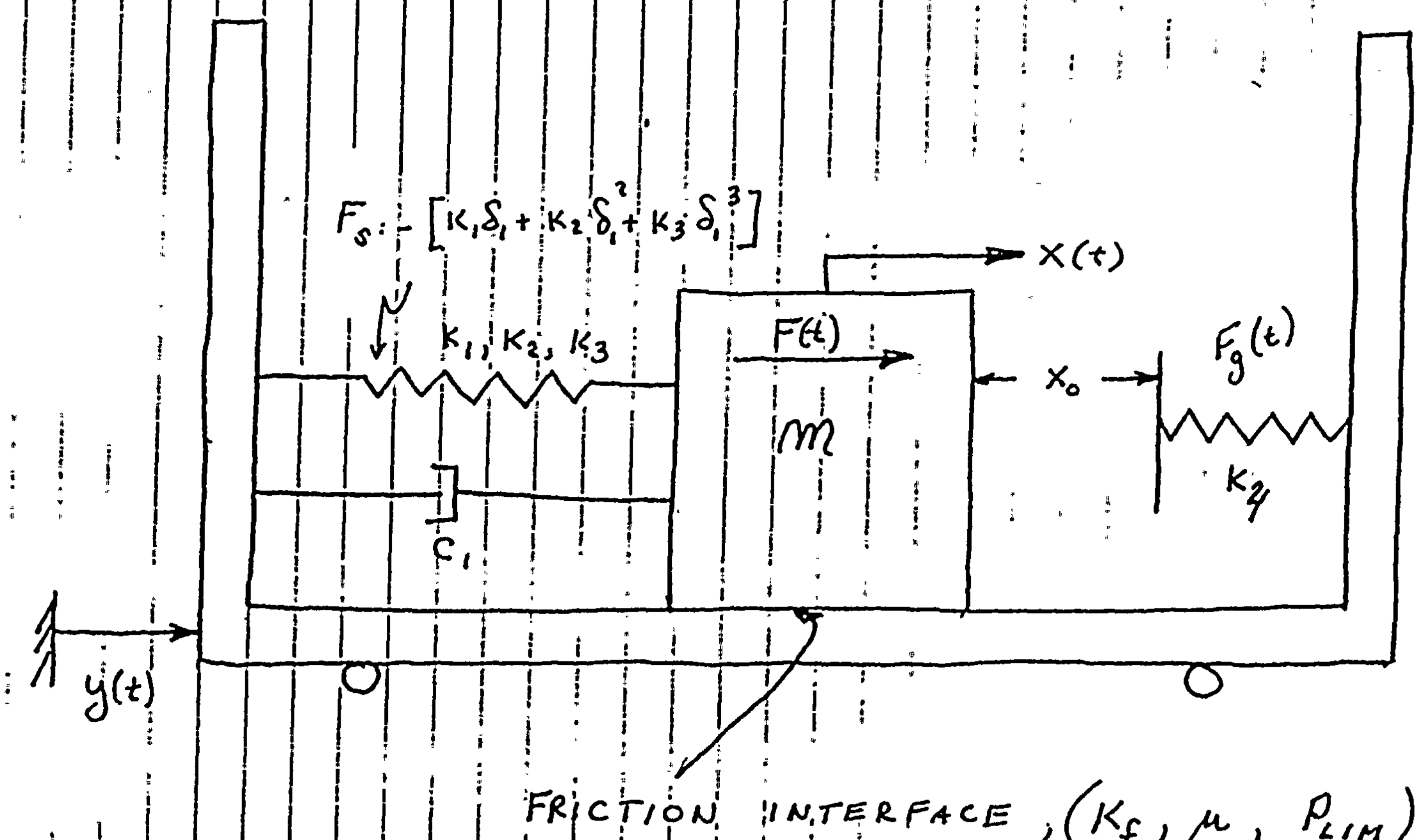
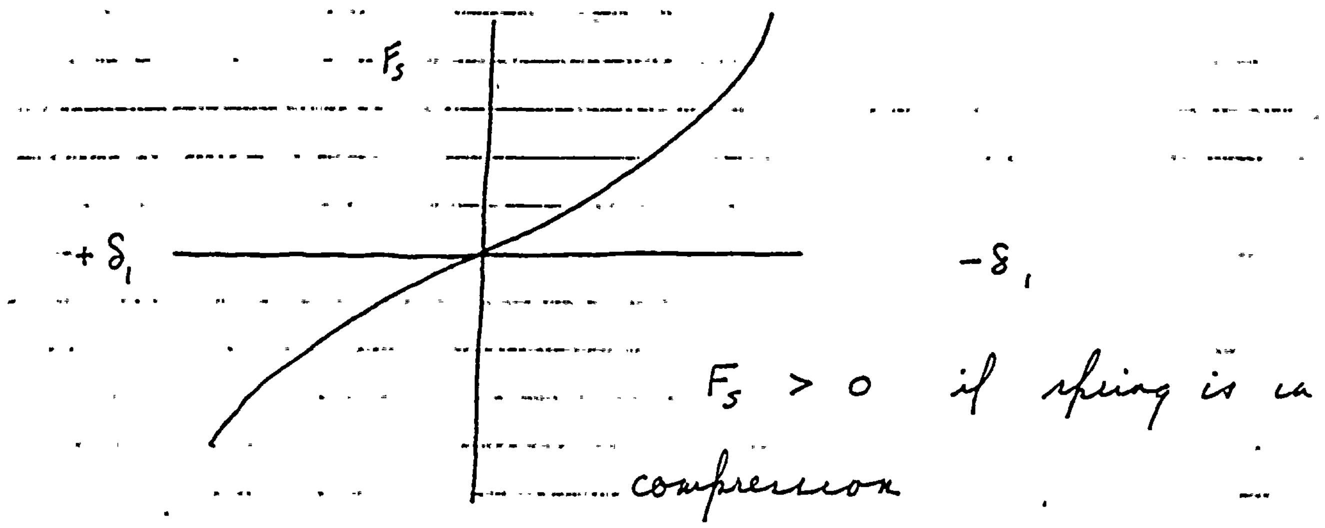


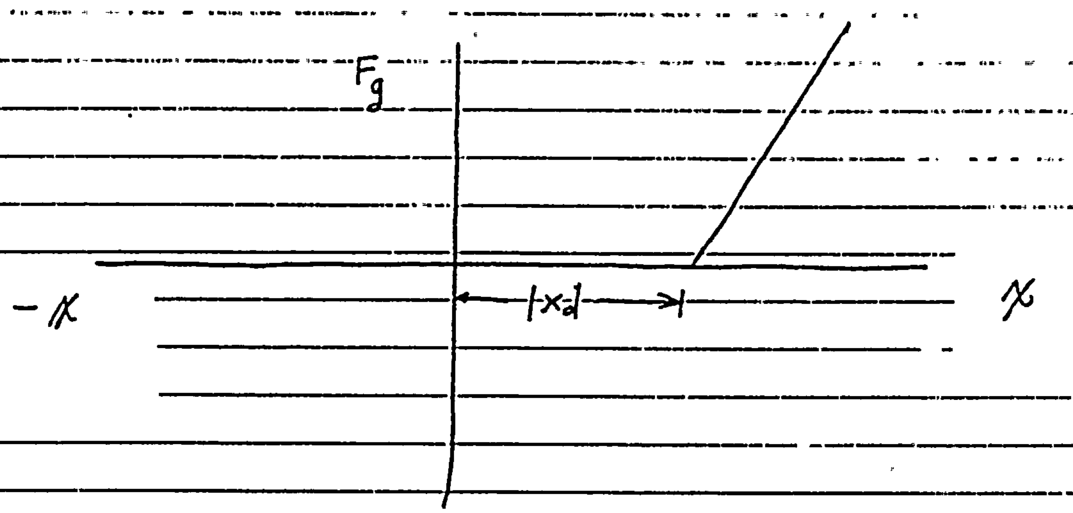
FIGURE 2.1 1-DOF MODEL

FIGURE 2.2 SPRING CHARACTERISTICS

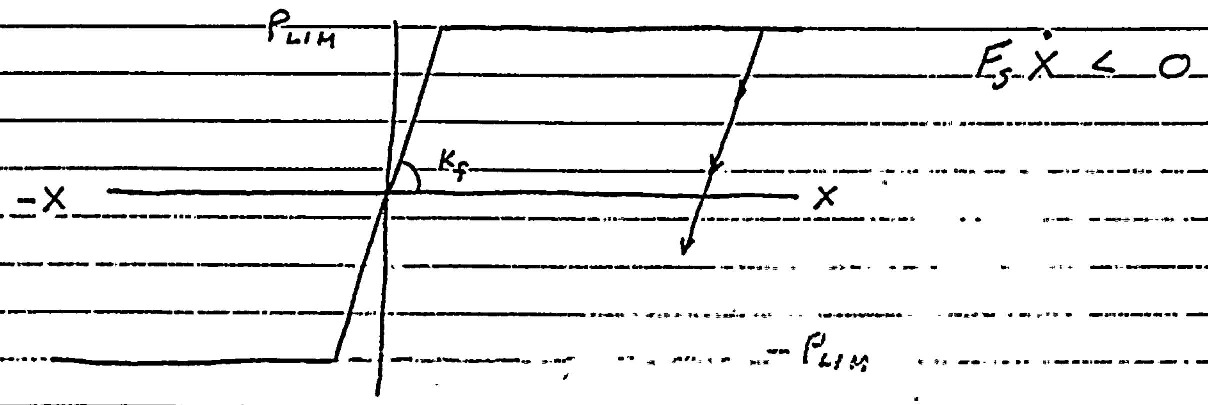
NON LINEAR ELASTIC SPRING



COMPRESSION ONLY GAP ELEMENT



FRICTION INTERFACE ELEMENT



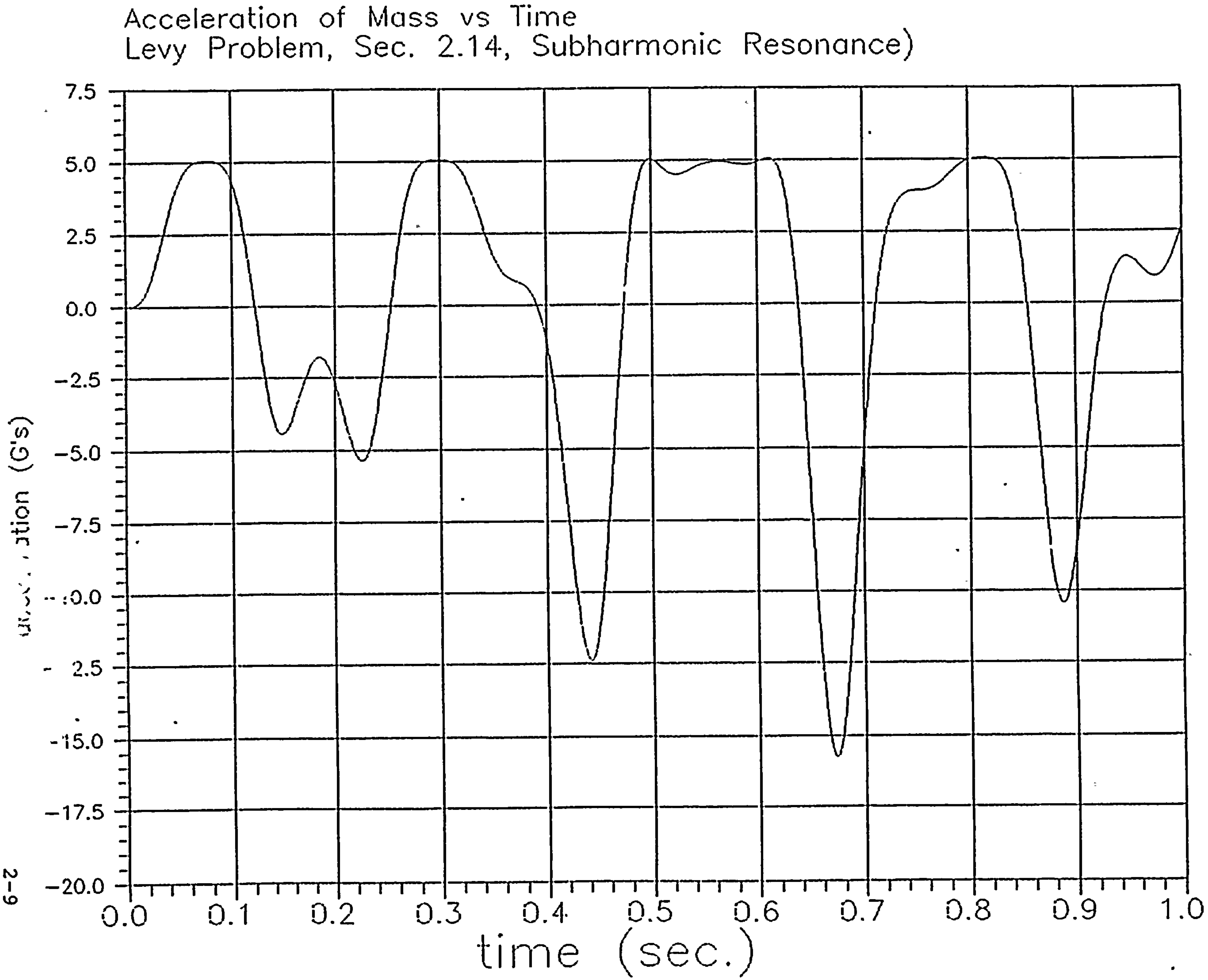
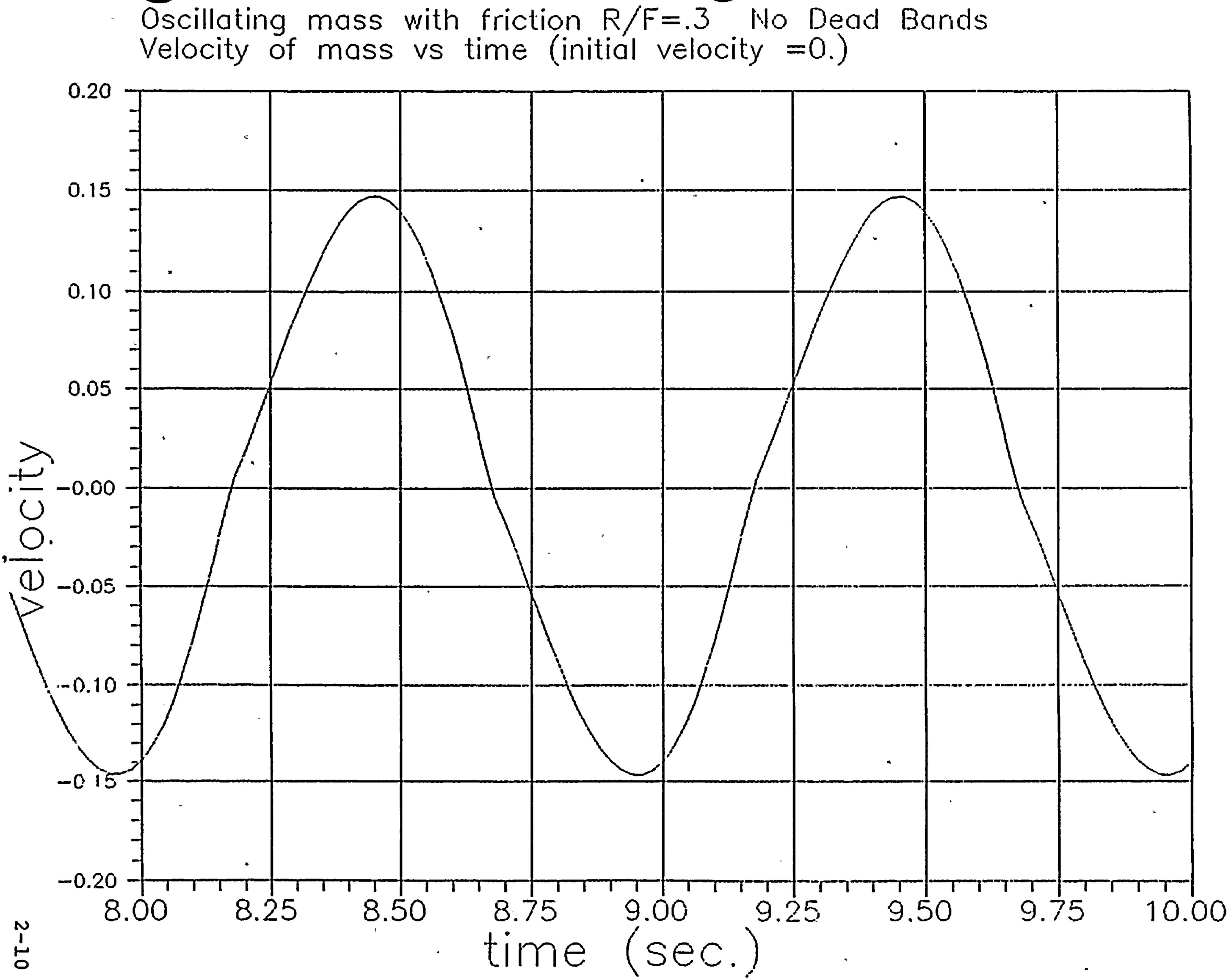
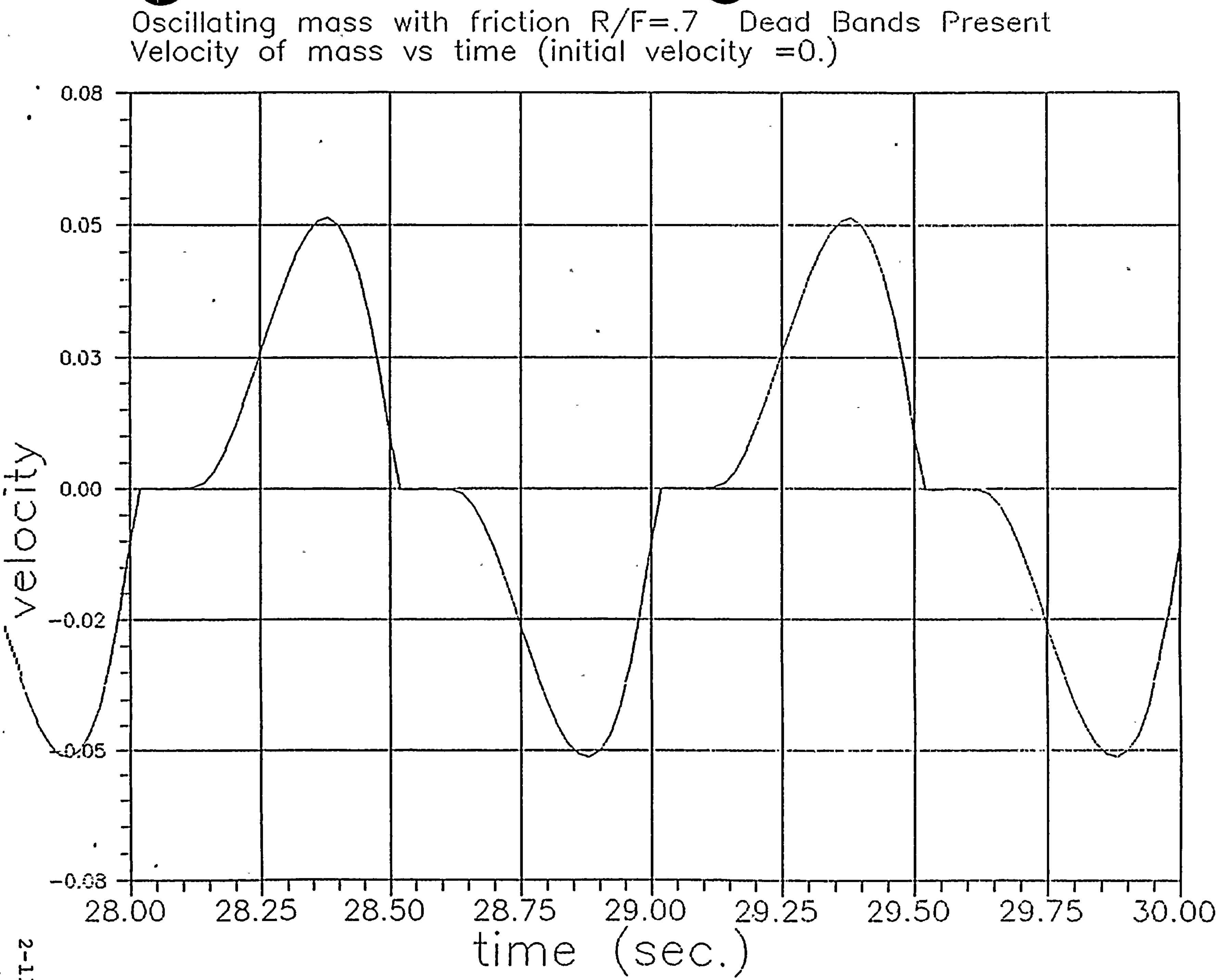

Acceleration of Mass vs Time
Levy Problem, Sec. 2.14, Subharmonic Resonance)
7.5
5.0
2.5
0.0
−2.5
−5.0
−7.5
−17.5
−20.0
(G's)
0.0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
time (sec.)

FIGURE 2.3



Oscillating mass with friction R/F=.3 No Dead Bands
Velocity of mass vs time (initial velocity =0.)
0.20
0.15
0.10
0.05
-0.00
-0.05
-0.10
-0.15
-0.20
velocity
8.00
8.25
8.50
8.75
9.00
9.25
9.50
9.75
10.00
time (sec.)

FIGURE 2.4


Oscillating mass with friction R/F=.7 Dead Bands Present
Velocity of mass vs time (initial velocity =0.)
0.08
0.05
0.03
0.00
-0.02
-0.05
-0.08
velocity
28.00
28.25
28.50
28.75
29.00
29.25
29.50
29.75
30.00
time (sec.)
FIGURE 2.5

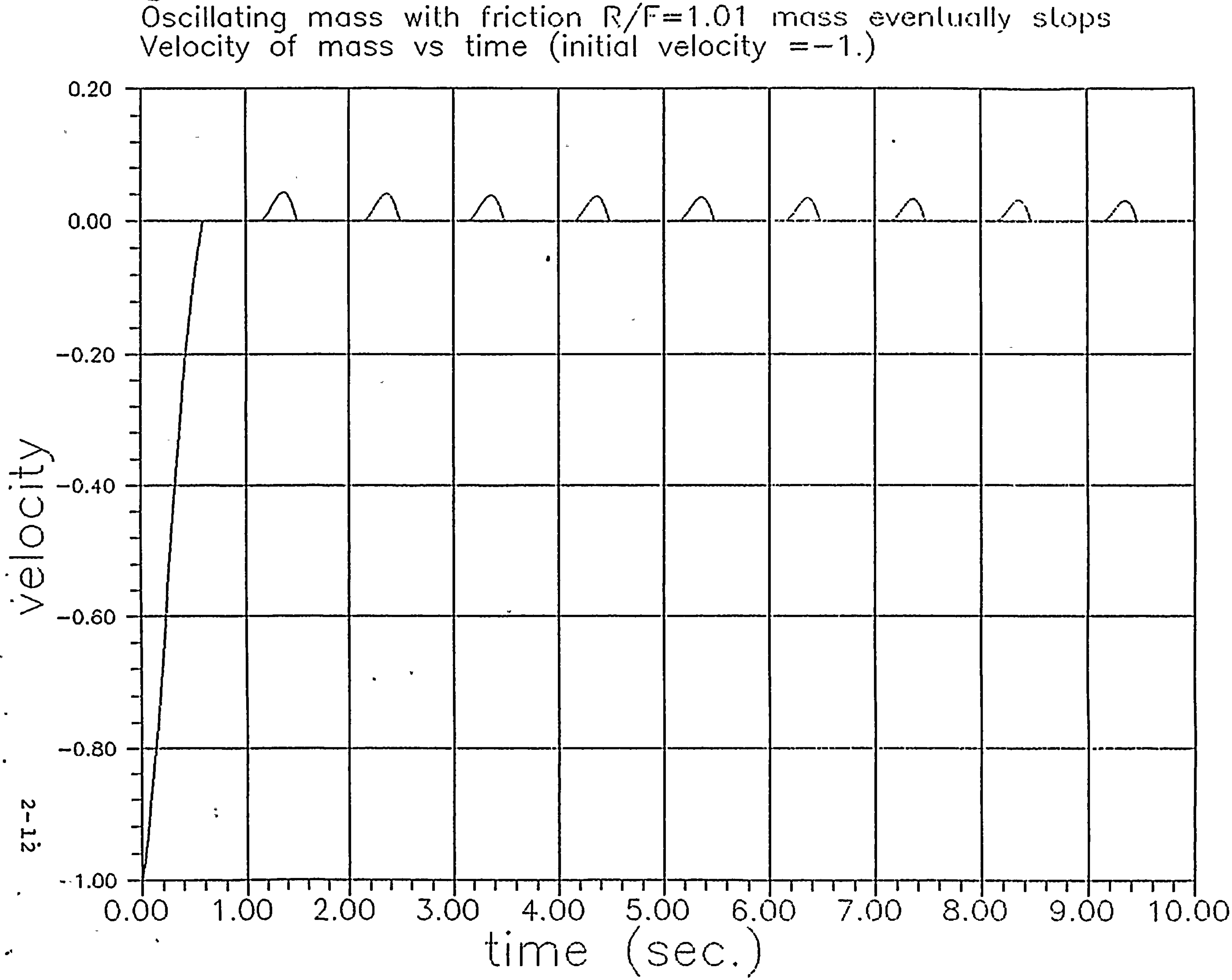

Oscillating mass with friction R/F=1.01 mass eventually stops
Velocity of mass vs time (initial velocity =-1.)
0.20
0.00
-0.20
-0.40
-0.60
-0.80
-1.00
velocity
0.00
1.00
2.00
3.00
4.00
5.00
6.00
7.00
8.00
9.00
10.00
time (sec.)

FIGURE 2.6

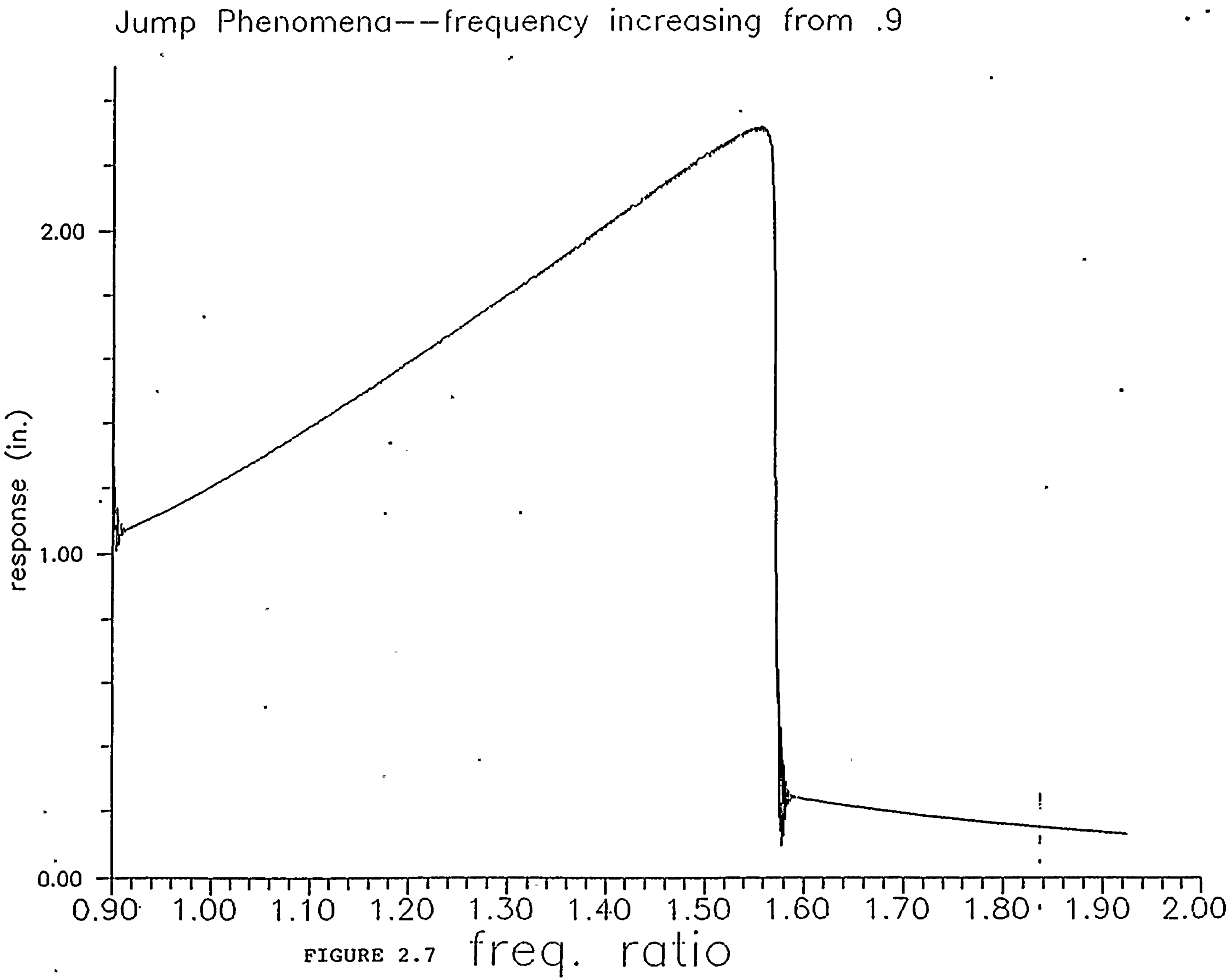
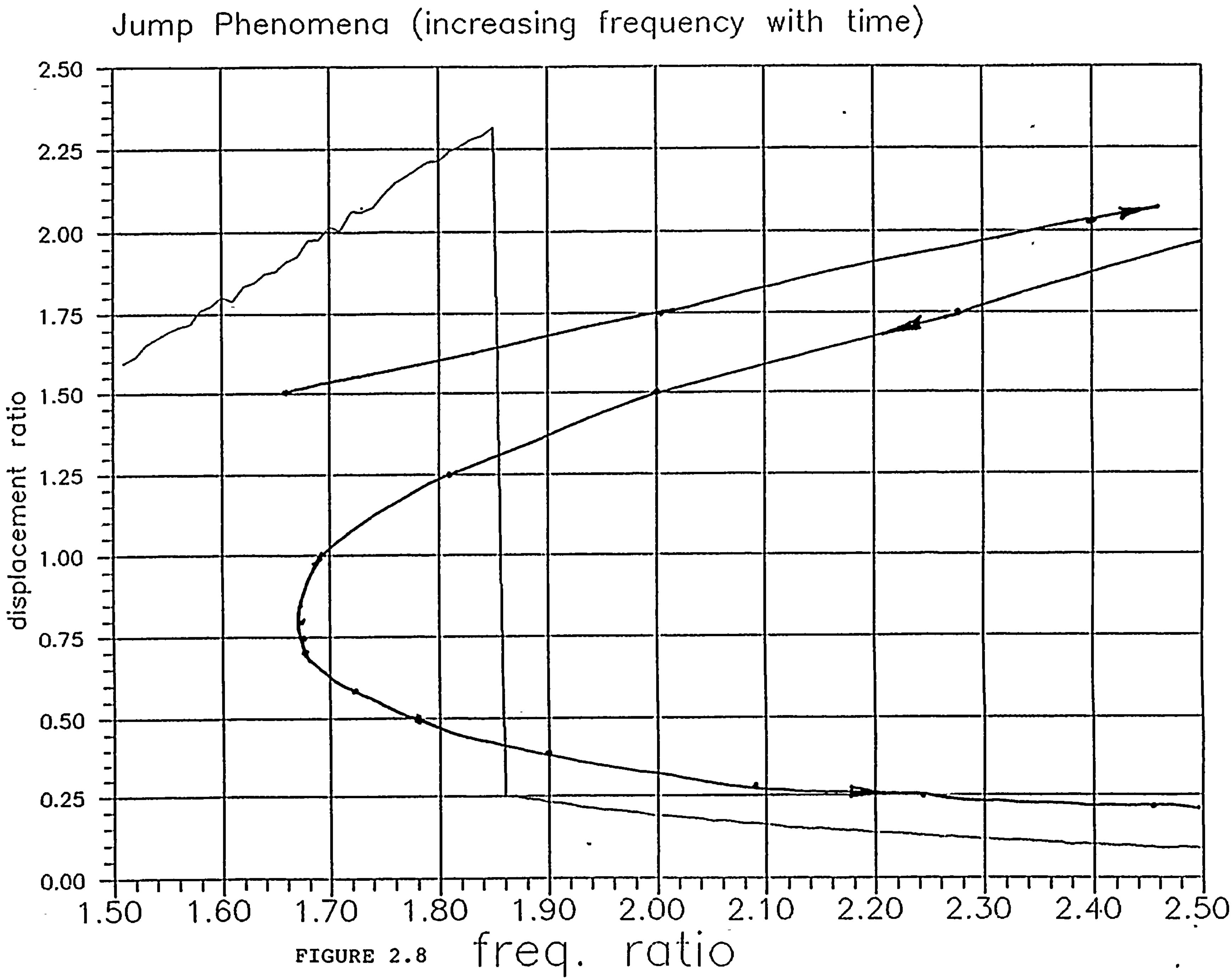

Jump Phenomena--frequency increasing from .9
response (in.)
2.00
1.00
0.00
0.90
1.00
1.10
1.20
1.30
1.40
1.50
1.60
1.70
1.80
1.90
2.00
freq. ratio

FIGURE 2.7



Jump Phenomena (increasing frequency with time)
displacement ratio
0.00
0.25
0.50
0.75
1.00
1.25
1.50
1.75
2.00
2.25
2.50
1.50
1.60
1.70
1.80
1.90
2.00
2.10
2.20
2.30
2.40
2.50
freq. ratio

FIGURE 2.8

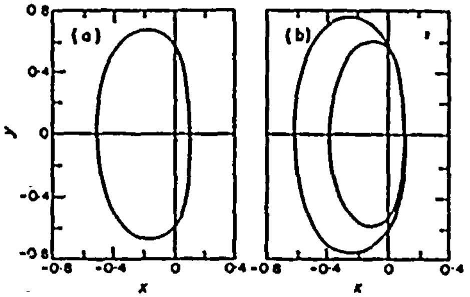


Figure 8. (a) Stable period one orbit at $\omega = 2{\cdot}40$, $\alpha = 0{\cdot}125$, $x_0 = 0$, $\beta = 1$, and $\tilde{\omega} = 4$, from digital simulation; (b) stable period two orbit at $\omega = 2{\cdot}42$, $\alpha = 0{\cdot}125$, $\beta = 1$, $x_0 = 0$, and $\tilde{\omega} = 4$, from digital simulation. Both in projected phase plane (x, y).

By using digital simulations, other subharmonic orbits are also found to exist. Figures 9(a) and (b) show the coexistence of single impact stable period one and stable period three orbits for $x_0 = 0$, $\alpha = 0{\cdot}026$, $\tilde{\omega} = \sqrt{2}$ and $\omega = 3{\cdot}5$. We conjecture that the period three orbit appeared in a saddle-node bifurcation [21] and that an unstable period three orbit also exists at these parameter values. Analysis of this bifurcation is much more difficult since no reasonable assumption regarding the times of flight can be made and higher iterates of the mapping (multiple impacts) appear to be involved.

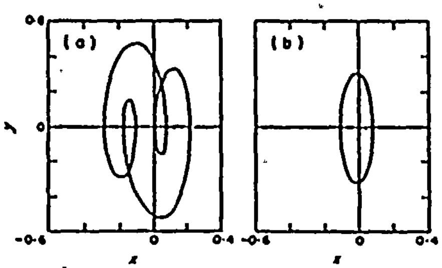


Figure 9. Coexistence of stable period three and stable period one at $\tilde{\omega} = \sqrt{2}$, $\alpha = 0{\cdot}026$, $\beta = 1$, $x_0 = 0$ and $\omega = 3{\cdot}5$. (a) Period 3; (b) period 1. Both in projected phase plane (x, y).

4.3. THE IMPACT LIMIT, ANALYSIS

In the impact limit an important simplification occurs, since the time of flight during the impact is taken to be zero. This allows more analysis to be done on periodic orbits. We also rescale: $x \to \beta x$, and take unit forcing amplitude.

The return map for the impact oscillator is very similar to the one for the general system. From points 0 to 1 in Figure 4 one uses the impact limit, i.e.,

$$t_1 = t_0, \quad \text{and} \quad y_1 = -r y_0. \tag{49}$$

From points 1 to 2 one uses the same mapping as described in the general case. Thus the mapping **P** still cannot be written down explicitly. As before, however, one can compute **DP** analytically. Moreover, in this limiting case one can compute periodic points corresponding to single impact, period n orbits directly. Such orbits correspond to those motions which strike the wall (the very stiff spring k_2) and then remain in $x < x_0$ for

exactly $2\pi n/\omega$ in time and then strike the wall again with the same velocity as the previous impact. The conditions for the existence of such an orbit are

$$t_2 - t_0 = 2\pi n/\omega, \qquad y_2 = y_0 = -y_1/r, \quad \text{with} \quad y_0 > 0. \tag{50, 51}$$

These two conditions allow one to compute the period n point $(\bar{t}, \bar{y})$. First one writes equation (7), using equations (5), (50) and (51), to obtain

$$0 = -x_0 \Gamma_- + \Lambda y_1 + s_1(\delta\Gamma_- + \Lambda\gamma\omega) + c_1(\gamma\Gamma_- - \Lambda\delta\omega), \tag{52}$$

where we have dropped the minus subscripts on γ, δ, and Ω and where $\Gamma_\pm = 1 - Ec \pm \alpha\Lambda$, $\Lambda = Es/\Omega$, $E = e^{-2\pi n\alpha/\omega}$, $s = \sin(2\pi n\Omega/\omega)$, and $c = \cos(2\pi n\Omega/\omega)$. Next one writes equation (51) using equation (50) and the time derivative of equation (5) to obtain

$$0 = y_1(-1 - r + r\Gamma_+) + \Lambda r x_0 + s_1 r(\gamma\omega\Gamma_+ - \Lambda\delta) - c_1 r(\delta\omega\Gamma_+ + \Lambda\gamma). \tag{53}$$

Since y_1 appears in a linear manner in both equations (52) and (53), it may be eliminated to obtain a single equation involving only t_1 as an unknown (in the terms $c_1 = \cos(\omega t_1)$ and $s_1 = \sin(\omega t_1)$):

$$\begin{aligned} 0 = (x_0/\Lambda)[r\Lambda^2 - \Gamma_-\psi] &+ s_1[r\gamma\omega\Gamma_+ - \Lambda\delta r + (\psi/\Lambda)(\delta\Gamma_- + \Lambda\gamma\omega)] \\ &+ c_1[-r\delta\omega\Gamma_+ - \Lambda\gamma r + (\psi/\Lambda)(\gamma\Gamma_- - \Lambda\delta\omega)], \end{aligned} \tag{54}$$

where $\psi = 1 + r - r\Gamma_+$. Straightforward association of terms allows this equation to be written as

$$0 = X + s_1 Y + c_1 Z, \tag{55}$$

which has a solution

$$\bar{t}_1 = (1/\omega)[\arctan(Y/Z) + \arccos(-X/W)], \tag{56}$$

where $W = \sqrt{Y^2 + Z^2}$. This expression gives the time (i.e., forcing phase) at impact on the period n orbit. The velocity just after impact $\bar{y}_1$ is then easily computed by using either equation (52) or equation (53).

It is important to note that a solution obtained as described above only satisfies $x(t_1) = x_0$ and $\dot{x}(t_1) = -r\dot{x}(t_2)$. If the value of $\bar{y}_1 = \dot{x}(t_1)$ is positive, then the solution corresponds to a non-physical, or "penetrating" orbit [13]. The orbits for $\bar{y}_1$ negative must also be checked since nowhere has one been assured that the desired x_0 crossing is the first on the orbit. In fact, the above conditions can be satisfied after several x_0 crossings, for some parameter values. Care must be taken to determine which of these orbits are physically possible.

Knowing the periodic point, one can now compute its stability. As before, one breaks the calculation of **DP** into two parts, from point 0 to 1 and from point 1 to point 2. Here

$$\left[\frac{\partial(t_1, y_1)}{\partial(t_0, y_0)}\right] = \begin{bmatrix} 1 & 0 \\ 0 & -r \end{bmatrix}, \quad \text{and} \quad \mathbf{DP} = \left[\frac{\partial(t_2, y_2)}{\partial(t_1, y_1)}\right]\begin{bmatrix} 1 & 0 \\ 0 & -r \end{bmatrix}, \tag{57, 58}$$

where $[\partial(t_2, y_2)/\partial(t_1, y_1)]$ was derived above in the analysis of the finite stiffness case. From this calculation one finds that **DP** has determinant

$$D = (-r y_1/y_2)\, e^{-2\alpha(t_2 - t_0)}, \tag{59}$$

and trace

$$T = [e^{-\alpha(t_2 - t_0)}/\Omega y_2][(N_1 + rN_2)\sin(\Omega(t_2 - t_0)) + \Omega(y_1 - r y_2)\cos(\Omega(t_2 - t_0))]. \tag{60}$$

Evaluating on a period n orbit gives

$$\bar{D} = r^2 E^2 \quad \text{and} \quad \bar{T} = (E/\Omega\bar{y}_0)[(1 + r)(\bar{c}_0 + x_0)s - 2r\bar{y}_0\Omega c], \tag{61, 62}$$

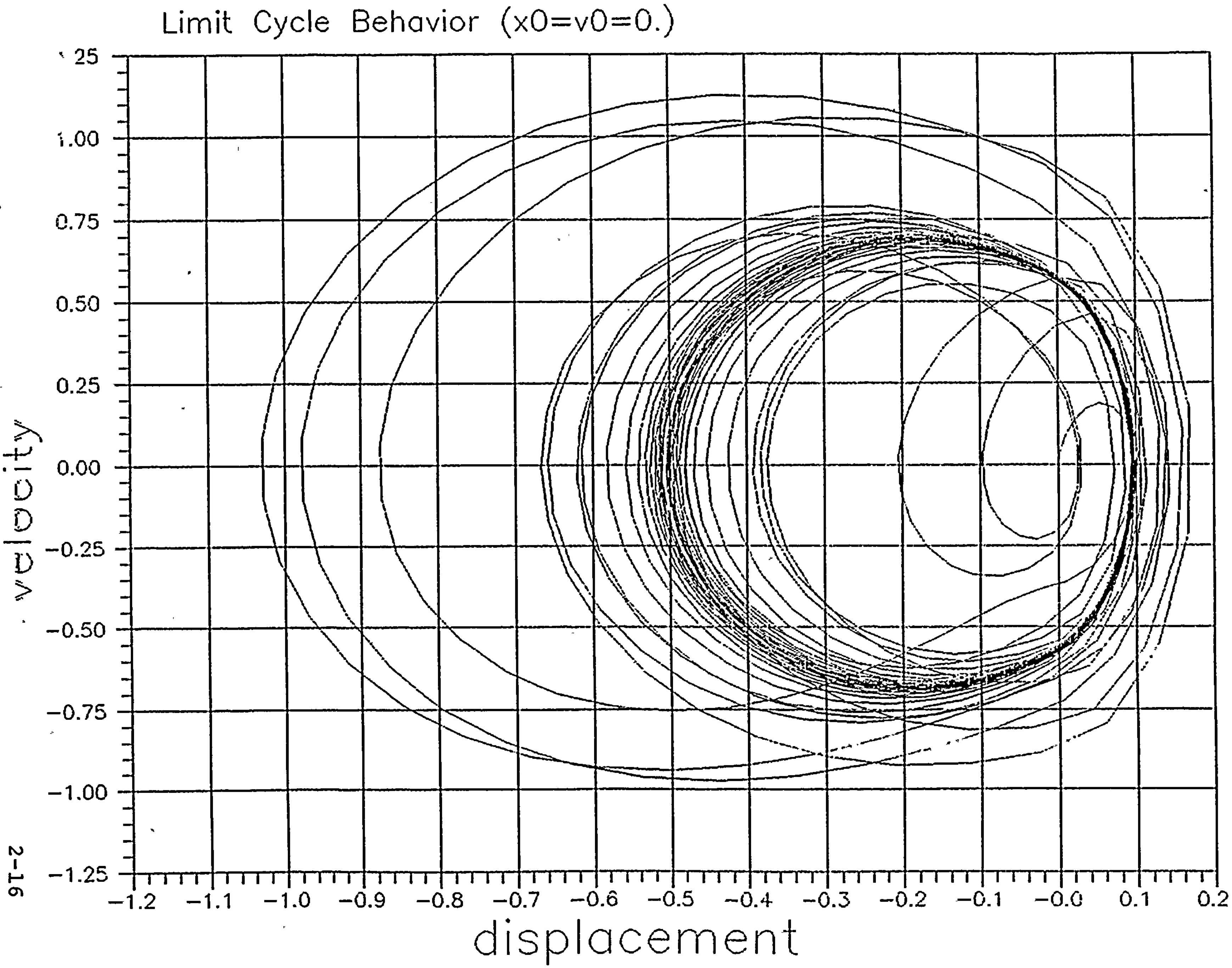

Limit Cycle Behavior (x0=v0=0.)
velocity
displacement
1.25
1.00
0.75
0.50
0.25
0.00
−0.25
−0.50
−0.75
−1.00
−1.25
−1.2
−1.1
−1.0
−0.9
−0.8
−0.7
−0.6
−0.5
−0.4
−0.3
−0.2
−0.1
−0.0
0.1
0.2

FIGURE 2.10

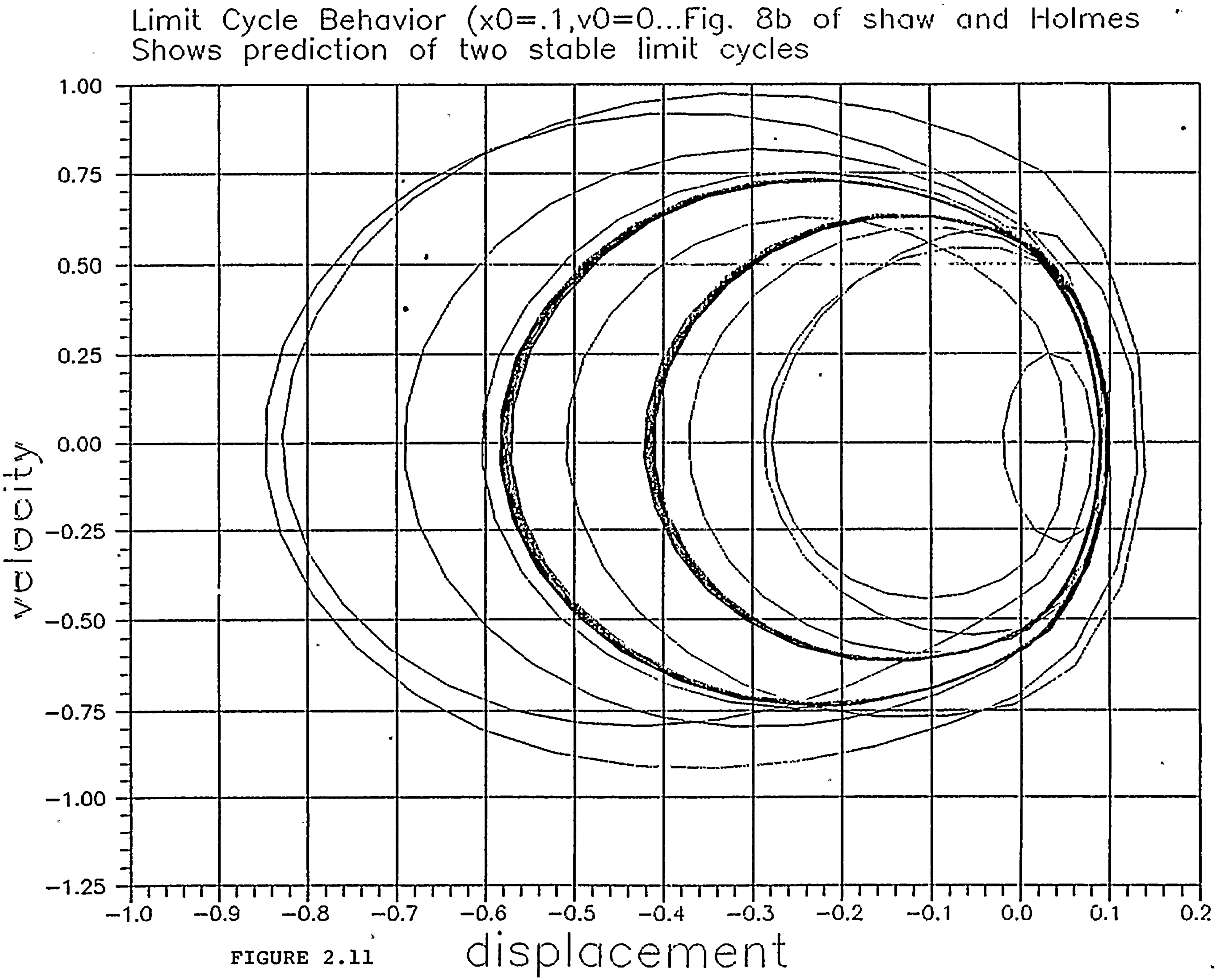

Limit Cycle Behavior (x0=.1,v0=0...Fig. 8b of shaw and Holmes
Shows prediction of two stable limit cycles
velocity
displacement
1.00
0.75
0.50
0.25
0.00
−0.25
−0.50
−0.75
−1.00
−1.25
−1.0
−0.9
−0.8
−0.7
−0.6
−0.5
−0.4
−0.3
−0.2
−0.1
0.0
0.1
0.2

FIGURE 2.11

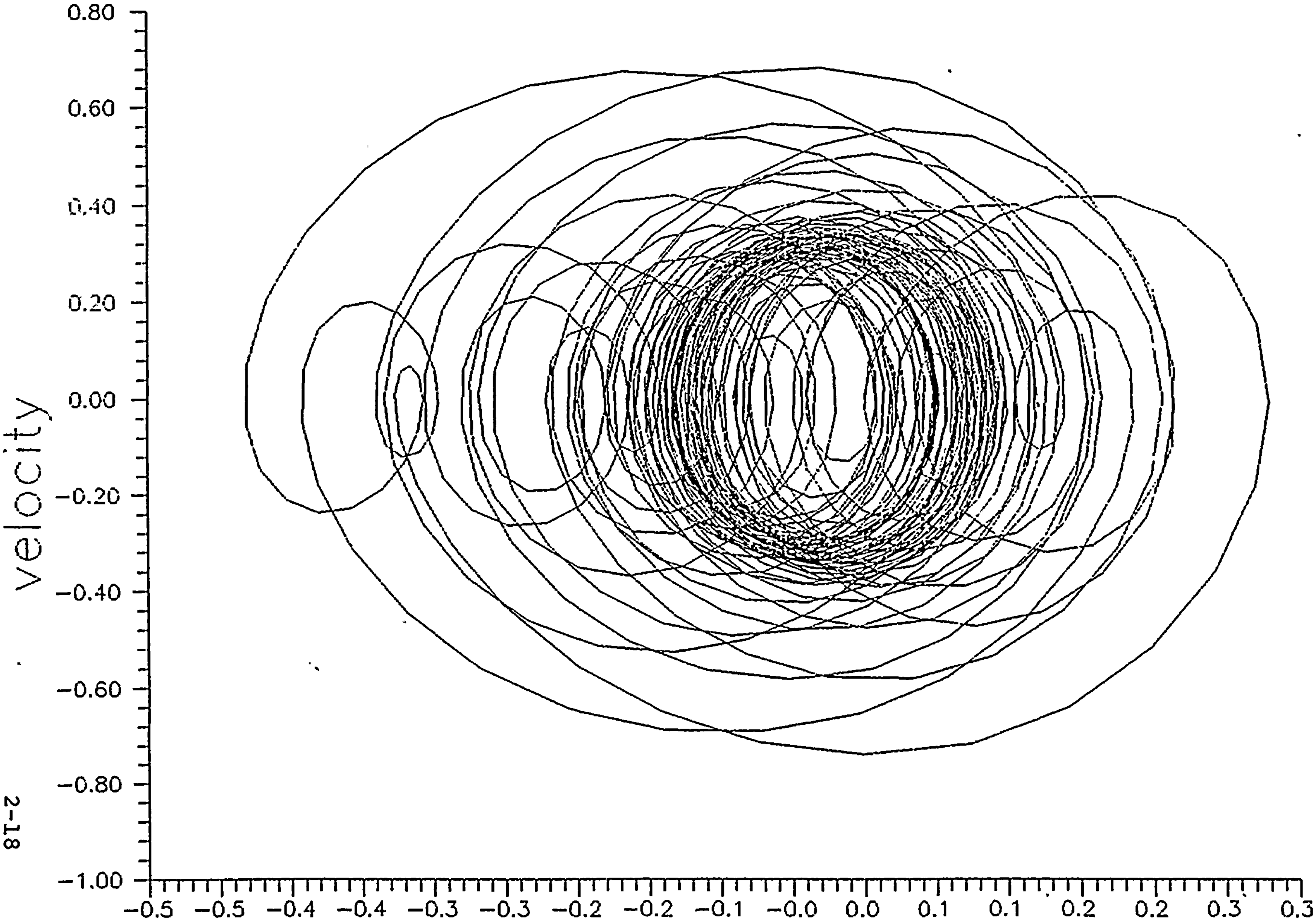

velocity
0.80
0.60
0.40
0.20
0.00
−0.20
−0.40
−0.60
−0.80
−1.00
−0.5 −0.5 −0.4 −0.4 −0.3 −0.3 −0.2 −0.2 −0.1 −0.0 0.0 0.1 0.1 0.2 0.2 0.3 0.3
displacement

FIGURE 2.12

Limit Cycle Behavior of a Bi-Linear Spring. x0=.2, v0=0
Fig. 9a shaw and holmes..cycles of order 1 and 3

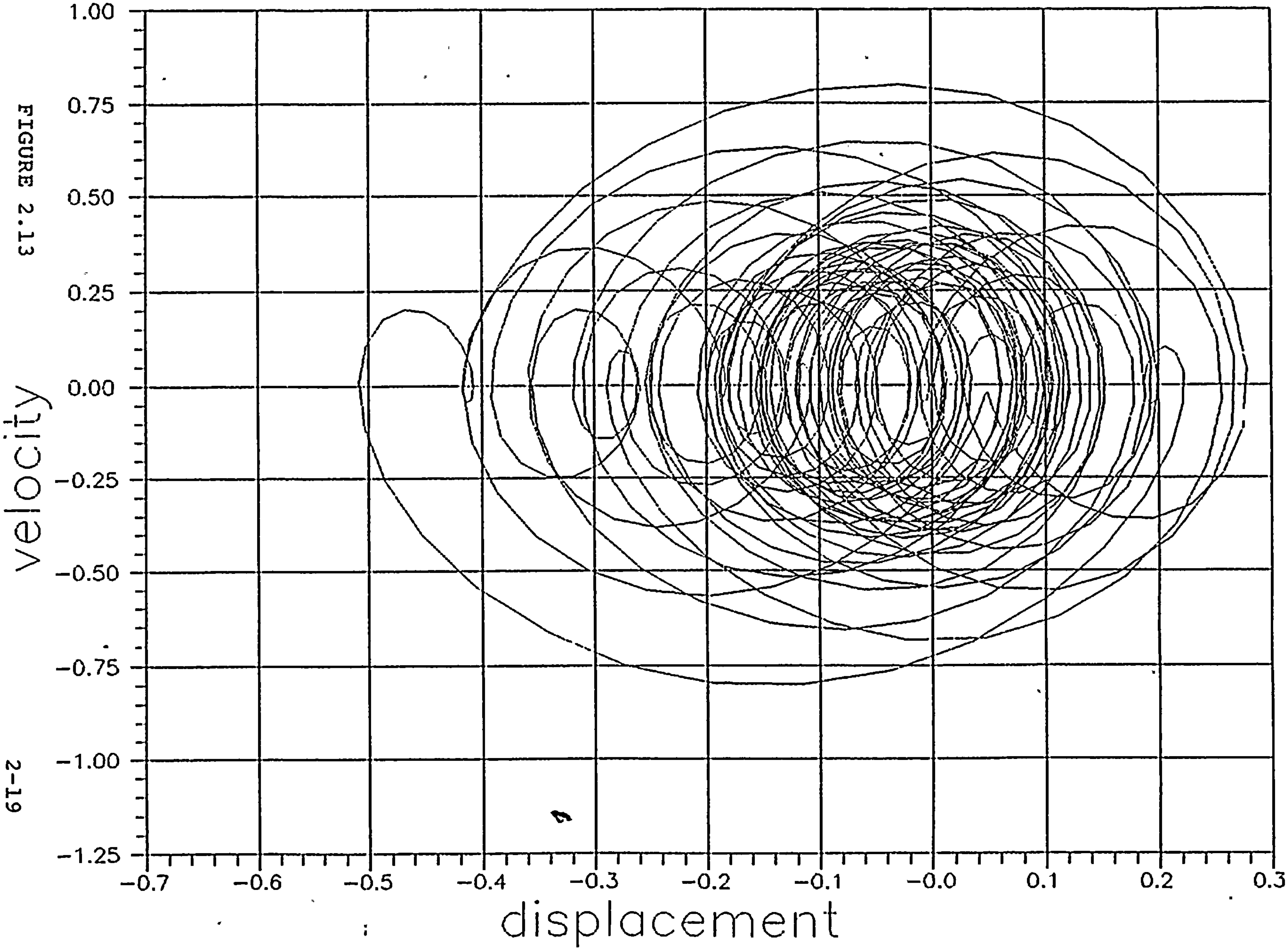


FIGURE 2.13

2-19

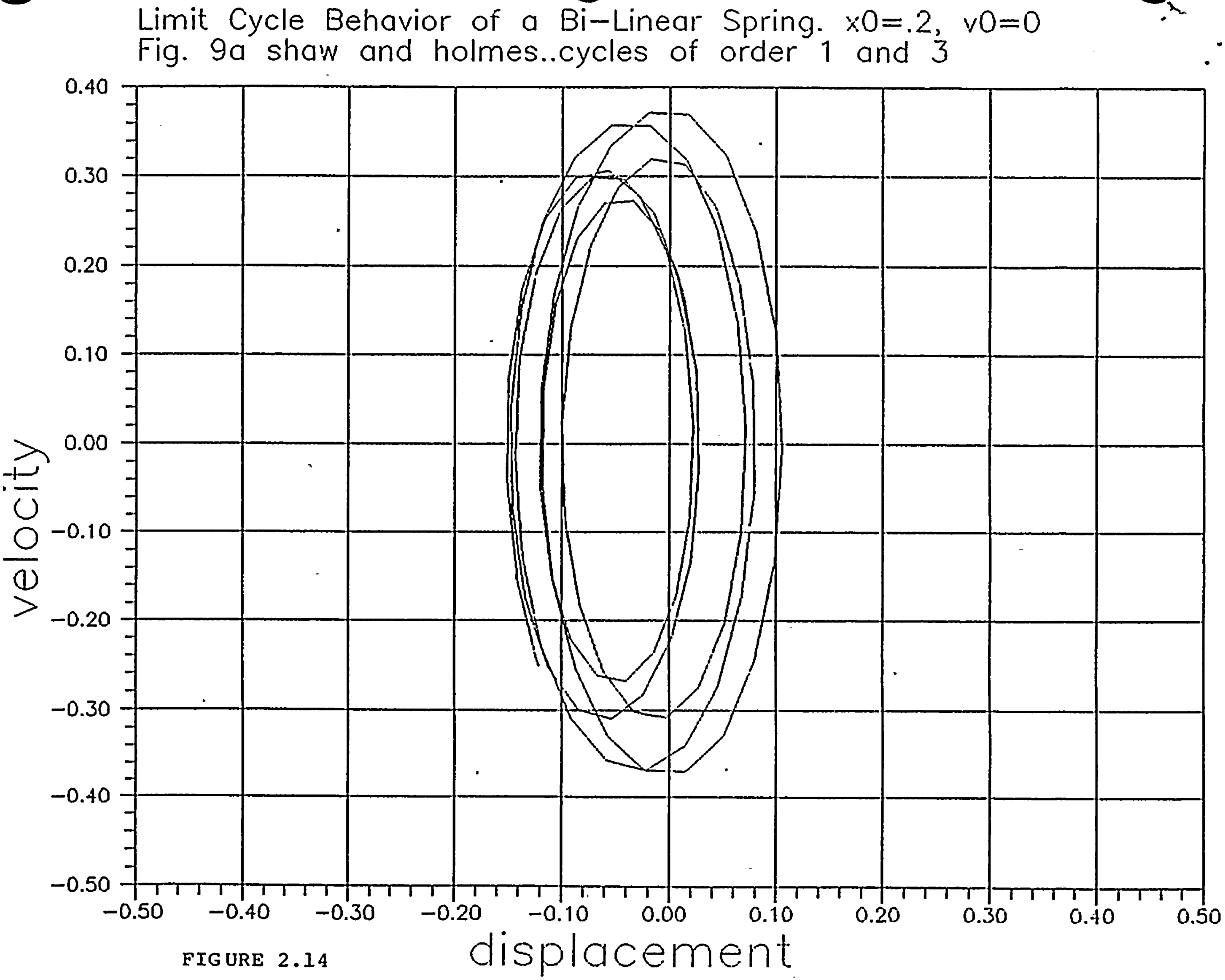

Limit Cycle Behavior of a Bi-Linear Spring. x0=.2, v0=0
Fig. 9a shaw and holmes..cycles of order 1 and 3
velocity
displacement
0.40
0.30
0.20
0.10
0.00
-0.10
-0.20
-0.30
-0.40
-0.50
-0.50
-0.40
-0.30
-0.20
-0.10
0.00
0.10
0.20
0.30
0.40
0.50

FIGURE 2.14

APPENDIX A

SEC. 2.14 From Textbook
"Component Element Method", McGraw Hill (1976)
by S. Levy and J. Wilkinson

(Subharmonic Resonance)

in the third, and 60 Hz to 120 Hz in the fourth. Let the g-level be $0{\cdot}31g$ at $t = 0$, $1{\cdot}25g$ at $t = 1{\cdot}0$, $5{\cdot}0g$ at $t = 2{\cdot}0$, $5{\cdot}0g$ at $t = 3{\cdot}0$, and $0{\cdot}0g$ at $t = 4{\cdot}0$. This description corresponds to constant displacement amplitude excitation below 30 Hz, and a roll-off to $0g$ between 60 Hz and 120 Hz.

2.46 In some cases at resonance, the forces are severe enough to cause a bolt to lift off its seat. In such a case, the effective spring stiffness drops markedly at lift-off. Rerun the example problem with $c = 0{\cdot}15$ lb sec/in. and add $k_u = -800$ lb/in., $y_u = 0{\cdot}222$ in., corresponding to lift-off at $50g$ ($900 \times 0{\cdot}222/4 = 50g$). The total stiffness after lift-off is $900 - 800 = 100$ lb/in. How does the peak amplitude compare with the value in Fig. 2.38?

2.47 Repeat problem 2.46 for decreasing frequency. (*Hint:* Let frequency be 100, 60, 30, 5 and g-level be 1, 5, 5, 1 for times 0·0, 1·0, 2·5, and 3·0, respectively.) Note the difference in response for increasing and decreasing frequency for this nonlinear system.

2.48 Repeat problems 2.46 and 2.47 with $c = 0{\cdot}20$. Does the increased damping cause a marked decrease in amplitude for this nonlinear problem?

2.14 SUBHARMONIC RESONANCE

Subharmonic resonance can occur in mechanical systems if the spring force is nonlinear. As an example, we will consider a system for which

$$\text{spring force} = 90(y - x) + 10(y - x)^2$$

We take the mass as 0·1036 lb sec²/in., so that the low amplitude natural frequency is 4·7 Hz. The system is excited by a support movement at 9·4 Hz.

The response is shown in Fig. 2.39. The response is quite nonlinear with both $c = 0{\cdot}0$ and $c = 0{\cdot}3$; however, it is evident that the 4·7 Hz component is much larger than the 9·4 Hz component. An interesting feature of the response curves is their tendency to treat the $+5g$ level as a barrier. In the spring force sketch, we note that the maximum compressive force developed by the nonlinear spring is 202·5 lb at 4·5 in. compression. Here m represents a weight of 40 lb, so the spring can at most apply $202{\cdot}5/40 = 5{\cdot}06g$ of upward acceleration. The damper might modify this value slightly. For the spring elongation, however, the restoring force becomes quite high and results in downward accelerations as high as $15g$.

Problems

2.49 Repeat the example problem with the support g-level increased from $10g$ to $15g$.

2.50 Repeat the example problem with the exciting frequency at three times the low amplitude natural frequency.

2.51 Repeat problems 2.49 and 2.50 with $k_2 = 5$ lb/in. instead of 10 lb/in.

2.15 SUMMARY

This chapter has concerned itself with the single mass system. In general, restoring forces of the system were nonlinear, consisting of nonlinearly hardening or softening springs, friction, or stops. Numerous examples of such systems were studied by means of a computer program given in FORTRAN in section 2.4 of this chapter. The numerical examples were designed to illustrate each feature of the computer program:

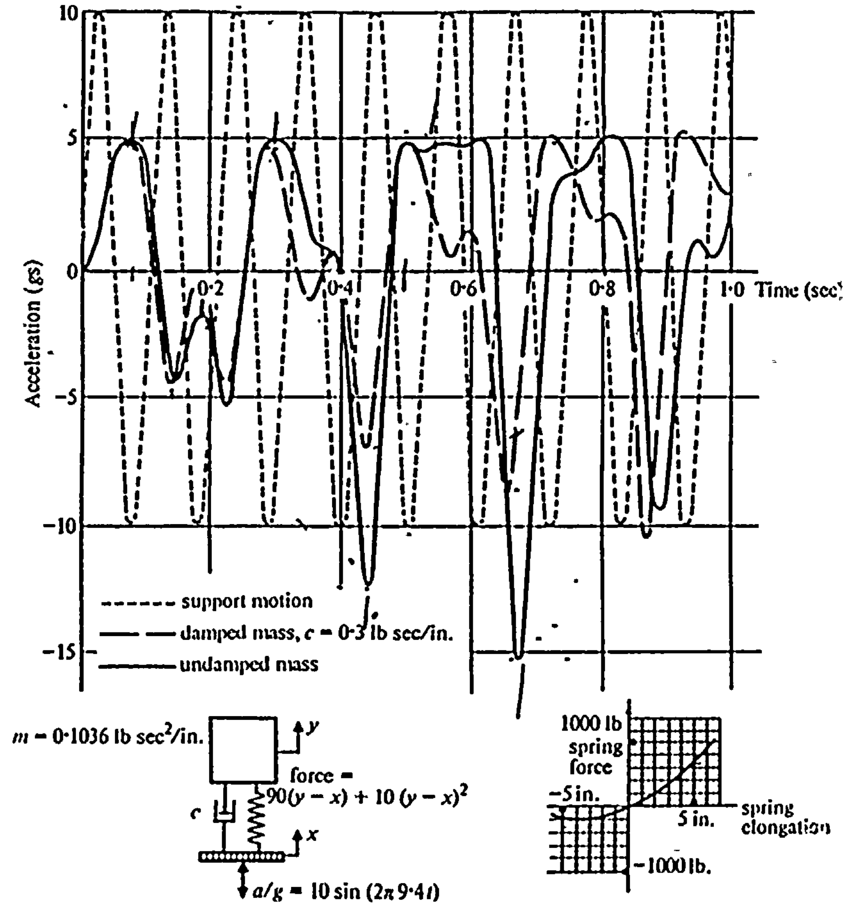


FIGURE 2.39
Response of nonlinear system with spring force = 90 (displacement) + 10 (displacement)² at excitation at 9·4 Hz. Low amplitude natural frequency $\omega_0 = 4{\cdot}7$ Hz. Response shows subharmonic resonance effect

namely, each element (springs of various types, stops, friction), and each type of excitation (forces on the mass, acceleration of the support, and time-varying excitations of one kind or another). In addition, we made comparisons between the numerical results of the computer program and some of the known solutions existing in the literature for nonlinear systems.

We have reached a point where we can inquire into the behavior of a system having more than one mass, each mass interacting with others by means of springs, dampers, friction, and stops. Such multi-mass systems will be discussed in the next chapter, where computer programs suitable for their study will also be given.

APPENDIX B

(SLIDING FRICTION AND DEAD BANDS)

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Static and Sliding Friction in Feedback Systems

J. Tou* and P. M. Schultheiss
Department of Electrical Engineering, Yale University, New Haven, Connecticut
(Received January 16, 1953)

One of the most common nonlinearities encountered in servomechanisms design is the friction phenomenon in electromechanical systems. Conventional linear theory fails to predict its effect upon system performance. This paper extends familiar techniques to the treatment of friction nonlinearity in servosystems. Frequency-response methods are employed throughout and the theoretical results are verified by means of an analog computer. Sliding friction and static friction are represented by describing functions which form the critical factors in determining system stability. The analysis indicates that certain series equalizers designed from linear theory may fail to achieve effective compensation in the presence of sliding and static friction. On the other hand, a subsidiary loop may avoid the stability problem while still realizing an essentially equivalent loop gain function

I. INTRODUCTION

WHILE basic analysis and synthesis procedures for linear feedback systems have become well established during the last decade, there is no correspondingly broad approach to nonlinear problems. Except in very simple cases, no general solutions are possible, and the designer must rely either on machine computation or on various linear or quasi-linear approximations. A variety of such approximations has been developed to fit numerous types of systems and successful design procedures have been discovered for a great many practical problems. It is the purpose of this paper to extend one of these techniques so as to make it applicable to the analysis of feedback systems involving sliding and static friction. Particular attention will be paid to certain loop gain functions which appear to be quite satisfactory on the basis of linear analysis but are found to be unstable in practice as a result of friction phenomena. Methods of predicting, and hence presumably preventing, such behavior will be outlined.

II. REVIEW OF BASIC PROCEDURE

The technique to be employed was first devised by Kochenburger[1] for the analysis of contractor servomechanisms and subsequently adapted for use with other nonlinear devices.[2,3] The basic procedure has been described extensively in the literature[1-5] and will therefore be outlined only briefly. It is unique in that it permits use of the frequency domain in an approach to problems involving certain types of nonlinear elements. If a sinusoidal voltage is applied to a nonlinear device, the output is generally not sinusoidal. However, under rather general conditions the fundamental component of the output will be greater than any harmonic, a difference which will be further emphasized by effective low-pass filters such as servomotors. Adequate accuracy can therefore be obtained in many cases by considering only the fundamental component of the output.[1-5] Since the amplitude and phase of the fundamental component varies with the amplitude of the applied sinusoid, the approximate characteristics of the nonlinear device are represented by an "amplitude-describing function" $H_0(x)=f(x)e^{j\theta(x)}$ (see Fig. 1). If x represents the amplitude of a sinusoidal input signal, $f(x)$ is the ratio of the fundamental output to the input amplitude and $\theta(x)$ is the phase shift of the output fundamental relative to the input signal. Note that $H_0(x)$ is frequency invariant, it depends only on the input amplitude.

Once $H_0(x)$ is known, a stability analysis can proceed essentially as in the linear case. Consider the simple loop shown in Fig. 2. System stability is governed by the roots of the equation

$$1+H_0(x)H(s)=0 \qquad (1)$$

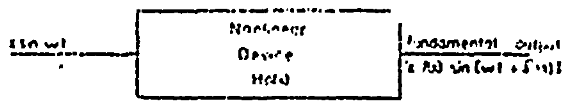

Fig. 1. Describing function of a nonlinear device.

or

$$H(s)=-1/H_0(x). \qquad (2)$$

In the linear case $H_0(x)=1$ and the stability problem reduces to the conventional one, solved easily by means of a Nyquist plot. The only modification required for the nonlinear case under the assumptions stated is a change in the critical point which now becomes $-1/H_0(x)$ instead of -1. Thus the critical point changes with the signal amplitude, and it becomes necessary to plot an amplitude locus $-1/H_0(x)$ in addition to the frequency locus $H(s)$. If the amplitude locus lies completely outside the frequency locus, the system is stable under all conditions of operations.[4] Figure 3 shows intersecting loci. Here the system is unstable for small disturbances, but stable for large disturbances so

* Now with Philco Corporation, Philadelphia, Pennsylvania.
[1] R. J. Kochenburger, Elec. Eng. 69, 687 (1950). See also Trans. Am. Inst. Elec. Engrs. 69, 270 (1950).
[2] E. C. Johnson, dissertation, Massachusetts Institute of Technology, 1951; Trans. Am. Inst. Elec. Engrs. 71II, 169 (1952).
[3] E. S. Sherrard, Trans. Am. Inst. Elec. Engrs. 71II, 312 (1952).
[4] Frequently the inverse loci, $1/H(s)$ and $-H_0(x)$, are plotted. The choice is governed simply by computational convenience in particular instances.

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that oscillations will tend to stabilize near the intersection point P which thus specifies the steady-state conditions, at least to a first approximation.[1]

In summary, analysis of the stability problem will require the following steps.

(a) Determination of the wave form at the output of the nonlinear device resulting from a sinusoidal input.

(b) Calculation of the describing function $H_0(x)$ from the wave shape obtained in (a).

(c) Plot and interpretation of the amplitude and frequency loci for the system under consideration. For the cases of particular interest here this step requires rearrangement of the conventional block diagram in order to secure effective separation of all transfer functions into two classes: The class of all linear but frequency sensitive components and that of all nonlinear but frequency insensitive elements.

The following definitions will be used throughout this paper. Static friction is the torque required to initiate rotation. Sliding friction is the velocity-independent component of the torque necessary to maintain such motion once started. Viscous friction is that component of the torque which is linearly proportional to the angular velocity of the rotating member.

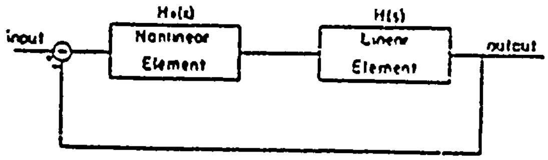

FIG. 2. Feedback loop with nonlinear element.

III. SLIDING FRICTION IN SERVOSYSTEMS

A. Wave Form Resulting from a Sinusoidal Input Torque to a System with Sliding Friction

If only sliding friction is considered the entire friction phenomenon can be represented by the characteristic curve of Fig. 4.

Consider a rotating member with moment of inertia J and angular acceleration $\ddot{\theta}$. Because of sliding friction the effective accelerating or decelerating torque τ_e is related to the applied torque τ_a through the equation

$$\tau_e = \tau_a \pm T_c, \qquad (3)$$

where T_c is defined by Fig. 4. From Newton's law of motion,

$$\tau_a = T_c + J\ddot{\theta} \text{ for angular velocity } \dot{\theta} > 0 \qquad (4)$$

$$\tau_a = -T_c + J\ddot{\theta} \text{ for } \dot{\theta} < 0. \qquad (5)$$

From Eqs. (3), (4), and (5), the angular acceleration of the rotating member is given by

$$\ddot{\theta}(t) = \tau_e(t)/J. \qquad (6)$$

Hence $\ddot{\theta}(t)$ has the same wave form as $\tau_e(t)$.

If the applied torque τ_a is sinusoidal,

$$\tau_a(t) = \Gamma_a \sin\omega t. \qquad (7)$$

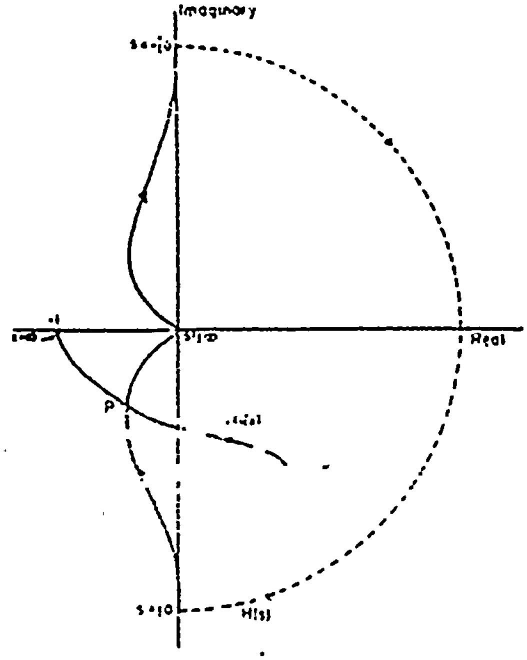

FIG. 3. Amplitude and frequency loci.

The corresponding steady-state wave forms are sketched in Figs. 5 and 6. The effective torque wave derived from Eq. (3) is shown in dotted lines. The discontinuities of the τ_e wave correspond to zeros of the $\dot{\theta}$ wave because the frictional torque T_c changes sign at those instants. On the $\dot{\theta}$ curve, point P is the point of inflection, corresponding to maximum acceleration. Since the steady state is of primary interest, the reference time is chosen after the oscillation has reached its steady-state value. $\dot{\theta}(t)$ passes through zero at

$$\omega t = n\pi - \alpha, \quad n = 0, 1, 2, 3 \cdots,$$

while

$$\tau_a = T_c \quad \text{at} \quad \omega t = \alpha.$$

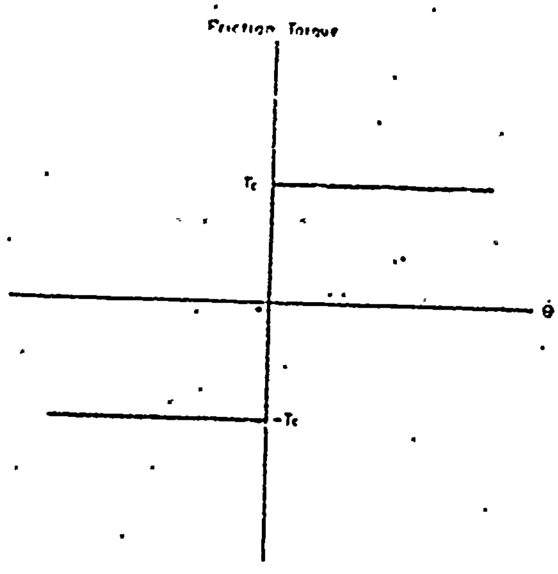

FIG. 4. Sliding friction characteristic.

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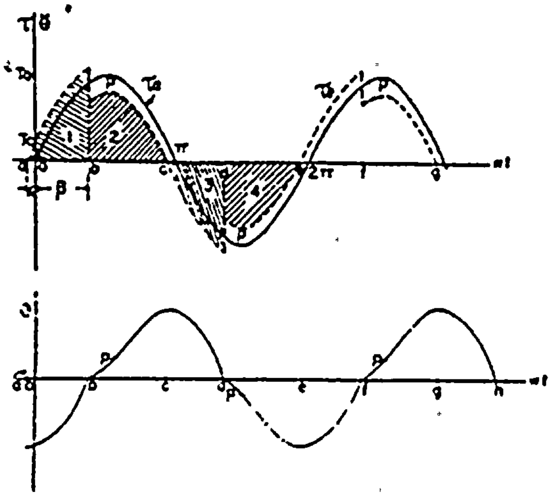

FIG. 5. Steady-state wave forms without dead zones.

It follows that

$$\alpha = \sin^{-1}\lambda, \quad (8)$$

where

$$\lambda = T_c/T_a. \quad (9)$$

Only the angle β corresponding to the first discontinuity point remains unknown. Once it has been evaluated in terms of λ, the wave form is completely determined. There are two possibilities: If $\alpha \leq \beta$, there is no dead zone in the τ_e wave (Fig. 5). If $\alpha > \beta$, there are dead zones as indicated in Fig. 6. These two cases will be considered separately.[5]

Mathematical Representation of the Steady-State Wave Forms

Case (1).—No dead zone. $\alpha \leq \beta$.

Refer to Fig. 5. In the absence of viscous friction, the following steady-state conditions exist.

Shaded area No. 1 = shaded area No. 2
= shaded area No. 3
= shaded area No. 4, etc.[6]

But between a and b,

$$\tau_e = T_a \sin\omega t + T_c$$
$$= T_a(\sin\omega t + \sin\alpha) = T_a(\sin\omega t + \lambda); \quad (10)$$

and between b and c,

$$\tau_e = T_a \sin\omega t - T_c$$
$$= T_a(\sin\omega t - \sin\alpha) = T_a(\sin\omega t - \lambda). \quad (11)$$

Then

$$\text{area No. } 1 = \int_{-\alpha}^{\beta} T_a(\sin\omega t + \sin\alpha)d(\omega t)$$
$$= T_a(-\cos\beta + \beta\sin\alpha + \cos\alpha + \alpha\sin\alpha), \quad (12)$$

and

$$\text{area No. } 2 = \int_{\beta}^{\pi-\alpha} T_a(\sin\omega t - \sin\alpha)d(\omega t)$$
$$= T_a[\cos\alpha - (\pi-\alpha)\sin\alpha + \cos\beta + \beta\sin\alpha]. \quad (13)$$

If Eqs. (12) and (13) are set equal and simplified, the result is

$$\cos\beta = \pi\sin\alpha/2$$

or

$$\beta = \cos^{-1}(\pi\lambda/2). \quad (14)$$

For the extreme case, $\beta = \alpha$, Eq. (14) becomes

$$\sin^{-1}\lambda = \cos^{-1}(\pi\lambda/2)$$

or

$$\lambda^2 + (\pi\lambda/2)^2 = 1.$$

A solution for λ yields

$$\lambda = \lambda_c = 0.536. \quad (15)$$

λ_c is the critical value of the quantity T_c/T_a. There is no dead zone for $\lambda \leq \lambda_c$, and there are dead zones for $\lambda > \lambda_c$.

Case (2).—With dead zones. $\alpha > \beta$.

In like manner, one obtains from Fig. 6:

shaded area No. 1 = shaded area No. 2
= shaded area No. 3
= shaded area No. 4, etc.

But, between a and b,

$$\tau_e = T_a(\sin\omega t + \lambda), \quad (16)$$

between b and c,

$$\tau_e = 0; \quad (17)$$

and between c and d,

$$\tau_e = T_a(\sin\omega t - \lambda). \quad (18)$$

Hence,

$$\text{area No. } 1 = \int_{-\alpha}^{\beta} T_a(\sin\omega t + \lambda)d(\omega t)$$
$$= T_a(-\cos\beta + \beta\lambda + \cos\alpha + \alpha\lambda), \quad (19)$$

$$\text{area No. } 2 = \int_{\alpha}^{\pi-\alpha} T_a(\sin\omega t - \lambda)d(\omega t)$$
$$= T_a[\cos\alpha - (\pi-\alpha)\lambda + \cos\alpha + \alpha\lambda]. \quad (20)$$

Equating (19) and (20) and simplifying

$$\lambda\beta - \cos\beta = (1-\lambda^2)^{\frac{1}{2}} - (\pi - \sin^{-1}\lambda)\lambda. \quad (21)$$

For the extreme case $\alpha = \beta = \sin^{-1}\lambda$,

$$\lambda \sin^{-1}\lambda - (1-\lambda^2)^{\frac{1}{2}} = (1-\lambda^2)^{\frac{1}{2}} - (\pi - \sin^{-1}\lambda)\lambda,$$

or

$$\lambda = \lambda_c = 0.536 \text{ as before.} \quad (15)$$

[5] Note that a dead zone or region of zero effective torque and velocity such as bc on Fig. 6 occurs whenever the applied torque is smaller in magnitude than T_c at the instant when the velocity reaches zero.

[6] Velocity is proportional to the integral of torque in the absence of viscous friction.

B. Calculation of the Describing Function

Since $\tau_e(t)$ is a periodic function of time, it can be expressed in terms of a Fourier series:

$$\tau_e(t)=\frac{b_0}{2}+\sum_{n=1}^{\infty}(a_n\sin n\omega t+b_n\cos n\omega t). \tag{22}$$

It has been pointed out that, to a first approximation, $\tau_e(t)$ can be represented by the fundamental component of its Fourier series. From symmetry considerations, $b_0=0$. This implies oscillation about the rest position which is a condition of primary interest in stability analyses.[7] Then

$$\tau_e(t)\doteq a_1\sin\omega t+b_1\cos\omega t, \tag{23}$$

where

$$a_1=\frac{1}{\pi}\int_{\theta}^{2\pi+\theta}\tau_e(t)\sin(\omega t)d(\omega t) \tag{24}$$

and

$$b_1=\frac{1}{\pi}\int_{\theta}^{2\pi+\theta}\tau_e(t)\cos(\omega t)d(\omega t). \tag{25}$$

Evaluation of the Fourier Coefficients

Case (1).—No dead zone, $\lambda\leq\lambda_c$ or $\alpha\leq\beta$.

$$\frac{2}{\pi}\int_{-\alpha}^{\beta}T_a(\sin\omega t+\lambda)\sin(\omega t)d(\omega t)$$

$$+\frac{2}{\pi}\int_{\beta}^{\pi-\alpha}T_a(\sin\omega t-\lambda)\sin(\omega t)d(\omega t)$$

$$=T_a(1-2\lambda^2). \tag{26}$$

Similarly

$$b_1=2T_a\lambda[(2/\pi)^2-\lambda^2]^{\frac{1}{2}}. \tag{27}$$

Equation (23) may also be written in the form

$$\tau_e(t)=C_1\sin(\omega t+\delta), \tag{28}$$

where

$$C_1=(a_1^2+b_1^2)^{\frac{1}{2}}=T_a\left[1-4\left(1-\frac{4}{\pi^2}\right)\lambda^2\right]^{\frac{1}{2}} \tag{29}$$

and

$$\delta=\tan^{-1}\frac{b_1}{a_1}=\tan^{-1}\frac{2\lambda[(2/\pi)^2-\lambda^2]^{\frac{1}{2}}}{1-2\lambda^2}. \tag{30}$$

Hence, with an applied torque $\tau_a(t)=T_a\sin\omega t$, the

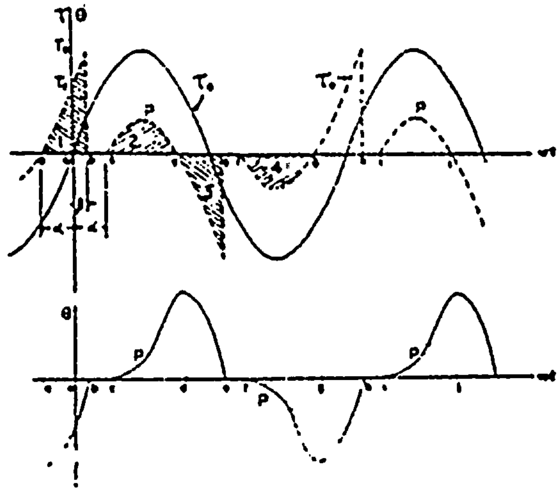

FIG. 6. Steady-state wave forms with dead zones.

effective torque is

$$\tau_e(t)=T_a\left[1-4\left(1-\frac{4}{\pi^2}\right)\right]^{\frac{1}{2}}$$

$$\times\sin\left[\omega t+\tan^{-1}\frac{2\lambda[(2/\pi)^2-\lambda^2]^{\frac{1}{2}}}{1-2\lambda^2}\right]. \tag{31}$$

In accordance with the definition of Sec. II, the describing function for the sliding-friction element is

$$H_0(\lambda)=f(\lambda)\angle\delta(\lambda)$$

$$\doteq\left[1-4\left(1-\frac{4}{\pi^2}\right)\lambda^2\right]^{\frac{1}{2}}\angle\tan^{-1}\frac{2\lambda[(2/\pi)^2-\lambda^2]^{\frac{1}{2}}}{1-2\lambda^2}. \tag{32}$$

Case (2).—With dead zones, $\lambda>\lambda_c$ or $\alpha>\beta$.

$$a_1=\frac{1}{\pi}\int_{\alpha}^{\pi+\beta}T_a(\sin\omega t-\lambda)\sin(\omega t)d(\omega t)$$

$$+\frac{1}{\pi}\int_{\pi+\alpha}^{2\pi+\beta}T_a(\sin\omega t+\lambda)\sin(\omega t)d(\omega t)$$

$$=(T_a/\pi)[\pi-(\alpha-\beta)-\sin\alpha(\cos\alpha+\cos\beta)$$

$$-\cos\beta(\sin\alpha+\sin\beta)]. \tag{33}$$

Similarly

$$b_1=(T_a/\pi)(\sin\alpha+\sin\beta)^2. \tag{34}$$

Hence in complete analogy with case (1) the friction-describing function is given by the expression

$$H_0(\lambda)=f(\lambda)\angle\delta(\lambda), \tag{35}$$

where

$$f(\lambda)=\frac{1}{\pi}\{[\pi-(\alpha-\beta)-\sin\alpha(\cos\alpha+\cos\beta)-\cos\beta(\sin\alpha+\sin\beta)]^2+[(\sin\alpha+\sin\beta)^2]^2\}^{\frac{1}{2}}$$

and

$$\delta(\lambda)=\tan^{-1}\frac{(\sin\alpha+\sin\beta)^2}{\pi-(\alpha-\beta)-\sin\alpha(\cos\alpha+\cos\beta)-\cos\beta(\sin\alpha+\sin\beta)}.$$

[7] Extensions to nonzero means are possible but complicate the analysis appreciably. See reference 1.

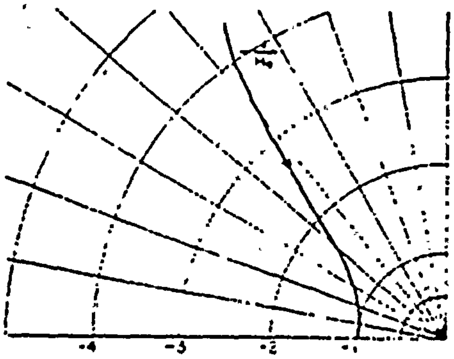

FIG. 7. The C locus.

C. C Locus

The C locus will be defined as a polar plot of the quantity $(-1/H_0)$. For any given system T_c is constant and H_0 is a function of T_a (because $T_a = T_c/\lambda$). For a fixed value of T_a, H_0 is a complex quantity which may be represented by a vector in the complex plane. The curve of $(-1/H_0)$, the C locus, forms the "critical locus" for system stability considerations, with the "critical point" $(-1, 0)$ as a special case for linear systems. The C locus is plotted in Fig. 7.

IV. SLIDING AND STATIC FRICTION IN SERVO SYSTEMS

A. Wave Form Resulting from a Sinusoidal Input Torque to a System with Sliding and Static Friction

If both sliding and static friction are considered, the friction phenomenon can be represented by the characteristic curve of Fig. 8. It will be satisfactory for most purposes to assume instantaneous transition from the static to the sliding friction value although this procedure does imply a degree of idealization.

Wave forms of $\tau_e(t)$ corresponding to an applied torque $\tau_a(t) = T_a \sin\omega t$ are easily sketched by a procedure identical with that used in Sec. III (Figs. 9 and 10). As before, there are two modes of oscillation, with and without dead zones. The latter is indistinguishable from the corresponding case discussed in Sec. III because the system is continually in motion. Thus only the former requires detailed discussion.

Mathematical Representation of the Steady-State Wave Forms

Case (1).—No dead zone, $\lambda \leq \lambda_c$, or $\alpha_2 \leq \beta$.

From Sec. III, Eqs. (10) and (11), one immediately obtains the equations for $\tau_e(t)$,

$$\tau_e(t) = T_a(\sin\omega t + \lambda) \quad \text{for} \quad -\alpha_1 < \omega t < \beta \tag{36}$$

$$\tau_e(t) = T_a(\sin\omega t - \lambda) \quad \text{for} \quad \beta < \omega t < \pi - \alpha_1, \tag{37}$$

where $\lambda = T_c/T_a$, $\alpha_1 = \sin^{-1}\lambda$ and $\beta = \cos^{-1}(\pi\lambda/2)$.

Case (2).—With dead zones,

$$\lambda > \lambda_c \quad \text{or} \quad \alpha_2 = \sin^{-1} T_s/T_a > \beta.$$

On Fig. 10,

shaded area No. 1

$$= \int_{\alpha_2}^{\pi-\alpha_1} T_a(\sin\omega t - \sin\alpha_1) d(\omega t)$$

$$= T_a[\cos\alpha_1 - (\pi - \alpha_1)\sin\alpha_1 + \cos\alpha_2 + \alpha_2 \sin\alpha_1], \tag{38}$$

shaded area No. 2

$$= \int_{\pi-\alpha_1}^{\pi+\beta} T_a(\sin\omega t - \sin\alpha_1) d(\omega t)$$

$$= T_a[\cos\beta - (\pi+\beta)\sin\alpha_1 - \cos\alpha_1 + (\pi - \alpha_1)\sin\alpha_1]. \tag{39}$$

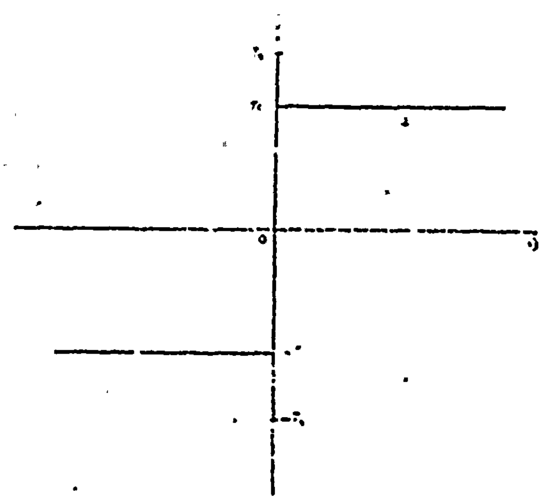

FIG. 8. Sliding and static friction characteristic.

With pure inertia load and under steady-state condition the shaded areas are equal. Equating (38) and (39) and simplifying,

$$\lambda\beta - \cos\beta = [1 - (\lambda T_s/T_c)^2]^{\frac{1}{2}} - [\pi - \sin^{-1}(\lambda T_s/T_c)]\lambda. \tag{40}$$

Equation (40) defines β in terms of known parameters. It is clear that the system will not move at all if $\lambda > T_c/T_s$, for then $T_s > T_a$.

B. Calculation of the Describing Function

Approximate expressions for effective torque are obtained by reasoning similar to that in Sec. III.

$$\tau_e(t) \doteq a_1 \sin\omega t + b_1 \cos\omega t, \tag{23}$$

where a_1 and b_1 are given by Eqs. (24) and (25), respectively.

Case (1).—No dead zone. $\lambda \leq \lambda_c$, $\alpha_2 \leq \beta$.

Since the equations for $\tau_e(t)$ are identical with those

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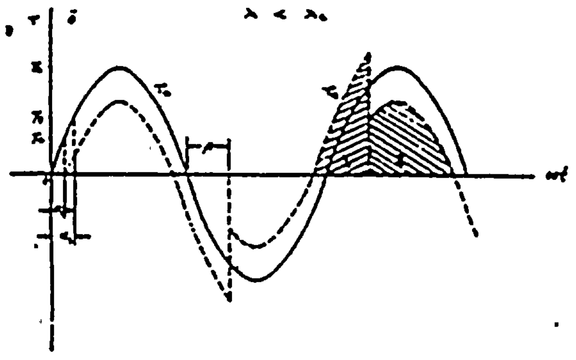

FIG. 9. Steady-state wave forms without dead zones.

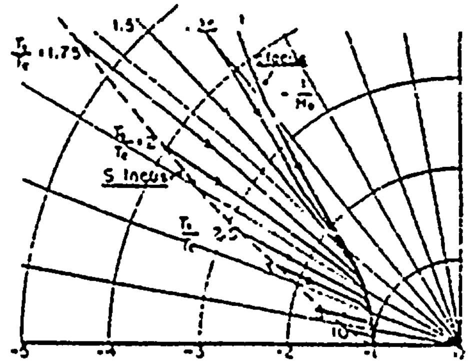

FIG. 11. Static friction loci, S locus.

in case (1) of Sec. III, the describing function $H_0(\lambda) = f(\lambda)\angle\delta(\lambda)$ is given by Eq. (32), and $f(\lambda)$ and $\delta(\lambda)$ have the following limiting values:

$$f(\lambda_c) = \frac{T_s/T_c}{[2.46 + (T_s/T_c)^2]^{\frac{1}{2}}} \tag{41}$$

$$\delta(\lambda_c) = \tan^{-1}\frac{(4/\pi)(T_s/T_c)}{(\pi/2)^2 + (T_s/T_c)^2 - 2[(\pi/2)^2 + (T_s/T_c)^2]^{\frac{1}{2}}}. \tag{42}$$

Case (2) —With dead zones, $\lambda > \lambda_c$, $\alpha_2 > \beta$.

$$a_1 = \frac{2}{\pi}\int_{\alpha_1}^{\pi+\beta} 2T_s(\sin\alpha_1 - \lambda)\sin(\omega t)d(\omega t)$$

$$= (T_s/\pi)[\pi - (\alpha_2 - \beta) + \cos\alpha_2(\sin\alpha_2 - 2\sin\alpha_1) - \cos\beta(2\sin\alpha_1 + \sin\beta)] \tag{43}$$

in like manner

$$b_1 = (T_s/\pi)[\sin\beta(2\sin\alpha_1 + \sin\beta) + \sin\alpha_2(2\sin\alpha_1 - \sin\alpha_2)]. \tag{44}$$

Hence the describing function for the static friction element is

$$H_0(\lambda) = f(\lambda)\angle\delta(\lambda),$$

where

$$f(\lambda) = \frac{1}{\pi}\{[\pi - (\alpha_2 - \beta) - \cos\alpha_2(2\sin\alpha_1 - \sin\alpha_2) - \cos\beta(2\sin\alpha_1 + \sin\beta)]^2 + [\sin\beta(2\sin\alpha_1 + \sin\beta) + \sin\alpha_2(2\sin\alpha_1 - \sin\alpha_2)]^2\}^{\frac{1}{2}} \tag{45}$$

and

$$\delta(\lambda) = \tan^{-1}\frac{\sin\beta(2\sin\alpha_1 + \sin\beta) + \sin\alpha_2(2\sin\alpha_1 - \sin\alpha_2)}{\pi - (\alpha_2 - \beta) - \cos\alpha_2(2\sin\alpha_1 - \sin\alpha_2) - \cos\beta(2\sin\alpha_1 + \sin\beta)}. \tag{46}$$

C. Static Friction-Loci and S Locus

The static friction loci are polar plots of the quantity $(-1/H_0)$ for various ratios of static friction to sliding friction. The S locus is the locus of the termini of the static friction loci. The static friction loci and the S locus are plotted in Fig. 11.

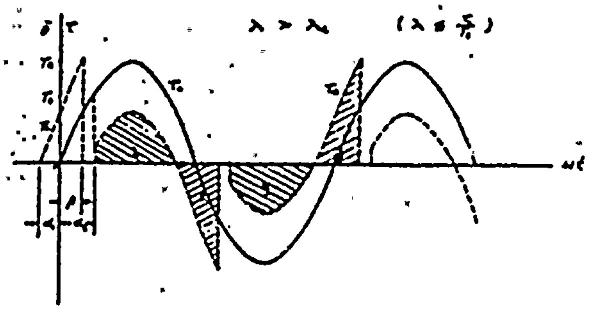

FIG. 10. Steady-state wave forms with dead zones.

V. PLOT AND INTERPRETATION OF THE AMPLITUDE AND FREQUENCY LOCI. EXAMPLES

Once the friction-describing function has been calculated, the desired stability information can be obtained easily by the method outlined in Sec. II. The procedure will be explained with the aid of a specific example which has been so chosen that a common type of corrective network will produce instability whereas another type of equalizer, apparently equivalent on a linear basis, will lead to a stable performance.

Consider the positioning loop shown in Fig. 12. In the absence of friction the loop gain function $A(s)$[4] has the form

$$A(s) = \frac{K_1}{s(1+Ts)}H_e(s). \tag{47}$$

A block diagram for this loop is shown in Fig. 13. The

[4] $A(s)$ is defined as the negative of the loop gain.

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friction describing function relates applied torque to effective torque, neither of which appears as an explicit quantity in Fig. 13. This diagram therefore does not lend itself readily to the analysis of friction. However, it is easy to construct the equivalent block diagram of Fig. 14 by rearranging the basic equations describing motor performance. Torque now appears explicitly, and it is a simple matter to write the loop gain equation in terms of the friction-describing function and the system parameters defined in Fig. 12. The equation

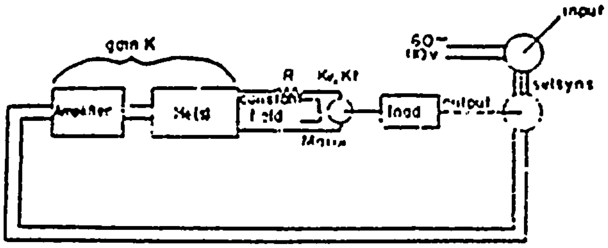

FIG. 12. Simple positioning system. $T=JR/K_vK_T$=motor time constant; $H_e(s)$=equalizer transfer function; K_T=motor torque per unit armature current; K_v=motor counter emf per unit speed; J=motor and load inertia; K_2=gain of amplifier and selsyn $=K_1K_e$; R=armature loop resistance.

has the form

$$A(s)=H_0(\lambda)\frac{K_T}{Js^2}\left[\frac{K_vs}{R}+\frac{K_2}{R}H_e(s)\right]. \qquad (48)$$

Note that viscous damping due to counter emf is represented by the additional feedback path rather than the conventional velocity-dependent load on the output member. This step is necessary because the friction-describing function was derived for the case of a pure inertia load. The procedure is obviously based on the assumption that a sinusoid applied to the nonlinear

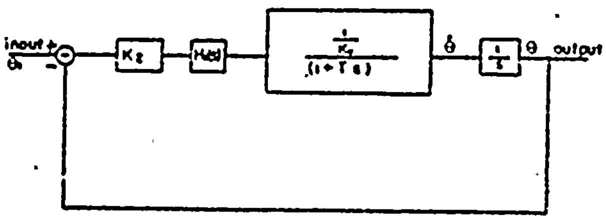

FIG. 13. Conventional block diagram.

device yields an output signal of which only the fundamental is significant in transmission around each loop, so that only sinusoids of the same frequency need be considered. This assumption becomes progressively poorer as damping, and hence the interior loop gain, increases. However, it will be seen that the method still leads to acceptable first approximations in cases of considerable practical interest.

Consider the system of Fig. 12 with the following arbitrary numerical parameters, all given in a consistent set of units.

$$K_T=K_v=0.1,\quad R=1,\quad J=0.001,\quad K_1=500,\quad K_2=50,\quad T=0.1. \qquad (49)$$

Equation (47) now becomes

$$A(s)=\frac{500}{s(1+0.1s)}H_e(s). \qquad (50)$$

In the absence of an equalizer $H_e(s)$ the loop would be close to instability. Let $H_e(s)$ be a two stage lag network or integral equalizer

$$H_e(s)=\left(\frac{1+0.5s}{1+5s}\right)^2.[9] \qquad (51)$$

Without friction the system would exhibit a phase margin of 26° and a fairly well-damped transient response. The Nyquist diagram of Fig. 15 confirms that deduction. Note, however, that the sliding friction (C) locus intersects the Nyquist plot at two points. The system is evidently stable for exceedingly small disturbances; it becomes unstable as soon as a significant input signal is applied and then sustains oscillations at an amplitude

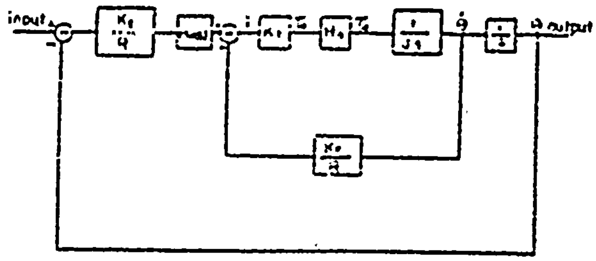

FIG. 14. Equivalent block diagram.

and frequency determined by the intersection point closer to the origin. An experimental check by analog computer resulted in oscillation at 1.6 rad/sec as compared to a theoretical value of about 2 rad/sec.

If a minor loop is used in place of the series equalizer (Fig. 16), the result is quite different. By a process entirely equivalent to that used in rearranging the block diagram of Fig. 13 the loop gain function may be written in the equivalent form

$$A(s)=H_0(\lambda)\frac{K_T}{Js}\left[\frac{K_v+H_3(s)}{R}+\frac{K_2}{R}\right]. \qquad (52)$$

Using the equalizer transfer function[10]

$$H_3(s)=\frac{(5s)^2(1+0.1s)}{10(1+0.5s)^2(1+0.003s)} \qquad (53)$$

[9] A more elaborate equalizer would probably be used in practice. This primitive form was chosen because it illustrates clearly the difficulties encountered because of friction effects.

[10] $H_3(s)$ has been chosen so as to yield a system with transfer from input to output as close as practical to the series equalized structure. Without the last denominator term (required for realizability) the two would be almost identical in the absence of friction. As it is, the step function responses are virtually indistinguishable.

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and the same numerical values as in the previous case one obtains the Nyquist diagram shown along with the series equalizer case in Fig. 15. The plot does not intersect the C locus so that the system remains stable in the presence of friction. Measured results from an analog computer bear out this conclusion.

The equalizer used in the above example introduced attenuation with the third power of frequency over a 10:1 frequency range, so that the resultant frictionless system is conditionally stable. It may therefore not appear surprising that oscillations should develop when a nonlinear element is introduced into the loop. Similar results can, however, be obtained with a single-stage lag network and a resultant loop of absolute stability. Consider the series equalizer

$$H(s) = (1+0.25s)/(1+100s). \qquad (54)$$

The gain factor K_2 is chosen as 100, all other parameters remain unchanged from the previous example. Figure 17

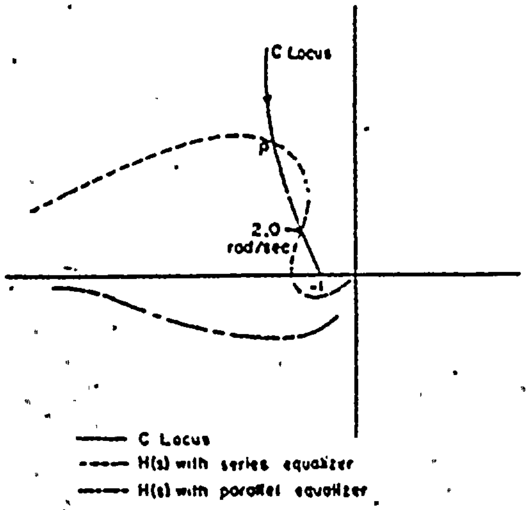

FIG. 15. Rough sketch of amplitude and frequency loci. Intersection at P is actually at a far greater distance from the origin.

shows the Nyquist plot as well as the curve for a corresponding minor loop structure. The latter is clearly stable. The Nyquist plot for the series equalizer case shows no intersection with the C locus and measurements in the presence of sliding friction alone indicate stability. However, the Nyquist diagram does cross the S locus twice, indicating that oscillations may occur when static as well as sliding friction is considered. Since the S locus traces the termini of the describing functions for various static-to-sliding-friction ratios, it is evident that stability depends critically on that ratio: only for values between those corresponding to the two intersection points can oscillations exist. It is also evident that the frequency of oscillations should rise from a theoretical minimum of 0.7 rad/sec to a maximum of 2.25 rad/sec. The corresponding range from computer measurement is 0.8 rad/sec to 2.0 rad/sec. The experimental correlation at intermediate values of the static-to-sliding-friction ratio is about equally close so that the method may be said to yield a reasonable first approximation.

VI. CONCLUSIONS

The describing function technique has been extended to cover problems of static and sliding friction in feedback systems. Under certain assumptions the stability problem can be handled adequately, and the frequency

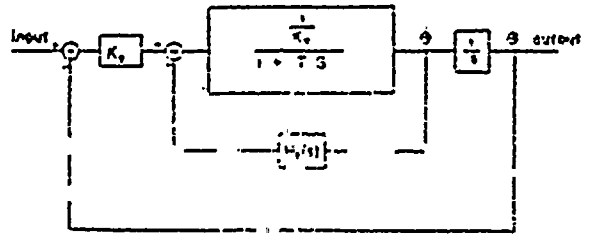

FIG. 16. Block diagram with minor loop (parallel equalizer).

of any sustained oscillations can be predicted to a first approximation. In particular it has been shown that the use of integral equalization in series with the loop may easily lead to instability when friction phenomena are important. Essentially equivalent minor loop equalizers may yield an entirely satisfactory system.

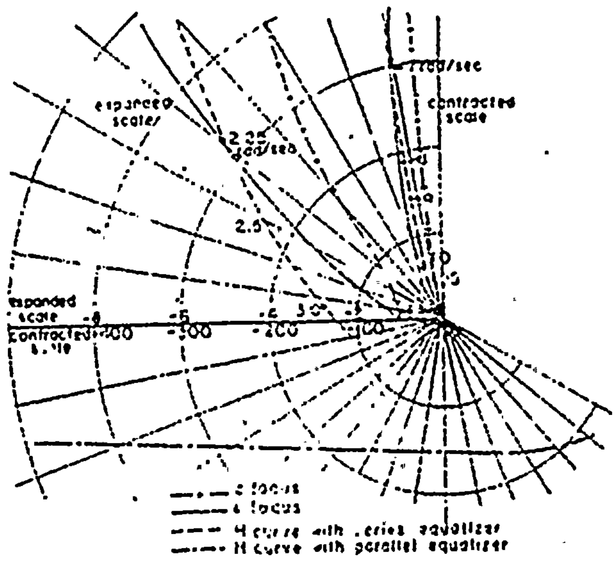

FIG. 17. Amplitude and frequency loci for an experimental system.

VII. ACKNOWLEDGMENT

The authors wish to acknowledge the aid of various members of the Electrical Engineering Department of Yale University, especially that of Professor F. B. Tuteur of the servomechanisms laboratory. The contents of this paper are taken in essence from a dissertation by Julius Tou presented to the faculty of the Yale School of Engineering in partial fulfilment of the requirements for the degree of Doctor of Engineering.

APPENDIX C

JUMP PHENOMENA

(MATHCAD SOLUTION)

CALCULATION OF HARDENING SPRING RESPONSE CURVE
Ref. Asymptotic Methods in the Theory of Non Linear Oscillations
by N.N Bogoliubov and Y.A. Mitropolsky, (U.of.Pa. Math/Physics)
Hindustan Publishing, 1961.

Theory....For assumed amplitudes, find frequencies

$a := 1.5 \qquad \delta := .2 \qquad E_1 := 1$

$:= 1 + .375 \cdot a^2 \qquad X := \left[\frac{E_1}{a}\right]^2 - \delta^2$

$N_1 := \sqrt{\omega_e^2 - \sqrt{X}} \qquad N_2 := \sqrt{\omega_e^2 + \sqrt{X}}$

$N_1 = 1.662 \qquad N_2 = 2.009$

C2

Table C-1

NONLINEAR RESONANCE CURVE

a	ν_1	ν_2
1	1.694	.954
1.1	1.732	1.101
1.05	1.711	1.033
.5	1.785	IMAGINARY
.6	1.715	IMAGINARY
.4	1.901	IMAGINARY
.3	2.097	IMAGINARY
.7	1.678	IMAGINARY
.8	1.665	.551
.75	1.669	.305
.704015078	1.677	.002
1.2	1.703	1.25
1.029	1.704	1.0
1.5	2.009	1.662
2	2.59	2.407
2.3	3.048	2.918
2.5	3.395	3.292
2.8	3.977	3.902
5.0	10.375	10.375

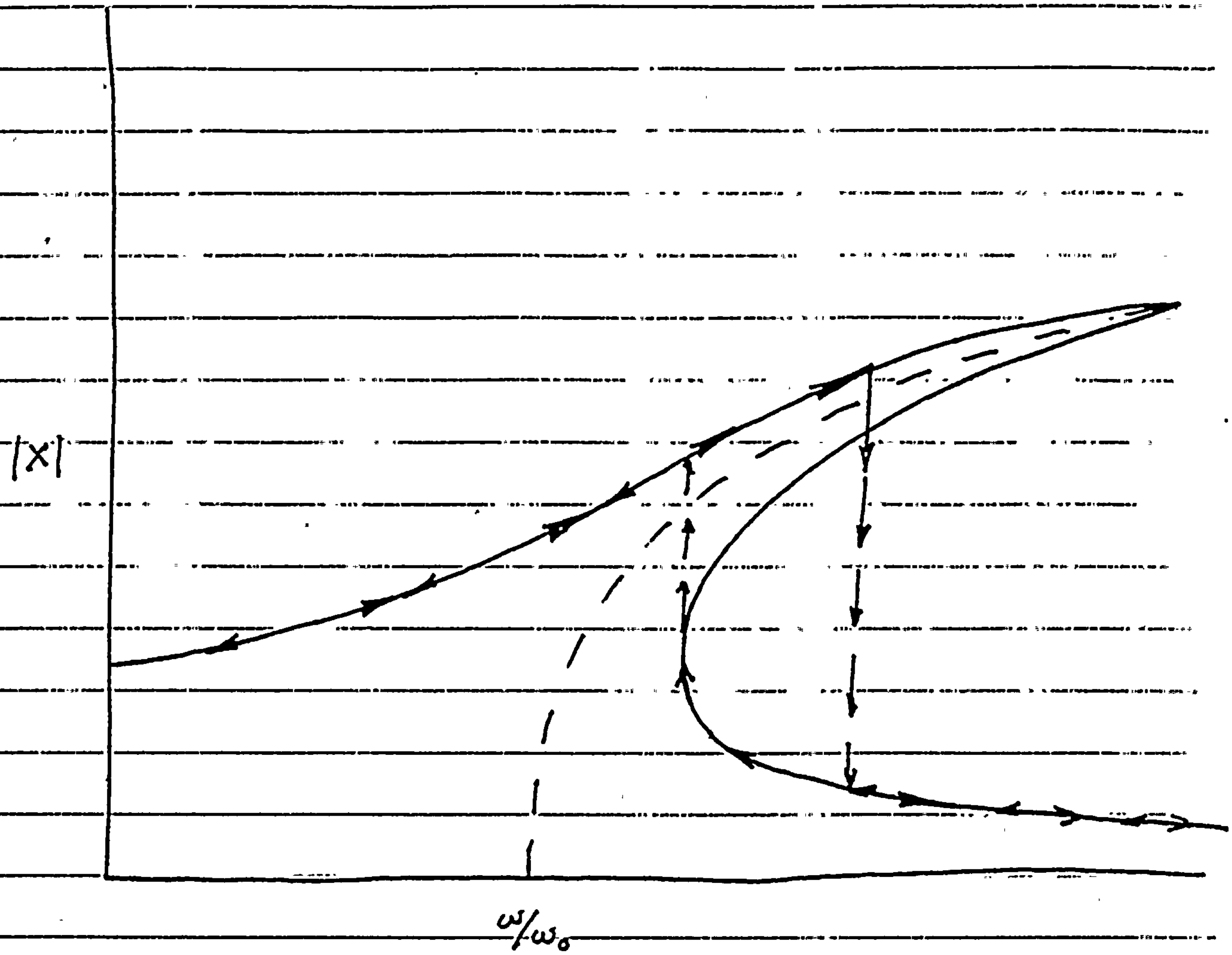

FIGURE C-1 RESONANCE CURVE SHOWING JUMP PHENOMENA

9103060045

DONALD C. COOK NUCLEAR PLANT
1990 ANNUAL OPERATING REPORT

February 28, 1991

COMPILED BY: Harry B. Brugger/WLMcKenzie for JFK
H. B. Brugger/J. F. Kurgan

REVIEWED BY: B A Svensson
B. A. Svensson
Licensing Activities Coordinator

APPROVED BY: M P Alexich
M. P. Alexich
Vice President-Nuclear Operations

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 PLANT DESCRIPTION

The Donald C. Cook Nuclear Plant is owned by Indiana Michigan Power company and is located five miles north of Bridgman, Michigan. The Plant consists of two nuclear power units, each employing a Westinghouse pressurized water reactor nuclear steam supply system. Each reactor unit employs an ice condenser reactor containment system. The American Electric Power Service Corporation was the architect-engineer and constructor.

Unit 1 and 2 reactor design power output (and licensed rating) are 3250 MWt and 3411 MWt, respectively. Unit 1 approximate gross and net electrical outputs are 1056 MWe and 1020 MWe, respectively. Unit 2 approximate gross and net electrical outputs are 1100 MWe and 1060 MWe, respectively. The main condenser cooling method is open cycle using Lake Michigan water as the cooling source for each unit.

1.2 REPORT PREPARATION

This report was compiled by H. B. Brugger/J. F. Kurgan with the following individuals contributing information as follows:

D. C. Loope	- Personnel Exposure Summary
C. A. Freer	- Steam Generator ISI Summary
B. A. Svensson	- Changes to Procedures
B. A. Svensson	- Challenges to Pressurizer PORVs and Safety Valves
B. A. Svensson	- Reactor Coolant Specific Activity
D. H. Malin	- Results of Irradiated Fuel Inspections
J. B. Kingseed	- Changes to Facility - RFCs, MMs, PMs
B. A. Svensson	- Changes to Facility - Temporary Modifications to Unit 2

2.0 PERSONNEL RADIATION EXPOSURE SUMMARY

Table 2, below, provides a summary of the radiation dose to station, utility, contractor (and other) personnel receiving exposures greater than 100 mRem in 1990. All of the doses shown in the table were measured by thermoluminescent dosimetry (TLD).

The grand total person-Rem dose shown in Table 2 is 484.537 Rem, and was determined in accordance with NRC Regulatory Guide 1.16. (The total record dose for Cook Nuclear Plant in 1990 was 579.589 Rem.)

TABLE 2
PERSONNEL RADIATION EXPOSURE SUMMARY

REG. GUIDE 1.16

	# of Personnel >100 mR			Total Person-Rem		
	Stat.	Util.	Cont.	Station	Utility	Contract
Rx Operations and Surveillance						
Maintenance Personnel	9	1	56	1.169	0.098	9.833
Operations Personnel	67	1	14	13.600	0.401	3.468
Health Physics Personnel	22	0	93	5.182	0.000	36.847
Supervisory Personnel	1	0	0	0.167	0.000	0.000
Engineering Personnel	7	0	1	2.068	0.000	0.086
Routine Maintenance						
Maintenance Personnel	83	1	362	24.680	0.346	142.976
Operations Personnel	10	2	20	2.380	0.596	7.820
Health Physics Personnel	9	0	34	1.405	0.000	8.427
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	5	0	2	0.936	0.000	0.308
In-Service Inspection						
Maintenance Personnel	9	0	146	1.164	0.000	51.425
Operations Personnel	3	1	12	0.544	0.117	5.440
Health Physics Personnel	1	0	8	0.226	0.000	2.253
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	0	0	0	0.000	0.000	0.000
Special Maintenance						
Mainenance Personnel	8	0	307	1.418	0.000	90.081
Operations Personnel	2	0	12	0.728	0.000	2.313
Health Physics Personnel	2	0	4	0.242	0.000	0.453
Supervisory Personnel	0	0	1	0.000	0.000	0.097
Engineering Personnel	4	4	9	0.604	0.635	1.545

TABLE 2 [illegible]'d)

	# of Personnel >100 mR			Total Person-Rem		
	Stat.	Util.	Cont.	Station	Utility	Contract
Waste Processing						
Maintenance Personnel	0	0	73	0.000	0.000	20.635
Operations Personnel	1	0	10	0.136	0.000	3.059
Health Physics Personnel	4	0	16	0.427	0.000	3.856
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	1	0	0	0.194	0.000	0.000
Refueling						
Maintenance Personnel	7	0	50	0.852	0.000	9.668
Operations Personnel	7	0	57	2.687	0.000	14.881
Health Physics Personnel	2	0	23	0.212	0.000	5.269
Supervisory Personnel	1	0	0	0.098	0.000	0.000
Engineering Personnel	1	0	2	0.223	0.000	0.261
TOTALS						
Maintenance Personnel	93	2	700	29.283	0.443	324.619
Operations Personnel	81	3	105	20.077	1.115	36.981
Health Physics Personnel	25	0	126	7.693	0.000	57.106
Supervisory Personnel	2	0	1	0.265	0.000	0.097
Engineering Personnel	17	4	13	4.026	0.634	2.199
GRAND TOTALS						
	218	9	945	61.343	2.192	421.001

3.0 STEAM GENERATOR IN-SERVICE INSPECTION

3.1 UNIT 1 INSPECTION SUMMARY

One hundred percent of the tubes in all four steam generators of Unit 1 were bobbin coil eddy current inspected. All tubes were tested utilizing a 0.720 inch diameter probe with the exceptions of rows 1 through 4 U-bends that would not allow a 0.720 inch probe to pass. A 0.700 or 0.680 inch probe was used for the U-bend inspections only.

The bobbin coil test frequencies used were as follows:

400 kHz as the prime test frequency.
200 kHz and the 100 kHz as supplemental frequencies.
10 kHz as the locator frequency

A total of 70 tubes were removed from service following this inspection due to the eddy current results. Appendix A provides inspection details concerning the indications found in each tube and whether or not the tube was plugged.

3.2 UNIT 2 INSPECTION SUMMARY

Six and one-half percent of the tubes in Unit 2 steam generators were inspected using the same method and test frequencies described for Unit 1 inspections.

All tests were conducted with a 0.720 inch diameter probe with the exception of three tubes in steam generator 23, located in Rows 1 and 2. These three tubes were tested with a 0.700 inch probe.

No degradation or pluggable indications were found.

TABLE 1

STEAM GENERATOR INSPECTIONS

UNIT 1	UNIT 2
Steam Generator 11	Steam Generator 21
All tubes tested from the inlet (HL) 624 full length (TE-CL TO TE-HL) 2,584 U-bend (7TSP-CL to TE-HL)	None
Steam Generator 12	Steam Generator 22
All tubes tested from the inlet (HL) 615 full length (TE-CL to TE-HL) 2,625 U-bend (7TSP-CL to TE-HL)	All tubes tested from the inlet (HL) 67 full length (TE-CL to TE-HL) 169 U-bend (7TSP-CL to TE-HL)
Steam Generator 13	Steam Generator 23
All tubes tested from the inlet (HL) 634 full length (TE-CL to TE-HL) 2,612 U-bend (7TSP-CL to TE-HL)	All tubes tested from the inlet (HL) 68 full length (TE-Cl to TE-HL) 167 U-bend (7TSP-CL to TE-HL)
Steam Generator 14	Steam Generator 24
All tubes tested from the inlet (HL) 619 full length (TE-CL to TE-HL) 2,602 U-bend (7TSP-CL to TE-HL)	None

4.0 CHANGES TO PROCEDURES

This section contains a brief description of the procedure changes implemented under the provisions of 10 CFR50.59 and the associated safety evaluations.

4.1 MAINTENANCE PROCEDURES

4.1.1 Procedure NO. 12 MHP 502.010.001, Revision 2

Description of Change:

During preparations for the complete replacement of the ice condenser containment divider barrier seal, it was discovered that the actual seal installation did not agree with the description in the subject maintenance procedure entitled "Removal and Replacement of Sections of the Ice Condenser Divider Barrier Seal". Further engineering review determined that the divider barrier seal was properly installed per the plant design drawings. It was also discovered that the description of the seal installation details in the UFSAR was the same as the maintenance procedure but different than the actual design. Specifically, page 5.2.-80 of the UFSAR states that the spacing between bolts holding the seal is three inches, and that the seal is laid with one-half inch extra length between bolt holes. Plant design drawings show the studs being three to five inches apart in various locations, and the seal having between zero and one-quarter inch extra length between bolt holes.

The procedure was revised to require installation of the seal as shown on the design drawings.

Safety Evaluation Summary:

The review of this issue indicated that during a loss of coolant accident, steam from the break is directed through the ice condenser and condensed, thus preventing containment pressure from exceeding design values. To ensure the maximum amount of steam is condensed, the flow area from lower to upper containment which bypasses the ice condenser must be limited.

The ice condenser is supported from the crane wall inside containment. During a seismic event, the relative movement between the crane wall and the containment wall is 1.3 inches. To accommodate this movement, a gap was left between the crane wall and the containment wall. This gap is sealed with a barrier designed to withstand the differential pressure between lower and upper compartments during postulated loss of coolant accidents and seismic events.

The statements on page 5.2-80 of the UFSAR regarding hole spacing and length imply that the extra one-half inch length is necessary to prevent failure of the seal due to elongation during a seismic event. A review by our design engineers determined that there is sufficient resiliency in the material to prevent failure during the maximum elongation provided the seal is installed per the design drawings.

The safety evaluation concluded that changing the divider barrier seal installation procedure to require installation of the seal per the design drawings does not involve an unreviewed safety question as defined in 10 CFR50.59.

The UFSAR will be revised to reflect the actual seal installation in the next annual update.

4.1.2 Procedure Numbers: 12 MHP 5021.001.016
12 MHP 5021.005.003

Description of Change:

These maintenance procedures were implemented to permit the use of "Chesterton" valve packing on safety related valves in systems described in the UFSAR. The Chesterton valve packing differs from the valve packing configurations described in the UFSAR for certain safety related valves.

Safety Evaluation Summary:

A technical evaluation determined that the packing system described in the UFSAR has been rendered obsolete by advances in technology. The Chesterton system of packing currently being installed in the majority of the safety related valves is a direct development of an EPRI study and is technically superior to that described in the UFSAR.

The safety evaluation determined that the changes do not involve and unreviewed safety question as defined in 10 CFR50.59. Changes to the UFSAR will be incorporated in the next annual update.

4.2 OPERATING PROCEDURES

4.2.1 Procedure Numbers: 1 - and 2 - OHP 4021.008.002
1 - and 2 - OHP 4020.STP.053A
1 - and 2 - OHP 4030.STP.053B
1 - and 2 - OHP 4030.STP.055

Description of Changes:

The above procedures were changed to require the RHR pump discharge cross-tie valves to be closed during normal plant operations. The changes were implemented as corrective action in response to NRC Bulletin 88-04, "Potential Safety Related Pump Loss". Closing the RHR pump discharge cross-tie valves makes the miniflow circuits for each RHR pump independent, thereby removing the potential for dead heading the weaker pump. When the RHR cross-tie valves are closed the safety injection pump discharge cross-tie valves must remain open.

Safety Evaluation Summary:

The changes were implemented after receiving NRC approval via a "Change to the Licensing Basis for SI and RHR Cross-ties (TACS Nos. 64962 and 64963)", dated January 30, 1989.

4.3 CHEMISTRY PROCEDURES

4.3.1 Procedure Number: 1 - and 2 - THP 6020.LAB.200

Description of Change:

The above procedures were implemented for the purpose of chlorinating the Unit 1 and Unit 2 main turbine condenser circulating water systems. The UFSAR, Section 10.6.2, describes chlorination of circulating water using chlorine gas. These procedures will accomplish the clorination using a sodium hypochlorite solution rather than chlorine gas.

Safety Evaluation Summary:

The use of sodium hypochlorite results in the same chemical compound in the water as would be the case if chlorine gas were used. The use of sodium hypochlorite is preferred because it is in a liquid form at room temperature and, therefore, does not present a toxic gas concern.

The safety evaluation concluded that the change does not involve an unreviewed safety question as defined in 10 CFR50.59. Changes to the UFSAR will be incorporated in the next annual update.

4.4 ENVIRONMENTAL MONITORING PROCEDURES

Procedure Number: 12 THP 6010 ENV.051, Revision 3

Description of Change:

Revision 3 to the above procedure included changes to the sample location map to reflect the updated sample station locations for all sample media.

Safety Evaluation Summary:

The changes have no effect on the environmental sampling requirements of Technical Specification 3.12. The safety evaluation concluded that the change does not represent an unreviewed safety question. The UFSAR will be revised to reflect the additional sample station location in the next annual update.

5.0 CHALLENGES TO PRESSURIZER POWER OPERATED RELIEF VALVES AND SAFETY VALVES

There were no challenges on either Unit 1 or Unit 2 to the pressurizer power operated relief valves (PORV's) or the pressurizer safety valves as a result of the valves being called upon to mitigate an actual overpressure condition.

One event occurred on Unit 2, on October 6, 1990, the inadvertent opening of pressurizer PORV, 2-NRV-153. The event occurred while performing a surveillance test on the residual heat removal pumps suction valves isolation actuation bistables. When the event occurred, the reactor coolant system (RCS) was in Mode 4. RCS average temperature was approximately 250°F and the RCS pressure was approximately 350 psig. The cold overpressure protection system was not required to be operable since the RCS temperature was greater than 152°F. However, the cold overpressure protection block switch was in the "unblocked" position to provide added protection for the RHR system.

During the test, when the bistable was made up to verify the automatic isolation of the RHR system from the RCS, a signal was also directed to the actuation bistable for PORV, 2-NRV-153, causing it to open.

To prevent recurrence, the surveillance test procedure has been revised to require the "cold overpressure block" switch to be placed in the position to block the RCS pressure signal prior to performing this test.

6.0 REACTOR COOLANT SPECIFIC ACTIVITY

There were no instances on either Unit 1 or Unit 2 in which the reactor coolant I-131 specific activity exceeded the limits of Technical Specification 3.4.8.

7.0 IRRADIATED FUEL EXAMINATIONS

During 1990 two separate examinations were performed on the irradiated fuel discharge from Unit 2, Cycle 7. These examinations were conducted in parallel with, or shortly after, the core was unloaded. The intent was to determine fuel pin failures and gross structural defects in the fuel assemblies. Also, during 1990 an examination was performed on the irradiated fuel discharged from Unit 1 Cycle 11. This examination was conducted in the same manner and for the same purpose as the examinations of Unit 2 discharged fuel.

7.1 VISUAL EXAMINATIONS

The first examination of Unit 2 fuel, (and the only examination of Unit 1 fuel) was by routine binocular inspections of the fuel assemblies per procedure numbers: **12 THP 6040 PER.353 and 12 SHP 4050 QC.002. As each assembly is downloaded to the spent fuel pool, it is visually examined on all four sides. The examiner is looking specifically for torn or missing gridstraps, missing or damaged fuel pins, excessive clad hydriding, or rod bow to gap closure. This inspection is primarily intended to detect fuel damage caused by mechanical interaction between assemblies or baffle jetting, and is done during each refueling. There was no indication of any fuel damage.

7.2 ULTRASONIC EXAMINATIONS

Because the Unit 2 steam generators were replaced, and because all previous plants that replaced steam generators experienced subsequent cycle fuel failures, a contract was let to Advanced Nuclear Fuels (ANF) to provide ultrasonic (UT) examination of the 116 irradiated assemblies out of 193 assemblies making up the Unit 2, Cycle 8 core. The ultrasonic system works by a probe transceiver sending a high frequency sound wave into a fuel pin and measuring the strength of the returning signal, or "ring back". A fuel pin can be determined to have water in it by monitoring the relative strength of this ring back. In this way, not only can an assembly be determined to have leaking pins, but the numbers and locations of the bad pins can be identified. None of the 116 fuel assemblies were found to contain leaking pins.

In addition, a 117th Unit 2 assembly, T24, was also examined. It was known to have leaking fuel pins after Cycle 6 but was reused in Cycle 7. The examination provided a check of the testing methodology. Fuel assembly T24 was again determined to be leaking.

8.0 CHANGES TO FACILITY

Brief descriptions and summary safety evaluations for design changes made to the facility as described in the UFSAR are presented in this section. These changes were completed pursuant to the provisions of 10 CFR50.59

8.1 COLD TO HOT LABORATORY CONVERSION

Description of Change:

RFC 01-1959 involved the conversion of the cold laboratory to provide additional hot laboratory space. The change did not affect systems required for safe plant shutdown nor did it affect systems whose failure could impair safety related systems. The change increased the amount of space available for performing analyses of radioactive materials and will reduce congestion in the hot lab.

Safety Evaluation Summary:

The changes made in this RFC did not affect safety related systems or components; however, changes were made to systems which handle radioactive materials.

This change was reviewed and it was concluded that it did not constitute an unreviewed safety question as defined in 10 CFR 50.59.

8.2 NUCLEAR INSTRUMENTATION SYSTEM UPGRADE

Description of Change:

The purpose of this subtask (RFC 12-2900, Subtask B.01, Unit 2 only) was to upgrade the nuclear instrumentation system (NIS) to meet the guidance provided by R.G. 1.97, Revision 3.

The neutron flux instrumentation was installed so that the performance of the reactor protection system (RPS) is not degraded. The scope of work for this subtask included providing two Gamma-Metrics environmentally and seismically qualified detectors inside containment, installing a pre-amplifier just outside containment, modifying the NIS cabinets, and running cable for signals and power supplies.

Safety Evaluation Summary:

This subtask has been classified as safety related. The instrumentation installed by this subtask will not provide any reactor trip signals to the reactor protection system (RPS). However, the neutron flux indicators will be mounted in Channels I and III of the safety related NIS cabinet. Furthermore, R.G. 1.97, Revision 3, classifies neutron flux as a category

1 variable whose purpose is to verify accident mitigation. Category 1 classification is intended for key variables important to safety. Consequently, the installation requirements for Category 1 variables include full equipment qualification, class 1E power supplies, redundancy, and a quality assurance program.

This change was reviewed and it was concluded that it did not involve a change in the T/S, did not constitute an unreviewed safety question as defined in 10 CFR50.59, "Changes, Tests and Experiments," and did not create a substantial hazard to the health and safety of the public.

8.3 CONTAINMENT ATMOSPHERE TEMPERATURE DETECTORS

Description of Change:

The purpose of RFC 12-1900 D.22, (Unit 2 only) was to upgrade six of the seventeen containment atmosphere temperature RTDs to meet R.G. 1.97, Rev. 3, requirements. Commitment for this change was made in AEP:NRC:07790, "June 12, 1984 Confirmatory Order - Final Status Report on Regulatory Guide 1.97 Compliance," dated October 15, 1985. Specifically, we agreed to replace six containment atmosphere temperature RTDs (ETR-12, 14, 18, 20, 21, and 22) with environmentally qualified equipment having a temperature range of 0 to 400°F. In addition, this RFC expanded the range of all seventeen containment atmosphere temperature RTDs by replacing them with wider range instruments.

Safety Evaluation Summary:

This RFC was classified as safety-interface in accordance with R.G. 1.97, Rev. 3. Containment atmosphere temperature serves no function associated with reactor trip or ECCS actuation, and failure of the RTDs will not result in an increase in the severity of an accident or prevent proper functioning of safety systems.

This change was reviewed and it was concluded that it did not constitute an unreviewed safety question as defined in 10 CFR50.59, "Changes, Tests and Experiments," and did not create a substantial hazard to the health and safety of the public.

8.4 PLANT RADIO COMMUNICATIONS UPGRADE

Description of Change:

RFC 12-2950 upgraded the Cook Nuclear Plant communication system radio to facilitate required operator actions during normal and emergency operations. The new radio incorporates two existing security emergency and fire protection radio frequencies as well as two recently obtained frequencies.

Safety Evaluation Summary:

This change upgraded existing equipment and it does not adversely impact any safety related equipment. The change provides a more reliable communications radio system to assist in normal operation and it provides a more efficient means of coordinating the safe shutdown of the reactor.

This change was reviewed and it was concluded that it did not constitute an unreviewed safety question as defined in 10 CFR50.59, "Changes, Tests and Experiments," and that the change did not create a substantial hazard to the health and safety of the public.

8.5 FUEL TRANSFER SYSTEM UPGRADE

Description of Change:

RFC 12-3024 replaced the existing air motor-driven fuel transfer system with a winch cable-driven fuel transfer system. The previous system was an underwater air motor-driven conveyor car that ran on tracks extending from the refueling canal through the transfer tube and into the transfer canal. Critical components were underwater, thus making repairs and maintenance difficult. The cable-driven system eliminates underwater operation for critical components making the system more reliable.

Safety Evaluation Summary:

RFC 12-3024 did not alter the function of the fuel transfer system. It changed the drive mechanism and moved major components to a more accessible area, making repairs and maintenance easier and quicker. No safety-related actions are associated with the fuel transfer mechanism.

This change was reviewed and it was concluded that it did not constitute an unreviewed safety question as defined in 10 CFR50.59, "Changes, Tests and Experiments," and that the change did not create a substantial hazard to the health and safety of the public.

8.6 ADDITIONAL AUXILIARY BUILDING RADIATION MONITORS

Description of Change:

RFC 12-4036 involved the installation of nine area radiation monitors. The monitors, which are located in the auxiliary building, measure radiation from the refueling water purification filter, north and south seal water injection filters, seal water filters and reactor coolant filters. The monitors were installed to measure dose rates in the filter cubicles, thereby preventing overexposure to workers changing the filters.

Safety Evaluation Summary:

The changes made via this RFC do not impact safety related equipment. The changes were made to aid in minimizing the exposure of personnel working in the areas where the monitors are installed. No accident analyses are impacted by the addition of these radiation monitors.

This change was reviewed and it was concluded that it did not constitute an unreviewed safety question as defined in 10 CFR50.59, "Changes, Test and Experiments," and that the change did not create a substantial hazard to the health and safety of the public.

8.7 MONITORING REDUCED RCS INVENTORY CONDITIONS

Description of Change:

The purpose of RFC 12-4058 is to comply with commitments to the NRC regarding Generic Letter 88-17. This RFC added a new, independent system for monitoring half loop and reduced inventory in the RCS, as well as other indications to enhance the ability of the control room operator to monitor and maintain RCS water level during reduced inventory conditions. Two instrument carts were set up in the control room to monitor wide and narrow range levels, incore temperatures and RHR flows during half-loop or reduced RCS inventory conditions.

Safety Evaluation Summary:

This RFC added instrumentation to measure RCS level as well as other parameters of importance during half-loop.

This change was reviewed and it was concluded that it did not constitute an unreviewed safety question as defined in 10 CFR50.59 and that it did not create a substantial hazard to the health and safety of the public.

8.8 CONTROL ROOM VENTILATION MODIFICATION

Description of Change:

Minor Modification 12-094 adjusted the control room ventilation system such that flow through the normal intake ductwork from the outside atmosphere is limited to 150 cfm. The adjustment was accomplished by modulating the existing manual balancing damper HV-ACRDV-1. (No hardware changes were required.) The modification was performed on both units.

The change was necessary to fulfill a commitment made to the NRC to satisfy concerns regarding the analyses found in Chapter 14.3.5 of the present Unit 1 UFSAR. The revised analyses assumed that flow through the normal intake damper was limited to 200 cfm for cases that assumed failure

of the normal intake isolation damper (HV-ACRDA-1) to isolate when required. The flow was limited to 150 cfm to provide margin to the 200 cfm analysis limit.

Safety Evaluation Summary:

The change was reviewed and it was concluded that it did not constitute an unreviewed safety question as defined in 10 CFR50.59, "Changes, Tests and Experiments," and did not create a substantial hazard to the health and safety of the public.

8.9 CONTAINMENT AIRLOCK DOOR SEAL MODIFICATION

Description of Change:

MM 12-098 involved the replacement of the packing gland from the interlock and handwheel hubs on the containment upper and lower airlock doors. A modified packing gland that used EPDM O-ring seals was installed in lieu of grafoil packing. The modification was made to avoid past difficulties in adjusting the handwheel packing gland cover to adequately compress the packing around the handwheel shaft while retaining rotational freedom of the shaft.

The containment airlocks are Seismic Class I structures. Further, the modification to the airlock doors had the potential to adversely increase the consequences of a loss-of-coolant accident. As a result, the seal modification was considered a safety related design change.

Safety Evaluation Summary:

This change was reviewed and it was concluded that it did not represent an unreviewed safety question. This conclusion was based, in part, on the performance of a post-installation test of the airlock doors to verify their functionality.

8.10 CONTROL AIR DRYER FILTER MODIFICATION

Description of Change:

PM 12-805 consisted of replacing the control air dryer after-filters with a filter incorporating a pressure guage which measures differential pressure across the filter. The change required piping modifications to the control air system, a non-safety related system, but did not alter the system's function nor did the change impact any safety related system or equipment.

Safety Evaluation Summary:

The change was made to a non-safety related system. The filters are located in the turbine building, and there is no safety related equipment in the vicinity. The change did not alter the function of the system and the inclusion of the differential pressure guage is considered to be a system enhancement.

This change was reviewed and it was concluded that it did not constitute an unreviewed safety question as defined in 10 CFR50.59, "Changes, Tests and Experiments," and that the change did not create a substantial hazard to the health and safety of the public.

8.11 CHLORINE INJECTION METHOD MODIFICATION

Description of Change:

The purpose of modification PM 12-810 was to change the method of injecting chlorine into the essential service water system. The change consisted of modifying non-safety-related piping to allow the use of liquid sodium hypochlorite rather than chlorine gas.

Safety Evaluation

Chlorination of the essential service water system is not a safety related function, and the change in the chemicals used to perform the chlorination does not adversely impact the capability of the essential service water system.

The physical changes required were made to non-safety grade piping which does not interface with any safety related systems.

This change was reviewed and it was concluded that it did not constitute an unreviewed safety question as defined in 10 CFR 50.59, "Changes, Tests and Experiments," and that the change did not create a substantial hazard to the health and safety of the public.

8.12 TEMPORARY JUMPERING OF BATTERY CELLS

Description of Change:

The UFSAR describes the plant 250 volt DC battery systems as to the number of cells in each battery. The "Train AB" and Train CD" batteries consist of 116 cells connected in series and the "Train N" battery system (turbine driven auxiliary feed pump DC power supply) consists of 117 cells. Temporary Modification No. 164 provided for jumpering three cells in Unit 2 Train N battery and Temporary Modification No. 171 provided for the jumpering of two cells in Unit 2 Train AB battery. In each case the

batteries had been declared inoperable per Technical Specification 4.8.2.3.2.6.1 due to the individual cell voltages having dropped more than 0.05 volts from the value observed during the original acceptance test.

Safety Evaluation Summary:

A technical evaluation and a safety evaluation was performed and it was concluded that jumpering of the cells did not significantly degrade the batteries capacity to perform their intended safety function and that the changes did not involve an unreviewed safety question as defined in 10 CFR50.59.

APPENDIX A

Steam Generator Inspection Details

APPENDIX A

STEAM GENERATOR NUMBER 11

ROW	COL	INDIC	LOCATION	PLUGGED	ROW	COL	INDIC	LOCATION	PLUGGED
5	3	DI	2SPH + 0 IN	NO	8	11	DI	TEH + 20.6 IN	YES
8	3	37%	2SPC + 0 IN	NO	18	11	DI	TEH + 20.4 IN	YES
10	3	47%	1SPC + 0 IN	YES	21	11	DI	1SPH + 0 IN	NO
13	3	15%	1SPC + 0 IN	NO	26	11	DI	TEH + 20.4 IN	YES
15	3	22%	TSC + 14.2 IN	NO	29	11	DI	TEH + 20.7 IN	YES
5	4	DI	TEH + 19.9 IN	YES	24	12	DI	1SPH + 0 IN	NO
7	4	14%	1SPC + 0 IN	NO	5	14	DI	3SPH + 0 IN	NO
9	4	DI	2SPH + 0 IN	NO	31	14	DI	TEH + 20.4 IN	YES
16	4	DI	1SPH + 0 IN	NO	10	15	DI	TEH + 20.4 IN	YES
15	5	32%	TEH + 20.3 IN	YES	1	16	35%	2SPH + 0 IN	NO
17	5	18%	2SPC + 0 IN	NO	18	16	DI	TEH + 20.9 IN	YES
19	5	DI	TEH + 20.6 IN	YES	23	16	DI	TEH + 20.8 IN	YES
18	6	14%	1SPC + 0 IN	NO	9	17	DI	TEH + 21.0 IN	YES
4	7	DI	TEH + 20.3 IN	YES	25	17	57%	TEH + 20.7 IN	YES
17	7	29%	1SPC + 0 IN	NO	32	17	11%	2SPC + 0 IN	NO
18	8	DI	1SPH + 0 IN	NO	35	17	DI	TEH + 20.0 IN	YES
20	8	DI	1SPH + 0 IN	NO	17	18	DI	TEH + 20.7 IN	YES
23	8	34%	1SPC + 0 IN	NO	31	18	DI	1SPH + 0 IN	NO
17	9	DI	TEH + 20.5 IN	YES	35	18	33%	2SPC + 0 IN	NO
22	9	DI	TEH + 20.5 IN	YES	36	18	28%	2SPC + 0 IN	NO
16	10	DI	TEH + 20.5 IN	YES	36	20	47%	2SPC + 0 IN	YES
19	10	DI	TEH + 20.2 IN	YES	37	20	18%	1SPC + 0 IN	NO
22	10	DI	TEH + 20.5 IN	YES	18	21	DI	TEH + 21.2 IN	YES
25	10	DI	TEH + 20.8 IN	YES	20	21	DI	2SPH + 0 IN	NO
3	11	DI	TEH + 20.6 IN	YES	37	21	31%	2SPC + 0 IN	NO

STEAM GENERATOR NUMBER 11

ROW	COL	INDIC	LOCATION	PLUGGED
38	21	38%	2SPC + 0 IN	NO
3	22	DI	2SPH + 0 IN	NO
38	22	26%	2SPC + 0 IN	NO
40	25	38%	2SPC + 0 IN	NO
2	27	DI	2SPH + 0 IN	NO
30	27	DI	TEH + 20.3 IN	YES
33	27	96%	TEH + 20.3 IN	YES
36	29	DI	TEH + 20.4 IN	YES
10	30	DI	1SPH + 0 IN	NO
12	30	DI	1SPH + 0 IN	NO
27	30	DI	TEH + 20.1 IN	YES
36	30	DI	TEH + 20.2 IN	YES
31	31	DI	TEH + 20.2 IN	YES
39	31	DI	TEH + 20.5 IN	YES
42	31	27%	2SPC + 0 IN	NO
43	31	37%	1SPC + 0 IN	NO
16	32	DI	TEH + 20.7 IN	YES
2	35	DI	1SPH + 0 IN	NO
42	35	DI	TEH + 20.6 IN	YES
42	36	DI	TEH + 20.8 IN	YES
7	37	DRI	TEH + 2.3 IN	YES
41	37	DI	TEH + 20.2 IN	YES
43	37	25%	2SPC + 0 IN	NO
45	37	30%	2SPC + 0 IN	NO
31	38	DI	2SPH + 0 IN	NO
36	39	DI	TEH + 20.6 IN	YES
39	39	DI	TEH + 20.3 IN	YES
33	40	DI	1SPH + 0 IN	NO
40	40	DI	2SPH + 0 IN	NO
44	40	33%	1SPC + 0 IN	NO
43	41	DI	TEH + 20.5 IN	YES
34	42	DI	TEH + 20.5 IN	YES
22	43	DI	TEH + 20.5 IN	YES
38	43	DI	TEH + 20.7 IN	YES
39	43	DI	TEH + 20.5 IN	YES
42	43	DI	3SPH + 0 IN	NO
22	44	DI	2SPH + 0 IN	NO
14	45	DI	TEH + 15.6 IN	YES
33	47	DI	1SPH + 0 IN	NO
5	48	DI	2SPH + 0 IN	NO
41	48	25%	AV4 + 0 IN	NO
12	49	95%	TEH + 14.9 IN	YES
31	49	DI	1SPH + 0 IN	NO
46	49	DI	1SPH + 0 IN	NO
33	51	DI	1SPH + 0 IN	NO
27	52	DI	1SPH + 0 IN	NO
45	52	32%	2SPC + 0 IN	NO
18	53	DI	TSH + .8 IN	YES
16	54	DI	TSH + 1.8 IN	YES
25	54	DI	1SPH + 0 IN	NO
25	54	DI	2SPH + 0 IN	NO

STEAM GENERATOR NUMBER 11

ROW	COL	INDIC	LOCATION	PLUGGED
10	55	DI	TEH + 20.9 IN	YES
13	55	DI	TEH + 13.2 IN	YES
21	55	DI	1SPH + 0 IN	NO
45	55	DI	1SPC + 0 IN	NO
33	56	DI	1SPH + 0 IN	NO
43	56	DI	1SPH + 0 IN	NO
43	56	32%	2SPC + 0 IN	NO
45	56	26%	3SPC + 0 IN	NO
45	56	37%	2SPC + 0 IN	NO
45	57	34%	3SPC + 0 IN	NO
1	58	DI	2SPH + 0 IN	NO
43	58	37%	2SPC + 0 IN	NO
44	58	25%	1SPC + 0 IN	NO
4	59	DI	TEH + 20.6 IN	YES
5	59	DI	1SPH + 0 IN	NO
44	59	29%	2SPC + 0 IN	NO
45	59	33%	2SPC + 0 IN	NO
5	60	DI	1SPH + 0 IN	NO
42	60	DI	1SPH + 0 IN	NO
29	61	DI	1SPH + 0 IN	NO
39	61	DI	2SPH + 0 IN	NO
18	62	DI	3SPH + 0 IN	NO
39	62	DI	TEH + 20.3 IN	YES
4	63	DI	1SPH + 0 IN	NO
4	63	27%	2SPH + 0 IN	NO
23	63	23%	AV2 + 0 IN	NO
23	63	21%	AV3 + 0 IN	NO
36	63	25%	AV3 + 0 IN	NO
36	63	18%	AV4 + 0 IN	NO
40	63	DI	TEH + 20.8 IN	YES
15	64	DI	2SPH + 0 IN	NO
2	65	DI	1SPH + 0 IN	NO
13	65	DRI	TEH + 2.7 IN	YES
26	65	DI	1SPH + 0 IN	NO
43	65	35%	1SPC + 0 IN	NO
25	66	DI	1SPH + 0 IN	NO
26	66	DI	1SPH + 0 IN	NO
27	66	DI	1SPH + 0 IN	NO
36	66	DI	TEH + 20.8 IN	YES
33	67	DI	1SPH + 0 IN	NO
36	68	DI	2SPH + 0 IN	NO
40	70	DI	TEH + 20.7 IN	YES
1	71	DI	1SPH + 0 IN	NO
8	71	DI	1SPH + 0 IN	NO
27	71	DI	1SPH + 0 IN	NO
1	72	DI	1SPH + 0 IN	NO
1	72	DI	2SPH + 0 IN	NO
25	73	DI	TEH + 20.7 IN	YES
1	74	DRI	TEH + 2.3 IN	YES
11	74	DI	1SPH + 0 IN	NO
14	74	DI	1SPH + 0 IN	NO
19	74	DI	TEH + 20.7 IN	YES
27	75	DI	TEH + 20.7 IN	YES

STEAM GENERATOR NUMBER 11

ROW	COL	INDIC	LOCATION	PLUGGED
32	76	DI	1SPH + 0 IN	NO
36	76	27%	1SPC + 0 IN	NO
30	77	DI	2SPH + 0 IN	NO
32	77	16%	2SPC + 0 IN	NO
31	79	16%	1SPC + 0 IN	NO
5	80	DI	1SPH + 0 IN	NO
30	80	19%	2SPC + 0 IN	NO
30	80	18%	1SPC + 0 IN	NO
24	81	DI	TEH + 20.7 IN	YES
30	81	25%	1SPC + 0 IN	NO
10	82	DI	1SPH + 0 IN	NO
23	82	DI	1SPH + 0 IN	NO
28	82	DI	1SPH + 0 IN	NO
19	83	DI	1SPH + 0 IN	NO
19	83	DI	2SPH + 0 IN	NO
28	83	DI	1SPH + 0 IN	NO
26	84	22%	1SPC + 0 IN	NO
27	84	18%	3SPC + 0 IN	NO
27	84	24%	2SPC + 0 IN	NO
26	85	31%	2SPC + 0 IN	NO
26	85	18%	1SPC + 0 IN	NO
10	86	DI	1SPH + 0 IN	NO
23	86	38%	2SPC + 0 IN	NO
24	86	30%	1SPC + 0 IN	NO
25	86	DI	2SPH + 0 IN	NO
1	87	DI	1SPH + 0 IN	NO
4	87	DI	1SPH + 0 IN	NO

ROW	COL	INDIC	LOCATION	PLUGGED
5	87	DI	TEH + 20.6 IN	YES
14	87	DI	1SPH + 0 IN	NO
19	87	DI	TEH + 20.9 IN	YES
21	87	DI	1SPH + 0 IN	NO
21	87	DI	4SPH + 0 IN	NO
9	88	DI	1SPH + 0 IN	NO
2	89	DI	1SPH + 0 IN	NO
2	89	DI	2SPH + 0 IN	NO
7	89	DI	1SPH + 0 IN	NO
10	90	DI	1SPH + 0 IN	NO
15	91	24%	1SPC + 0 IN	NO
10	92	DI	2SPH + 0 IN	NO
9	93	29%	1SPC + 0 IN	NO
11	93	30%	1SPC + 0 IN	NO

<u>STEAM GENERATOR NUMBER 12</u>

ROW	COL	INDIC	LOCATION	PLUGGED
8	2	DI	4SPH + 0 IN	NO
9	2	52%	1SPC + 0 IN	YES
7	3	43%	1SPC + 0 IN	YES
8	3	DI	2SPH + 0 IN	NO
9	3	28%	1SPC + 0 IN	NO
13	3	35%	1SPC + 0 IN	NO
13	4	29%	1SPC + 0 IN	NO
14	4	25%	1SPC + 0 IN	NO
15	4	50%	1SPC + 0 IN	YES
8	5	DI	1SPH + 0 IN	NO
16	5	22%	2SPC + 0 IN	NO
4	6	DI	2SPH + 0 IN	NO
8	6	DI	3SPH + 0 IN	NO
8	7	DI	2SPH + 0 IN	NO
9	7	DI	1SPH + 0 IN	NO
9	7	DI	2SPH + 0 IN	NO
4	9	DI	2SPH + 0 IN	NO
8	9	DI	1SPH + 0 IN	NO
17	9	DI	1SPH + 0 IN	NO
18	9	DI	TEH + 20.8 IN	YES
18	10	DI	1SPH + 0 IN	NO
4	12	DI	1SPH + 0 IN	NO
8	12	DI	1SPH + 0 IN	NO
8	13	DI	1SPH + 0 IN	NO
3	13	DI	1SPH + 0 IN	NO
2	14	DI	1SPH + 0 IN	NO

ROW	COL	INDIC	LOCATION	PLUGGED
12	14	DI	1SPH + 0 IN	NO
8	15	DI	1SPH + 0 IN	NO
23	15	DI	1SPH + 0 IN	NO
30	15	17%	2SPC + 0 IN	NO
23	17	DI	1SPH + 0 IN	NO
15	18	DRI	TEH + 2.4 IN	YES
8	20	DI	2SPH + 0 IN	NO
21	20	DI	1SPH + 0 IN	NO
18	21	DI	1SPH + 0 IN	NO
18	22	DI	1SPH + 0 IN	NO
8	23	DI	1SPH + 0 IN	NO
8	24	DI	2SPH + 0 IN	NO
16	24	DI	1SPH + 0 IN	NO
20	24	DI	1SPH + 0 IN	NO
16	31	DI	TSH + .6 IN	YES
35	31	45%	AV3 + 0 IN	YES
11	32	DI	1SPH + 0 IN	NO
12	37	DI	2SPH + 0 IN	NO
44	37	53%	2SPC + 0 IN	YES
44	38	14%	1SPC + 0 IN	NO
43	41	38%	AV3 + 0 IN	NO
43	41	35%	AV4 + 0 IN	NO
43	42	DI	TEH + 20.8 IN	YES
14	43	DI	TSH + 1.6 IN	YES
38	43	DI	1SPH + 0 IN	NO
15	44	DI	TSH + 1.8 IN	YES

STEAM GENERATOR NUMBER 12

ROW	COL	INDIC	LOCATION	PLUGGED
35	44	32%	AV1 + 0 IN	NO
35	44	25%	AV2 + 0 IN	NO
45	44	14%	2SPC + 0 IN	NO
45	44	14%	1SPC + 0 IN	NO
12	46	DI	TEH + 17.8 IN	YES
23	47	DI	1SPH + 0 IN	NO
38	47	DI	2SPH + 0 IN	NO
43	47	19%	AV1 + 0 IN	NO
11	48	DI	2SPH + 0 IN	NO
14	48	DRI	TEH + 2.5 IN	YES
13	51	DI	TEH + 21.1 IN	YES
18	52	DI	1SPH + 0 IN	NO
34	54	26%	AV4 + 0 IN	NO
37	54	39%	AV4 + 0 IN	NO
3	55	DI	1SPH + 0 IN	NO
36	55	41%	AV1 + 0 IN	YES
19	57	DI	1SPH + 0 IN	NO
43	60	21%	1SPC + 0 IN	NO
23	61	DI	1SPH + 0 IN	NO
43	61	DI	2SPH + 0 IN	NO
44	62	23%	2SPC + 0 IN	NO
42	63	27%	AV3 + 0 IN	NO
42	63	34%	AV4 + 0 IN	NO
42	64	14%	1SPC + 0 IN	NO
18	65	DI	1SPH + 0 IN	NO
42	65	17%	AV1 + 0 IN	NO
41	66	DI	2SPC + 0 IN	NO

ROW	COL	INDIC	LOCATION	PLUGGED
27	71	DI	TEH + 20.6 IN	YES
38	72	24%	AV3 + 0 IN	NO
38	72	28%	AV4 + 0 IN	NO
36	74	15%	1SPC + 0 IN	YES
37	74	DI	4SPH + 0 IN	NO
37	74	DI	1SPC + 0 IN	NO
21	76	DI	1SPH + 0 IN	NO
37	76	15%	2SPC + 0 IN	NO
37	76	DI	1SPC + 0 IN	NO
35	77	27%	1SPC + 0 IN	NO
36	77	33%	2SPC + 0 IN	NO
37	77	33%	1SPC + 0 IN	NO
28	78	DI	1SPH + 0 IN	NO
33	78	22%	1SPC + 0 IN	NO
34	78	15%	1SPC + 0 IN	NO
31	79	28%	2SPC + 0 IN	NO
31	79	11%	1SPC + 0 IN	NO
29	82	27%	2SPC + 0 IN	NO
30	82	25%	2SPC + 0 IN	NO
15	86	DI	TEH + 20.6 IN	YES
23	86	35%	2SPC + 0 IN	NO
23	86	14%	1SPC + 0 IN	NO
24	86	45%	2SPC + 0 IN	YES
7	89	DI	2SPH + 0 IN	NO
16	89	DI	2SPH + 0 IN	NO
19	89	DI	1SPC + 0 IN	NO
13	90	10%	1SPC + 0 IN	NO
6	91	30%	1SPC + 0 IN	NO

STEAM GENERATOR NUMBER 12

ROW	COL	INDIC	LOCATION	PLUGGED	ROW	COL	INDIC	LOCATION	PLUGGED
10	91	33%	1SPC + 0 IN	NO					
7	93	22%	1SPC + 0 IN	NO					

STEAM GENERATOR NUMBER 13

ROW	COL	INDIC	LOCATION	PLUGGED
4	1	14%	1SPC + 0 IN	NO
2	2	DI	2SPH + 0 IN	NO
7	2	DI	3SPH + 0 IN	NO
12	2	DI	2SPC + 0 IN	NO
10	3	18%	2SPC + 0 IN	NO
13	3	26%	2SPC + 0 IN	NO
14	3	DI	2SPC + 0 IN	NO
16	4	28%	2SPC + 0 IN	NO
4	5	DI	3SPH + 0 IN	NO
15	6	DI	1SPH + 0 IN	NO
26	9	DI	TEH + 20.3 IN	YES
[illegible]7	10	29%	2SPC + 0 IN	NO
6	12	DI	1SPH + 0 IN	NO
28	12	36%	2SPC + 0 IN	NO
4	13	DI	TEH + 20.1 IN	YES
10	14	DI	1SPH + 0 IN	NO
28	16	DI	TEH + 19.9 IN	YES
35	18	14%	1SPC + 0 IN	NO
9	19	DRI	TEH + 2.4 IN	YES
10	19	DI	2SPH + 0 IN	NO
34	19	52%	1SPC + 0 IN	YES
35	19	24%	1SPC + 0 IN	NO
19	20	DRI	TEH + 2.2 IN	YES
35	21	48%	2SPC + 0 IN	YES
[illegible]8	22	DRI	TEH + 2.5 IN	YES

ROW	COL	INDIC	LOCATION	PLUGGED
37	22	27%	2SPC + 0 IN	NO
10	23	DRI	TEH + 2.3 IN	YES
24	24	DRI	TEH + 2.4 IN	YES
8	27	DRI	TEH + 2.5 IN	YES
35	27	DI	2SPH + 0 IN	NO
3	28	DI	TEH + 20.8 IN	YES
31	28	DRI	TEH + 2.4 IN	YES
38	28	28%	AV2 + 0 IN	NO
38	28	30%	AV3 + 0 IN	NO
4	29	DRI	TEH + 2.5 IN	YES
22	29	DI	TEH + 20.3 IN	YES
29	29	DRI	TEH + 2.5 IN	YES
23	31	DRI	TEH + 2.6 IN	YES
25	31	DRI	TEH + 2.6 IN	YES
25	33	DRI	TEH + 2.7 IN	YES
29	33	DRI	TEH + 2.7 IN	YES
6	34	DRI	TEH + 2.9 IN	YES
14	34	DRI	TEH + 2.9 IN	YES
44	34	31%	2SPC + 0 IN	NO
24	35	DRI	TEH + 2.4 IN	YES
25	35	DRI	TEH + 3.0 IN	YES
4	36	DRI	TEH + 2.8 IN	YES
28	36	DRI	TEH + 2.4 IN	YES
44	36	32%	2SPC + 0 IN	NO
28	37	DRI	TEH + 2.6 IN	YES

STEAM GENERATOR NUMBER 13

ROW	COL	INDIC	LOCATION	PLUGGED
41	38	16%	AV3 + 0 IN	NO
41	38	30%	AV4 + 0 IN	NO
6	40	DI	TEH + 20.4 IN	YES
36	41	45%	AV2 + 0 IN	YES
41	41	19%	AV1 + 0 IN	NO
11	42	DRI	TEH + 2.3 IN	YES
38	42	28%	AV3 + 0 IN	NO
38	42	29%	AV4 + 0 IN	NO
6	45	DI	TEH + 19.9 IN	YES
15	46	DI	TSH + .1 IN	YES
30	46	DRI	TEH + 2.7 IN	YES
39	47	26%	AV3 + 0 IN	NO
39	47	23%	AV4 + 0 IN	NO
38	48	23%	AV3 + 0 IN	NO
38	48	32%	AV4 + 0 IN	NO
42	48	18%	AV1 + 0 IN	NO
42	48	16%	AV2 + 0 IN	NO
33	49	16%	AV3 + 0 IN	NO
40	50	28%	AV3 + 0 IN	NO
40	50	17%	AV4 + 0 IN	NO
41	50	34%	AV3 + 0 IN	NO
41	50	25%	AV4 + 0 IN	NO
2	51	DRI	TEH + 2.4 IN	YES
6	51	DRI	TEH + 2.4 IN	YES
22	51	DI	TSH + .3 IN	YES
38	51	18%	AV3 + 0 IN	NO
38	51	17%	AV4 + 0 IN	NO
39	51	DI	2SPH + 0 IN	NO
44	51	DI	1SPH + 0 IN	NO

ROW	COL	INDIC	LOCATION	PLUGGED
12	52	DRI	TEH + 2.3 IN	YES
19	52	DI	TSH + .9 IN	YES
28	52	DRI	TEH + 2.4 IN	YES
33	52	41%	AV3 + 0 IN	YES
4	53	DRI	TEH + 2.3 IN	YES
18	53	DI	TSH + 1.1 IN	YES
36	53	30%	AV3 + 0 IN	NO
36	53	18%	AV4 + 0 IN	NO
7	54	DRI	TEH + 2.6 IN	YES
37	55	18%	AV2 + 0 IN	NO
40	55	29%	AV3 + 0 IN	NO
40	55	18%	AV4 + 0 IN	NO
5	56	DRI	TEH + 2.8 IN	YES
6	56	DRI	TEH + 2.5 IN	YES
36	56	38%	AV1 + 0 IN	NO
36	56	37%	AV2 + 0 IN	NO
37	56	32%	AV1 + 0 IN	NO
37	56	37%	AV2 + 0 IN	NO
37	56	30%	AV3 + 0 IN	NO
24	57	DI	1SPH + 0 IN	NO
36	57	35%	AV3 + 0 IN	NO
36	57	28%	AV4 + 0 IN	NO
6	58	DI	1SPH + 0 IN	NO
8	58	DI	1SPH + 0 IN	NO
39	58	17%	AV3 + 0 IN	NO
39	58	23%	AV4 + 0 IN	NO
41	58	45%	AV3 + 0 IN	YES
26	59	DRI	TEH + 2.5 IN	YES
5	60	DI	1SPH + 0 IN	NO

STEAM GENERATOR NUMBER 13

ROW	COL	INDIC	LOCATION	PLUGGED
23	60	DRI	TEH + 2.4 IN	YES
30	60	DRI	TEH + 2.5 IN	YES
34	60	18%	AV4 + 0 IN	NO
44	60	16%	1SPC + 0 IN	NO
34	61	18%	AV3 + 0 IN	NO
34	61	25%	AV4 + 0 IN	NO
36	61	30%	AV3 + 0 IN	NO
36	61	35%	AV4 + 0 IN	NO
41	61	23%	AV3 + 0 IN	NO
32	62	25%	AV3 + 0 IN	NO
40	62	15%	AV1 + 0 IN	NO
40	62	31%	AV2 + 0 IN	NO
40	62	28%	AV3 + 0 IN	NO
[illegible]0	62	27%	AV4 + 0 IN	NO
43	62	DI	1SPC + 0 IN	NO
10	64	DI	1SPH + 0 IN	NO
33	64	25%	AV1 + 0 IN	NO
33	64	17%	AV2 + 0 IN	NO
36	64	25%	AV3 + 0 IN	NO
36	64	23%	AV4 + 0 IN	NO
40	64	30%	AV1 + 0 IN	NO
43	64	32%	1SPC + 0 IN	NO
24	65	DI	1SPH + 0 IN	NO
36	65	24%	AV3 + 0 IN	NO
36	65	26%	AV4 + 0 IN	NO
39	65	18%	AV3 + 0 IN	NO
39	65	26%	AV4 + 0 IN	NO
40	65	19%	AV2 + 0 IN	NO
40	65	27%	AV3 + 0 IN	NO
[illegible]9	66	DI	TEH + 20.6 IN	YES
5	67	DI	1SPH + 0 IN	NO
11	67	DI	1SPH + 0 IN	NO
3	68	DRI	TEH + 2.4 IN	YES
10	69	DRI	TEH + 2.4 IN	YES
39	69	26%	3SPC + 0 IN	NO
38	70	33%	3SPC + 0 IN	NO
39	70	38%	3SPC + 0 IN	NO
4	71	DRI	TEH + 2.4 IN	YES
9	71	DRI	TEH + 2.2 IN	YES
39	71	DI	2SPH + 0 IN	NO
4	73	DRI	TEH + 2.5 IN	YES
24	73	DI	2SPH + 0 IN	NO
37	73	29%	1SPC + 0 IN	NO
4	74	DRI	TEH + 2.4 IN	YES
34	76	28%	2SPC + 0 IN	NO
36	76	29%	AV3 + 0 IN	NO
8	77	DRI	TEH + 2.3 IN	YES
32	78	14%	1SPC + 0 IN	NO
6	80	DI	1SPH + 0 IN	NO
31	80	16%	4SPC + 0 IN	NO
30	81	37%	3SPC + 0 IN	NO
30	81	33%	1SPC + 0 IN	NO
10	83	DI	2SPH + 0 IN	NO
28	83	DI	1SPC + 0 IN	NO
29	84	DI	1SPH + 0 IN	NO
6	86	DI	2SPH + 0 IN	NO

STEAM GENERATOR NUMBER 13

ROW	COL	INDIC	LOCATION	PLUGGED	ROW	COL	INDIC	LOCATION	PLUGGED
10	86	DI	1SPH + 0 IN	NO					
21	87	11%	1SPC + 0 IN	NO					
7	89	DI	2SPH + 0 IN	NO					
9	91	DI	2SPH + 0 IN	NO					
10	92	17%	1SPC + 0 IN	NO					
10	93	28%	1SPC + 0 IN	NO					
4	94	DI	1SPH + 0 IN	NO					
5	94	19%	1SPC + 0 IN	NO					
7	94	25%	1SPC + 0 IN	NO					

STEAM GENERATOR NUMBER 14

ROW	COL	INDIC	LOCATION	PLUGGED
6	3	DI	TEH + 20.8 IN	YES
9	3	28%	1SPC + 0 IN	NO
6	4	DI	TEH + 20.8 IN	YES
15	5	DI	1SPC + 0 IN	NO
19	6	11%	1SPC + 0 IN	NO
11	7	DI	1SPH + 0 IN	NO
20	7	DI	1SPC + 0 IN	NO
21	7	30%	1SPC + 0 IN	NO
22	7	12%	1SPC + 0 IN	NO
11	8	DI	3SPH + 0 IN	NO
11	9	DI	1SPH + 0 IN	NO
25	9	24%	1SPC + 0 IN	NO
16	10	DI	2SPH + 0 IN	NO
25	10	DI	1SPH + 0 IN	NO
27	10	16%	1SPC + 0 IN	NO
22	12	DI	TEH + 20.6 IN	YES
26	12	16%	1SPC + 0 IN	NO
28	12	DI	5SPH + 0 IN	NO
28	13	DI	1SPH + 0 IN	NO
30	13	DI	1SPH + 0 IN	NO
18	14	DI	TEH + 21.1 IN	YES
31	14	18	1SPC + 0 IN	NO
8	15	DI	1SPH + 0 IN	NO
9	15	DI	TEH + 20.1 IN	YES
11	15	DI	1SPH + 0 IN	NO
28	15	DI	TEH + 20.5 IN	YES
31	15	17%	1SPC + 0 IN	NO
8	17	DI	2SPH + 0 IN	NO
34	17	38%	1SPC + 0 IN	NO
2	19	DI	2SPH + 0 IN	NO
8	20	DI	1SPH + 0 IN	NO
9	20	DI	1SPH + 0 IN	NO
15	20	DI	TEH + 20.7 IN	YES
37	20	DI	1SPH + 0 IN	NO
28	21	DI	1SPH + 0 IN	NO
28	21	DI	2SPH + 0 IN	NO
30	21	DI	1SPH + 0 IN	NO
37	21	DI	TEH + 20.6 IN	YES
38	21	DI	1SPH + 0 IN	NO
11	22	DI	1SPH + 0 IN	NO
11	22	DI	3SPH + 0 IN	NO
28	22	DI	1SPH + 0 IN	NO
28	22	DI	2SPH + 0 IN	NO
11	24	DI	3SPH + 0 IN	NO
15	26	DRI	TEH + 2.5 IN	YES
38	27	DI	1SPH + 0 IN	NO
6	28	DRI	TEH + 2.6 IN	YES
19	28	DRI	TEH + 2.3 IN	YES
40	29	DI	1SPH + 0 IN	NO
42	29	DI	1SPC + 0 IN	NO
5	30	DI	1SPH + 0 IN	NO
5	30	DI	2SPH + 0 IN	NO
5	30	DI	3SPH + 0 IN	NO

STEAM GENERATOR NUMBER 14

ROW	COL	INDIC	LOCATION	PLUGGED
8	30	DI	TEH + 20.9 IN	YES
32	30	DI	1SPH + 0 IN	NO
34	30	DI	TEH + 20.7 IN	YES
37	30	DI	TEH + 20.5 IN	YES
42	30	23%	1SPC + 0 IN	NO
5	31	DI	2SPH + 0 IN	NO
41	31	13%	1SPC + 0 IN	NO
32	32	DI	1SPH + 0 IN	NO
42	32	38%	1SPC + 0 IN	NO
12	33	29%	2SPH + 0 IN	NO
38	33	DI	TEH + 20.4 IN	YES
30	34	DI	1SPH + 0 IN	NO
44	34	27%	1SPC + 0 IN	NO
17	35	32%	TSH + .5 IN	YES
33	35	DI	1SPH + 0 IN	NO
44	35	19%	1SPC + 0 IN	YES
44	36	26%	1SPC + 0 IN	NO
45	36	16%	1SPC + 0 IN	NO
17	37	DI	1SPH + 0 IN	NO
29	37	DI	1SPH + 0 IN	NO
38	37	DI	2SPH + 0 IN	NO
44	37	39	2SPC + 0 IN	NO
45	37	14%	2SPC + 0 IN	NO
45	37	16%	1SPC + 0 IN	NO
[illegible]9	39	DI	1SPH + 0 IN	NO
38	39	DI	1SPH + 0 IN	NO

ROW	COL	INDIC	LOCATION	PLUGGED
44	39	37%	4SPC + 0 IN	NO
42	40	DI	2SPH + 0 IN	NO
29	41	DI	1SPH + 0 IN	NO
38	41	DI	1SPH + 0 IN	NO
42	41	DI	1SPH + 0 IN	NO
45	41	19%	2SPC + 0 IN	NO
46	41	16%	2SPC + 0 IN	NO
21	43	DI	1SPH + 0 IN	NO
44	43	DI	2SPC + 0 IN	NO
46	43	24%	2SPC + 0 IN	NO
10	45	DI	1SPH + 0 IN	NO
17	45	DI	TSH + .5 IN	YES
41	45	23%	AV3 + 0 IN	NO
41	45	18%	AV4 + 0 IN	NO
16	46	DI	TSH + 1.1 IN	YES
22	46	DI	TEH + 21.2 IN	YES
33	46	24%	AV2 + 0 IN	NO
33	46	34%	AV3 + 0 IN	NO
33	46	26%	AV4 + 0 IN	NO
42	46	DI	TEH + 20.8 IN	YES
46	46	19%	2SPH + 0 IN	NO
38	48	18%	TSH + 42.8 IN	NO
32	49	DI	2SPH + 0 IN	NO
46	49	DI	1SPH + 0 IN	NO
8	50	DRI	TEH + 2.4 IN	YES
12	50	DI	1SPH + 0 IN	NO
46	50	DI	1SPH + 0 IN	NO

STEAM GENERATOR NUMBER 14

ROW	COL	INDIC	LOCATION	PLUGGED
23	51	DI	1SPH + 0 IN	NO
45	51	13%	2SPC + 0 IN	NO
18	52	DRI	TEH + 2.1 IN	YES
18	54	DRI	TEH + 2.2 IN	YES
44	55	DI	2SPH + 0 IN	NO
30	57	DI	1SPH + 0 IN	NO
30	57	DI	3SPH + 0 IN	NO
22	60	DI	1SPH + 0 IN	NO
44	60	14%	3SPC + 0 IN	NO
12	61	DI	1SPH + 0 IN	NO
17	61	DI	1SPH + 0 IN	NO
20	61	DI	1SPH + 0 IN	NO
30	61	DI	1SPH + 0 IN	NO
43	61	31%	1SPC + 0 IN	YES
13	62	DI	1SPH + 0 IN	NO
35	62	DI	1SPH + 0 IN	NO
19	63	22%	AV2 + 0 IN	NO
19	63	19%	AV3 + 0 IN	NO
19	63	11%	AV4 + 0 IN	NO
6	65	DI	1SPH + 0 IN	NO
17	65	DI	1SPH + 0 IN	NO
41	65	34%	2SPC + 0 IN	NO
19	66	15%	AV1 + 0 IN	NO
19	66	22%	AV2 + 0 IN	NO
19	66	24%	AV3 + 0 IN	NO
25	66	DI	1SPH + 0 IN	NO
34	67	DI	TEH + 20.7 IN	YES
26	69	15%	AV1 + 0 IN	NO.
26	69	16%	AV2 + 0 IN	NO
18	70	DI	1SPH + 0 IN	NO
31	70	DI	1SPH + 0 IN	NO
30	73	DI	1SPH + 0 IN	NO
33	73	DI	1SPH + 0 IN	NO
39	73	16%	2SPC + 0 IN	NO
4	74	DI	2SPH + 0 IN	NO
5	74	DI	1SPH + 0 IN	NO
18	74	DI	TEH + 20.6 IN	YES
33	74	DI	1SPH + 0 IN	NO
33	74	DI	2SPH + 0 IN	NO
13	75	DI	1SPH + 0 IN	NO
33	75	DI	2SPH + 0 IN	NO
35	75	24%	2SPC + 0 IN	NO
36	75	38%	2SPC + 0 IN	NO
4	76	DI	2SPH + 0 IN	NO
17	76	DI	TEH + 20.9 IN	YES
18	76	DI	TEH + 20.9 IN	YES
19	76	DI	TEH + 20.6 IN	YES
31	76	DI	3SPH + 0 IN	NO
33	76	DI	1SPH + 0 IN	NO
36	76	21%	1SPC + 0 IN	NO
23	77	DI	TEH + 20.8 IN	YES
33	77	54%	2SPC + 0 IN	YES
23	78	DI	3SPH + 0 IN	NO

STEAM GENERATOR NUMBER 14

ROW	COL	INDIC	LOCATION	PLUGGED	ROW	COL	INDIC	LOCATION	PLUGGED
26	78	DI	2SPH + 0 IN	NO					
26	80	22%	AV3 + 0 IN	NO					
24	81	DI	2SPH + 0 IN	NO					
29	81	10%	2SPC + 0 IN	NO					
30	81	28%	2SPC + 0 IN	NO					
1	82	DI	2SPH + 0 IN	NO					
11	82	DI	2SPH + 0 IN	NO					
30	82	31%	2SPC + 0 IN	NO					
29	83	DI	1SPC + 0 IN	NO					
28	84	10%	1SPC + 0 IN	NO					
26	86	DI	2SPH + 0 IN	NO					
26	86	38%	2SPC + 0 IN	NO					
23	87	31	2SPC + 0 IN	NO					
12	88	DI	TEH + 21.1 IN	YES					
16	91	28%	1SPC + 0 IN	NO					
12	92	25%	1SPC + 0 IN	NO					
12	93	24%	2SPC + 0 IN	NO					
12	93	23%	1SPC + 0 IN	NO					
5	94	11%	1SPC + 0 IN	NO					

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DONALD C. COOK NUCLEAR PLANT (COOK NUCLEAR PLANT)

UNIT NUMBERS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

LICENSE NOS. DPR-58 AND DPR-74

UPDATED QUALITY ASSURANCE PROGRAM DESCRIPTION FOR THE COOK NUCLEAR PLANT

JULY, 1991

Concurred by: E. E. Fitzpatrick Date: 4/10/91
AEPSC Vice President - Nuclear Operations

: [signature] Date: 4/10/91
AEPSC Senior Vice President and Chief Engineer

Approved by: P. G. Barry Date: 4/11/91
AEPSC Director - Quality Assurance

QUALITY ASSURANCE PROGRAM DESCRIPTION

FOR THE

COOK NUCLEAR PLANT

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American Electric Power
Service Corporation
1 Riverside Plaza
Columbus OH 43215
614 223 1000

AMERICAN ELECTRIC POWER

STATEMENT OF POLICY
FOR THE DONALD C. COOK NUCLEAR PLANT
QUALITY ASSURANCE PROGRAM

POLICY

American Electric Power Company, Inc., recognizes the fundamental importance of controlling the design, modification, and operation of Indiana Michigan Power Company's Donald C. Cook Nuclear Plant (Cook Nuclear Plant) by implementing a planned and documented Quality Assurance Program, including Quality Control, that complies with applicable regulations, codes, and standards.

The Quality Assurance Program has been established to control activities affecting safety-related functions of structures, systems, and components in the Cook Nuclear Plant. The Quality Assurance Program supports the goal of maintaining the safety and reliability of the Cook Nuclear Plant at the highest level through a systematic program designed to assure that safety-related items are conducted in compliance with the applicable regulations, codes, standards, and established corporate policies and practices.

As President and Chief Executive Officer of American Electric Power Company, Inc., I maintain the ultimate responsibility for the Quality Assurance Program associated with the Cook Nuclear Plant. I have delegated functional responsibility for the Quality Assurance Program to the American Electric Power Service Corporation (AEPSC) Senior Executive Vice President-Engineering and Construction. He has, with my approval, delegated further responsibilities as outlined in this statement.

IMPLEMENTATION

The AEPSC Director-Quality Assurance, under the direction of the AEPSC Senior Executive Vice President-Engineering and Construction, has been assigned the overall responsibility for specifying the Quality Assurance program requirements for the Cook

Nuclear Plant and verifying their implementation. The AEPSC Senior Executive Vice President-Engineering and Construction has given the AEPSC Director-Quality Assurance authority to stop work on any activity affecting safety-related items that does not meet applicable administrative, technical, and/or regulatory requirements. The AEPSC Director-Quality Assurance does not have the authority to stop unit operations, but shall notify appropriate plant and/or corporate management of conditions not meeting the aforementioned criteria and recommend that unit operations be terminated.

The AEPSC Vice President-Nuclear Operations, under the direction of the AEPSC Senior Executive Vice President-Engineering and Construction, has been delegated responsibility for effectively implementing the Quality Assurance Program. The AEPSC Vice President-Nuclear Operations is the Manager of Nuclear Operations. All other AEPSC divisions and departments, except Quality Assurance, having a supporting role for the Cook Nuclear Plant are functionally responsible to the Manager of Nuclear Operations.

The Plant Manager, under the direction of the AEPSC Vice President-Nuclear Operations, is delegated the responsibility for establishing the Cook Nuclear Plant Quality Control Program and implementing the Quality Assurance Program at the Cook Nuclear Plant.

The AEPSC Director-Quality Assurance is responsible for providing technical direction to the Plant Manager for matters relating to the Quality Assurance Program at the Cook Nuclear Plant. The AEPSC Director-Quality Assurance is also responsible for maintaining a Quality Assurance Section at the Cook Nuclear Plant to perform required reviews, audits, and surveillances, and to provide technical liaison services to the Plant Manager.

The implementation of the Quality Assurance Program is described in the AEPSC General Procedures (GPs) and subtier department/ division procedures, Plant Manager's Instructions (PMIs), and subtier Department Head Instructions and Procedures, which in total document the requirements for implementation of the Program.

Each AEPSC and Cook Nuclear Plant organization involved in activities affecting safety-related functions of structures, systems, and components in the Cook Nuclear Plant has the responsibility to implement the applicable policies and requirements of the Quality Assurance Program. This responsibility includes being familiar with, and complying with, the requirements of the applicable Quality Assurance Program requirements.

COMPLIANCE

The AEPSC Director-Quality Assurance shall monitor compliance with the established Quality Assurance Program. Audit programs shall be established to ensure that AEPSC and Cook Nuclear Plant activities comply with established program requirements, identify deficiencies or noncompliances and obtain effective and timely corrective actions.

Employees engaged in activities affecting safety-related functions of structures, systems, and components in the Cook Nuclear Plant who believe that the Quality Assurance Program is not being complied with, or that a deficiency in quality exists, should notify their supervisor, the AEPSC Director-Quality Assurance, and/or the Plant Manager. If the notification does not in the employee's opinion receive prompt or appropriate attention, the employee should contact successively higher levels of management. Employees reporting such conditions shall not be discriminated against by companies of the American Electric Power System. Discrimination includes discharge or other actions relative to compensation, terms, conditions, or privileges of employment.

R. E. Disbrow
President and
Chief Executive Officer
American Electric Power Company, Inc.

1.7.1 ORGANIZATION

1.7.1.1 SCOPE

American Electric Power Service Corporation (AEPSC) is responsible for establishing and implementing the Quality Assurance (QA) Program for the operational phase of the Donald C. Cook Nuclear Plant (Cook Nuclear Plant). Although authority for development and execution of various portions of the program may be delegated to others, such as contractors, agents or consultants, AEPSC retains overall responsibility. AEPSC shall evaluate work delegated to such organizations. Evaluations shall be based on the status of safety importance of the activity being performed and shall be initiated early enough to assure effective quality assurance during the performance of the delegated activity.

This section of the Quality Assurance Program Description (QAPD) identifies the AEPSC organizational responsibilities for activities affecting the quality of safety-related nuclear power plant structures, systems, and components, and describes the authority and duties assigned to them. It addresses responsibilities for both attaining quality objectives and for the functions of establishing the QA Program, and verifying that activities affecting the quality of safety-related items are performed effectively in accordance with QA Program requirements.

1.7.1.2 IMPLEMENTATION

1.7.1.2.1 <u>Source of Authority</u>

The President and Chief Executive Officer of American Electric Power Company, Inc. (AEP) and AEPSC is responsible for safe operation of the Cook Nuclear Plant. Authority and responsibility for effectively implementing the QA Program for plant modifications, operations and maintenance are delegated through the AEPSC Senior Executive Vice President - Engineering and Construction, to the AEPSC Vice President - Nuclear Operations (Manager of Nuclear Operations).

In the operation of a nuclear power plant, the licensee is required to establish clear and direct lines of responsibility, authority and accountability. This requirement is applicable to the organization providing support to the plant, as well as to the plant staff. |*

The AEPSC corporate support of the Cook Nuclear Plant is the responsibility of the entire organization under the direction of the Manager of Nuclear Operations who maintains primary responsibility for the Cook Nuclear Plant within the corporate organization. The AEPSC Vice President - Nuclear Operations is the Manager of Nuclear Operations. All other AEPSC divisions and departments, other than the Quality Assurance Division, having a supporting role for the Cook Nuclear Plant are functionally responsible to the Manager of Nuclear Operations (reference Figure 1.7-1).

In order to facilitate a more thorough understanding of the support functions, some of the responsibilities, authorities, and accountabilities within the organization are as follows:

1) The responsibilities of the Manager of Nuclear Operations shall be dedicated to the area of Cook Nuclear Plant operations and support.

2) The Manager of Nuclear Operations shall be responsible for, and has the authority to direct, all Cook Nuclear Plant operational |* and support matters within the corporation and shall make, or |* concur, in all final decisions regarding significant nuclear |* safety matters.

3) AEPSC organization managers responsible for Cook Nuclear Plant matters shall be familiar with activities within their scope of responsibility that affect plant safety and reliability. They shall be cognizant of, and sensitive to, internal and external |* factors that might affect the operations of Cook Nuclear Plant.

4) AEPSC organization managers responsible for Cook Nuclear Plant matters have a commitment to seek and identify problem areas and take corrective action to eliminate unsafe conditions, or to improve trends that will upgrade plant safety and reliability.

5) The Manager of Nuclear Operations shall ensure that Cook Nuclear Plant personnel are not requested to perform inappropriate work or tasks by corporate personnel, and shall control assignments and requests that have the potential for diverting the attention of the Plant Manager from the primary responsibility for safe and reliable plant operation. |*

6) AEPSC division and department managers having Cook Nuclear Plant support responsibilities, as well as the Plant Manager and plant organization managers, shall be familiar with the policy statements from higher management concerning nuclear safety and operational priorities. They shall be responsible for ensuring that activities under their direction are performed in accordance with these policies. |* |*

1.7.1.2.2 <u>Responsibility for Attaining Quality Objectives in AEPSC Nuclear Operations</u>

The AEP President and Chief Executive Officer has delegated the functional responsibility of the Quality Assurance Program to the AEPSC Senior Executive Vice President - Engineering and Construction. |

The AEPSC Director - Quality Assurance, under the direction of the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for specifying QA Program requirements and verifying their implementation.

The AEPSC Vice President - Nuclear Operations, under the direction of the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for effectively implementing the QA Program.

The Plant Manager, under the direction of the AEPSC Vice President - Nuclear Operations, is responsible for establishing the Cook Nuclear Plant Quality Control Program and implementing the QA Program at the Cook Nuclear Plant.

Management/supervisory personnel receive functional training to the level necessary to plan, coordinate, and administrate those day-to-day verification activities of the QA Program for which they are responsible.

AEPSC has an independent off-site Nuclear Safety and Design Review Committee (NSDRC) which has been established pursuant to the requirements of the Technical Specifications for the Cook Nuclear Plant. The function of the NSDRC is to oversee the engineering, design, operation, and maintenance of the Cook Nuclear Plant by performing audits and independent reviews of activities which are specified in the Facility Operating Licenses.

The Cook Nuclear Plant on-site review group is the Indiana Michigan Power Company (I&M) Plant Nuclear Safety Review Committee (PNSRC). This committee has also been established pursuant to the requirements of the Cook Nuclear Plant Technical Specifications. The function of the PNSRC is to review plant operations on a continuing basis and advise the Plant Manager on matters related to nuclear safety.

1.7.1.2.3 Corporate Organization

American Electric Power Company

AEP, the parent holding company, wholly owns the common stock of all AEP System subsidiary (operating) companies. The major operating companies and generation subsidiaries are shown in Figure 1.7-2. The President and Chief Executive Officer of AEP is the Chief Executive Officer of AEPSC and all operating companies. The responsibility for the functional management of the major operating companies is vested in the President of each operating company reporting to the AEPSC President and Chief Operating Officer who reports to the AEPSC Chairman of the Board.

American Electric Power Service Corporation

The responsibility for administrative and technical direction of the AEP System and its facilities is delegated to AEPSC. AEPSC provides management and technological services to the various AEP System companies.

Operating Companies

The operating facilities of the AEP System are owned and operated by the respective operating companies. The responsibility for executing the engineering, design, construction, specialized technical training, and certain operations' supervision is vested in AEPSC, while all, or part, of the administrative functional responsibility is assigned to the operating companies. In the case of Cook Nuclear Plant, I&M general office staff (headquarters) provides public affairs, accounting, industrial safety direction and procurement support. |*

The Cook Nuclear Plant is owned and operated by I&M which is part of the AEP System.

1.7.1.2.4 Quality Assurance Responsibility of AEPSC

1) AEPSC provides the technical direction for the Cook Nuclear Plant, and as such makes the final decisions pertinent to safety-related changes in plant design. Further, AEPSC reviews Nuclear Regulatory Commission (NRC) letters, bulletins, notices, etc., for impact on plant design, and the need for design changes or modifications.

2) AEPSC furnishes quality assurance, engineering, design, construction, licensing, NRC correspondence, fuel management and radiological support activities.

3) AEPSC provides additional service in matters such as supplier qualification, procurement of original equipment and replacement parts, and the process of dedicating commercial grade items or services to safety-related applications. |*

4) The AEPSC QA Division provides technical direction in quality assurance matters to AEPSC and the Cook Nuclear Plant, and oversees the adequacy, effectiveness and implementation of the QA Program through review and audit activities.

5) Cognizant Engineer - (e.g., System Engineer, Equipment Engineer, Lead Engineer, Responsible Engineer, etc.) - The cognizant engineer, and/or engineer with the other titles noted, is that AEPSC individual who provides the engineering/design expertise for a particular area of responsibility. This responsibility includes the implementation of the quality assurance and quality control measures for systems, equipment, structures, or functional areas included in that individual's responsibility. The various titles used for the identification of an individual's responsibility and assignment shall be understood to mean the same as cognizant engineer in the respective areas of responsibility.

Quality Assurance Responsibility of I&M - Cook Nuclear Plant

I&M's Cook Nuclear Plant staff operates the Cook Nuclear Plant in accordance with licensing requirements, including the Technical Specifications and such other commitments as established by the operating licenses. The Plant Manager Instruction (PMI) system and subtier instructions and procedures describe the means by which compliance is achieved and responsibilities are assigned, including interfaces with AEPSC. Figure 1.7-3 indicates the organizational relationships within the AEP System pertaining to the operation and support of the Cook Nuclear Plant.

1.7.1.2.5 Organization (AEPSC)

The President and Chief Executive Officer is ultimately responsible for the QA Program associated with the Cook Nuclear Plant. This responsibility has been functionally delegated to the AEPSC Senior Executive Vice President - Engineering and Construction. The AEPSC Senior Executive Vice President - Engineering and Construction has further delegated responsibilities which are administered through the following division and department management personnel:

- AEPSC Director - Quality Assurance
- AEPSC Vice President - Nuclear Operations
- AEPSC Senior Vice President and Chief Engineer
- AEPSC Vice President - Project Management and Construction

Quality Assurance Division

The AEPSC Director - Quality Assurance, reporting to the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for the Quality Assurance Division (QAD). The QAD consists of the following sections (Figure 1.7-4):

- Quality Assurance Engineering Section
- Nuclear Software Quality Assurance Section
- Audits and Procurement Section

- Quality Assurance Support Section
- Quality Assurance Section (Site)

The QAD is organizationally independent and is responsible to perform the following:

- Specify QA Program requirements.
- Identify quality problems.
- Initiate, recommend, or provide solutions through designated channels.
- Verify implementation of solutions, as appropriate.
- Prepare, issue and maintain QA Program documents, as required.
- Verify the implementation of the QA Program through scheduled audits and surveillances.
- Verify the implementation of computer software quality assurance through reviews, surveillances and audits.
- Audit engineering, design, procurement, construction and operational documents for incorporation of, and compliance with, applicable quality assurance requirements to the extent specified by the AEPSC management-approved QA Program. |*
- Organize and conduct the QA auditor orientation, training, certification and qualification of AEPSC audit personnel.
- Provide direction for the collection, storage, maintenance, and retention of quality assurance records.
- Maintain, on data base, a list of suppliers of nuclear (N) items and services, plus other selected categories of suppliers.
- Identify noncompliances of the established QA Program to the responsible organizations for corrective actions, and report significant occurrences that jeopardize quality to senior AEPSC management. |*
- Follow up on corrective actions identified by QA during and after disposition implementation.
- Review the disposition of conditions adverse to quality to assure that action taken will preclude recurrence.
- Conduct in-process QA audits or surveillances at supplier's facilities, as required. |

- Assist and advise other AEP/AEPSC groups in matters related to the QA Program.
- Conduct audits as directed by the NSDRC.
- Review AEPSC investigated Problem Reports and associated corrective and preventive action recommendations.
- Maintain cognizance of industry and governmental quality assurance requirements such that the QA Program is compatible with requirements, as necessary.
- Recommend for revision to, or improvements in, the established QA Program to senior AEPSC management. |*
- Audit dedication plans for commercial grade items and services.
- Issue "Stop Work" orders when significant conditions adverse to safety-related items are identified to prevent unsafe conditions from occurring and/or continuing.
- Provide AEPSC management with periodic reports concerning the status, adequacy and implementation of the QA Program.
- Prepare and conduct special verification and/or surveillance programs on in-house activities, as required or requested.
- Routinely attend, and participate in, daily plant work schedule and status meetings. |*
- Provide adequate QA coverage relative to procedural and inspection controls, acceptance criteria, and QA staffing and qualification of personnel to carry out QA assignments.
- Establish and maintain a central file for equipment environmental qualification documentation.

<u>Amplification of Specific Responsibilities</u>

- <u>Qualification of the AEPSC Director - Quality Assurance</u>
  The AEPSC Director - Quality Assurance shall possess the following position requirements:
  - Bachelor's degree in engineering, scientific, or related discipline. |*
  - Ten (10) years experience in one of, or a combination of, the following areas: engineering, design, |*

construction, operations, maintenance of fossil or nuclear power generation facilities' or utility facilities' QA, of which at least four (4) years must be experience in nuclear quality assurance related activities. |*

- Knowledge of QA regulations, policies, practices and standards.
- The same, or higher, organization reporting level as the highest line manager directly responsible for performing activities affecting the quality of safety-related items, such as engineering, procurement, construction and operation, and is sufficiently independent from cost and schedule. |*
- Effective communication channels with other senior management positions.
- Responsibility for approval of QA Manual(s).
- Performance of no other duties or responsibilities unrelated to QA that would prevent full attention to QA matters.

- <u>Stop Work Orders</u>

The AEPSC QAD is responsible for ensuring that activities affecting the quality of safety-related items are performed in a manner that meets applicable administrative, technical, and regulatory requirements. In order to carry out this responsibility, the AEPSC Senior Executive Vice President - Engineering and Construction has given the AEPSC Director - Quality Assurance the authority to stop work on any activity affecting the quality of safety-related items that does not meet the aforementioned requirements. Stop work authority has been further delegated by the AEPSC Director - Quality Assurance to the AEPSC Quality Assurance Superintendent (site). |*

The AEPSC Director - Quality Assurance and the AEPSC Quality Assurance Superintendent do not have the authority to stop unit operations, but will notify appropriate Cook Nuclear Plant and/or corporate management of conditions which do not meet the aforementioned criteria, and recommend that unit operations be terminated.

- <u>QA Auditor, Qualification and Certification Program</u>

- AEPSC has established and maintains a QA auditor training and certification program for all AEPSC QA auditors.

- <u>Problem Identification, Reporting and Escalation</u>

- AEPSC has established mechanisms for the identification, reporting and escalation of problems affecting the quality of safety-related items to a level of management whereby satisfactory resolutions can be obtained.

<u>Nuclear Operations Division</u>

The AEPSC Vice President - Nuclear Operations (Manager of Nuclear Operations), reporting to the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for the Nuclear Operations Division (NOD).

The organization and responsibilities of the Plant Manager are defined further within this section under 1.7.1.2.6 <u>Organization (Cook Nuclear Plant)</u>.

NOD is responsible for the following:

- Formulate policies and practices relative to safety, licensing, operation, maintenance, fuel management, and radiological support.

- Provide the Plant Manager with the technical and managerial guidance, direction and support to ensure the safe operation of the plant.
- Provide direction to all other AEPSC engineering and design organizations on engineering and design matters pertaining to the Cook Nuclear Plant.
- Maintain liaison with the AEPSC Director - Quality Assurance.
- Implement the requirements of the AEPSC QA Program.
- Maintain knowledge of the latest safety, licensing, and regulatory requirements, codes, standards, and federal regulations applicable to the operation of Cook Nuclear Plant. |*
- Accomplish the procurement, economic, technical, licensing and quality assurance activities dealing with the reactor core and its related fuel assemblies and components.
- Prepare bid specifications, evaluate bids, and negotiate and administer contracts for the procurement of all nuclear fuel and related components and services.
- Maintain a special nuclear material accountability system.
- Provide analyses to support nuclear steam supply system operation, including reactor physics, fuel economics, fuel mechanical behavior, core thermal hydraulic and LOCA and non-LOCA transient safety analysis and other analysis activities as requested, furnish plant Technical Specification changes and other licensing work, and participate in NRC and NSDRC meetings as required by these analyses. |*
- Perform reactor core operation follow-up activities and other reactor core technical support activities as requested, and arrange for support from the fuel fabricator, when needed. |*
- Contract for, and provide technical support for, disposal of both high level and low level radioactive waste.
- Coordinate the development of neutronics and thermal hydraulic safety codes and conduct safety analyses.
- Conduct studies of the Cook Nuclear Plant licensing bases to determine the optimal changes to support unit operations at a lower primary pressure and temperature.

- Coordinate NOD computer code development, and provide the interface control for NOD with the AEPSC Information System Department and Cook Nuclear Plant. |*
- Obtain and maintain the NRC Operating License and Technical Specifications for the Cook Nuclear Plant.
- Act as the communication link between the NRC, AEPSC, and the plant staff.
- Perform and coordinate efforts involved in gathering information, performing calculations and generic studies; preparing criteria, reports, and responses; reviewing items affecting safety; and interpreting regulations. |* |*
- Review, coordinate, and resolve all matters pertaining to nuclear safety between Cook Nuclear Plant and AEPSC. This includes, but is not limited to: the review of certain plant design changes to ensure that the requirements of 10CFR50.59 are met; the preparation of safety evaluations, or reviews, for any designated subject; the preparation of changes to, and appropriate interpretation of, the plant Technical Specification submittals of license amendments; and the analysis of plant compliance with regulatory requirements. |*
- Primary corporate contact for most oral and written communication with the NRC.
- Provide support in key areas of expertise, such as nuclear engineering, probabilistic analysis, thermohydraulic analysis, chemical engineering, mechanical engineering, electrical engineering, and technical writing. |*
- Interface with vendors and other outside organizations on matters connected with the nuclear steam supply system and other areas affecting the safe design and operation of nuclear plants.
- Participate, as appropriate, in the review of nuclear plant operating experiences, and relate those experiences to the design and safe operation of Cook Nuclear Plant. |*

- Review, evaluate, and respond to NRC requests for information and NRC notifications of regulatory changes resulting in plant modifications or new facilities. Such responses are generated in accordance with appropriate AEPSC Administrative Procedures.
- Develop, specify, and/or review conceptual nuclear safety criteria for Cook Nuclear Plant in accordance with established regulations. |* This includes all information contained in the FSAR, as well as specialized information such as environmental qualification and seismic criteria.
- Review and evaluate performance requirements for systems, equipment and materials for compliance with specified safety criteria.
- Review, on a conceptual basis, plant reports and proposed plant safety-related design changes, to the extent that they are related to the ultimate safe operation of the plant, for compliance with safety regulations, plant Technical Specifications, the Updated | FSAR design basis, and with any other requirements under the Operating License, to determine if there are any unreviewed safety |* questions as defined in 10CFR50.59.
- Perform reviews of Problem Reports and 10CFR21 reviews in accordance with corporate requirements.
- Operate the Action Item Tracking System (AIT) for AEPSC internal commitment tracking.
- Coordinate design changes for the Cook Nuclear Plant, acting as a |* focal point within AEPSC. This program primarily involves project management responsibilities for scheduling and implementing Request for Changes (RFCs), and includes extensive interfacing |* with engineering, design, construction, and Cook Nuclear Plant.
- Provide working-level coordination with the Institute of Nuclear |* Power Operations (INPO). This effort includes providing AEPSC access to INPO resources, such as NUCLEAR NETWORK and Nuclear |* Plant Reliability Data System (NPRDS), and effectively integrating AEPSC and Cook Nuclear Plant efforts towards utilizing INPO recommendations contained in operating experience reports to improve Cook Nuclear Plant performance.

- Coordinate daily communication with the Cook Nuclear Plant, provide AEPSC management with a daily plant status report, and make presentations to senior management at regularly scheduled construction staff meetings.
- Process incoming vendor information.
- Coordinate operations within AEPSC that support the Cook Nuclear Plant Facility Data Base (FDB).
- Contribute to the annual FSAR updates through reviews of Licensee Event Reports, design changes and the Annual Operating Report. |*
- Radiological, emergency and security planning.
- Corporate support of the Cook Nuclear Plant's radiation protection and health physics program, technical service and advice on the radiological aspects of design changes, modifications or capital improvements, the ALARA program, the radiation monitoring system, the environmental radiological monitoring and sampling program, dose and shielding analysis, radiochemistry review, and meteorological monitoring.
- Cook Nuclear Plant and corporate emergency planning, including procedure development, exercise scheduling, facility procurement and maintenance, and the liaison with off-site emergency planning groups, such as FEMA and the Michigan State Police. |*
- Review federal codes and regulations as they relate to the development, implementation, revision and distribution of the Modified Amended Security Plan (MASP).
- Interface with the plant's security department providing support for the security plan, reviewing security facilities, maintaining security document files, and developing the employee fitness for duty/background screening program.
- Provide Nuclear General Employee Training (NGET) for AEPSC personnel.
- Coordinate the development of training for AEPSC personnel who support the operation and maintenance of Cook Nuclear Plant, ensuring a unified training program meeting annual goals and objectives. |
- Participate on ALARA subcommittees.

- Prepare responses to the NRC on radiological, emergency planning and security issues.
- Serve as technical advisors on plant audits.
- Remain cognizant of current decommissioning practices and developments.

AEPSC Engineering and Design

The AEPSC Senior Vice President and Chief Engineer, reporting to the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for certain engineering and design functions through the AEPSC Assistant Vice President - Civil Engineering, the AEPSC Assistant Vice President - Design and the AEPSC Assistant Vice President - Nuclear Engineering.

The AEPSC Electrical Engineering, and System Planning Departments provide periodic, technical assistance for the Cook Nuclear Plant. The administrative and quality assurance controls for this assistance, along with that provided by the Civil Engineering Department, are controlled through documented interface agreements with the AEPSC Nuclear Engineering Department.

Civil Engineering Department

The AEPSC Assistant Vice President - Civil Engineering, reporting to the AEPSC Senior Vice President and Chief Engineer, is responsible for the Civil Engineering Department.

The Civil Engineering Department (CED) is responsible for the technical aspects for the following:

- Make recommendations and assist in the formulation of policies and practices relating to the design and engineering of office and service buildings, miscellaneous structures and material handling equipment, and provide the general supervision of the engineering of such facilities, structures and equipment.

- Review the activities of equipment, facilities, buildings and other structures at the Cook Nuclear Plant and approve, as required, all design changes and modifications, including the preparation of specifications; and procurement of, and modifications to, equipment.
- Provide training and development programs necessary for personnel of the department, (including the company's safety and health program), which are consistent with the written policies of AEPSC.
- Prepare design criteria, engineering standards, conceptual layouts, studies and procedures in conjunction with equipment, facilities, buildings and other structures at the Cook Nuclear Plant.
- Identify critical engineering and design input, and ensure that appropriate analysis and reviews are conducted. |*
- Prepare, review and approve specifications, sketches, drawings, design verifications and calculations, as required.
- Provide input for special studies and reports which may be requested by other organizations or governmental agencies such as the NRC.
- Initiate and/or review, approve and control laboratory and field investigations and feasibility studies.
- Prepare and review improvement requisitions for capital and lease expenditures.
- Review and evaluate proposals and make recommendations for the award of purchase orders and contracts.
- Prepare and administer equipment, labor and service contracts.
- Provide technical guidance, when requested, in support of |*
  activities at the Cook Nuclear Plant under the department's |*
  responsibilities.
- Prepare and approve design changes pertaining to Cook Nuclear Plant in accordance with the GPs.
- Arrange for outside engineering and consulting assistance, as required. |*

- Arbitrate disputes which arise between construction forces and outside suppliers of materials and services.
- Coordinate consultant's reports with other interfacing engineering organizations.
- Perform shop and field inspections on equipment being fabricated, or installed, which is within the scope of the department's responsibility. |*
- Approve invoices for outside services.
- Provide field services to the Cook Nuclear Plant, including the assigning of personnel, as are required, during construction, normal or emergency outages, or as requested. |*
- Assist in the planning and execution of maintenance work on equipment, facilities, buildings and other structures.
- Supervise maintenance and repairs of all masonry and concrete work at Cook Nuclear Plant, including supplying trained inspection personnel.
- Direct testing of materials used in concrete and testing of soils to be used in work at the Cook Nuclear Plant.
- Review and recommend concrete mix formulations for all new construction.
- Prepare site studies.
- Implement a corrective action system, with regard to all activities of the department affecting quality of safety-related items, that will control and document all items, services or activities which do not conform to requirements.
- Direct the review of, and response to, assigned corrective actions.
- Assist in the preparation of applications for federal, state and local permits relative to installations being made which require such permits.

- Conduct periodic management reviews of the activities of the department to ensure compliance with the objectives of the QA Program, and external technical surveillance, as necessary, of consultants, outside organizations and vendors over which the department is cognizant.
- Establish and maintain a permanent file for QA records.

Design Department

The AEPSC Assistant Vice President - Design, reporting to the AEPSC Senior Vice President and Chief Engineer, is responsible for the Design Department. |*

The Design Department is responsible for the following:

- Develop, review and approve designs and drawings for mechanical, electrical and structural systems, equipment and facilities of the Cook Nuclear Plant.
- Initiate, develop, approve and maintain design procedures, specifications, standards, criteria and guidelines.
- Perform required calculations and analyses, including pipe stress, pipe support design, cable sizing, conduit and cable tray support and structural steel and concrete. |*
- Initiate and develop design changes in the areas of responsibility of the Design Department.
- Provide NRC responses, as required. |*
- Assist field personnel in the resolution of problems stemming from the installation of design changes, or from as-found plant conditions, including assigning design personnel to the field. |*
- Participate, as assigned, on the NSDRC and NSDRC subcommittees, and participate in matters covered in the committee's charter. |
- Participate in the evaluation and remedy of any situation requiring activation of the emergency response organization, including resource allocation. |*
- Formulate, administer, and implement policies and practices relating to the design of the Cook Nuclear Plant.

- Direct the development, maintenance, procedural review and implementation by which the Design Department adheres to the QA Program elements as established by AEPSC General Procedures.
- Conduct functions of the department so as to be in conformance with the operating licenses of the Cook Nuclear Plant.
- Investigate and evaluate problems.
- Coordinate special projects and studies, as required.
- Establish and maintain files of design documents for record purposes.
- Coordinate the development and maintenance of the computerized Design Drawing Control (DDC) and the Vendor Drawing Control (VDC) programs which include coordinating the programs with interfacing divisions/departments.
- Control the issuance and distribution of drawings for the Cook Nuclear Plant, including monitoring of the Aperture Card Microfilm Program.
- Supervise and control the work of consultants, architect/engineers and outside design agencies supplying services to AEPSC in their discipline and process notification of defects in accordance with company requirements. Also perform detailed reviews of design work submitted by outside agencies. |*
- Provide input to the list of major approved materials, and maintain current specifications used within the group's scope of responsibility. |*
- Provide engineering and design support to NOD.
- Review and update applicable sections of Cook Nuclear Plant Updated FSAR as assigned. |
- Participate on committees that review nuclear activities as members, when assigned. |*
- Coordinate and resolve design comments made by interfacing departments/divisions.
- Prepare, review, approve and administer design specifications and purchase documents for design services and/or materials.

Nuclear Engineering Department

The AEPSC Assistant Vice President - Nuclear Engineering, reporting to the AEPSC Senior Vice President and Chief Engineer, is responsible for the Nuclear Engineering Department.

The Nuclear Engineering Department (NED) is responsible for the following:

- Provide planning and engineering, in conjunction with other specialists, sections, and divisions, of the electrical facilities inside Cook Nuclear Plant up to the high voltage (HV) bushings of the main generator transformers and mechanical facilities inside Cook Nuclear Plant including:

    * determination of general layout and design;
    * selection of equipment;
    * preparation of one-line and flow diagrams; and,
    * coordination of inside and outside plant facilities.

- Provide engineering and design of all controls for operation and protection of nuclear steam supply, steam generator, turbine generator, auxiliary equipment and general plant protection, including checking and approving elementary, one-line, and flow drawings.
- Interface with other organizations to ensure that all purchased equipment conforms to accepted standards and fulfills the desired function.
- Closely follow manufacturer's engineering and design processes to assure provision of adequate and reliable equipment upon which depend the safety, reliability, economy, and performance of the unit and plant. |*
- Prepare cost estimates and improvement requisitions for plant facilities, including review of improvement requisitions and cost estimates prepared by others.

- Prepare and/or approve specifications and purchase requisitions, and perform drawing review of equipment, as appropriate. |*
- Review and approve procedures, correspondence, Requests for Design Changes or modifications, as appropriate. |*
- Obtain, review and perform engineering evaluations, including environmental equipment qualification (EQ). |*
- Perform calculations for proper application of equipment.
- Perform and evaluate economic studies, investigations, analyses and reports for facilities pertaining to the design, operation and maintenance of the Cook Nuclear Plant.
- Assist field personnel in installation, start-up, and subsequent locating of problems in equipment, and in determining proper operation of equipment during normal, or after, emergency operations. |*
- Maintain a constant awareness for improvements and more reliable design of equipment and facilities, maintenance and operating methods or procedures.
- Maintain a constant awareness of activities to ensure compliance with all applicable policies and procedures, initiating, when required, training or retraining programs. |*
- Participate, as assigned, on the NSDRC and NSDRC subcommittees, and participate in matters covered in the committee's charter. |*
- Provide responses to NRC correspondence, as required. |*
- Participate in the evaluation and remedy of any situation requiring activation of the emergency response organization.
- Provide technical engineering support in areas of operation and maintenance, including: the Inservice Inspection (ISI) Program; the QA Program; the Fire Protection QA Program; the AEP ALARA Program covering radiation protection; and, the corporate and plant Industrial Safety program.
- Provide engineering support to the AEPSC NOD.
- Provide technical direction and assistance to the AEPSC Design Division in the layout and arrangement of equipment piping, systems, controls, etc., for the development of drawings.

- Initiate and develop design changes in areas of responsibility of the NED.
- Develop System Descriptions.
- Provide support personnel for the emergency response organization.
- Provide analytical support in engineering disciplines (e.g., heat transfer, thermodynamics, fluid dynamics).
- Provide engineering evaluations for PRs, LERs, INPO SOERs, and NRC Bulletins.
- Participate, as assigned, on the AEPSC Problem Assessment Group (PAG).

Project Management and Construction Department

The AEPSC Vice President - Project Management and Construction, reporting to the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for the Project Management and Construction Department.

Reporting to the AEPSC Vice President - Project Management and Construction are the following:

- Site Construction Manager, reporting administratively to the AEPSC Vice President - Project Management and Construction, and functionally to the Cook Nuclear Plant, Plant Manager. |* |* |

The Project Management and Construction Department is responsible for the following:

- Administer and implement construction job orders issued by the Cook Nuclear Plant organization for major modifications, replacement and maintenance work with outside contractors. |* |*
- Administer and monitor contractor's industrial safety programs and performance. |*
- Administer human resources' functions for site construction organization. |*

- Manage construction labor relations with the International Building and Construction Trades Unions.
- Scope, bid, recommend awards and administer construction labor and services contracts.
- Plan, organize and control major construction projects, as assigned by the AEPSC Senior Executive Vice President - Engineering and Construction.
- Maintain cognizance on matters pertaining to the Cook Nuclear Plant and corporate emergency response organization.
- Prepare of construction labor estimates.
- Provide constructability guidance when requested in support of engineering and design changes.
- Participate on the Nuclear Safety Design Review Committee.

<u>Purchasing and Stores Department</u> (not charted)

The AEPSC Executive Vice President - Operations, reporting to the AEPSC President and Chief Executive Officer, is responsible for the Purchasing and Stores Department through the AEPSC Vice President - Purchasing and Materials Management.

The Purchasing and Stores Department is responsible for the following:

- Procurement of "N" items from only qualified and approved suppliers.
- Provide supervision to Cook Nuclear Plant Purchasing Section.
- Provide ordering and stocking descriptions of "N" items and include these descriptions in the Cook Nuclear Plant inventory catalog, including necessary communications with suppliers, cognizant engineers, the Cook Nuclear Plant Stores Supervisor and other appropriate personnel.

- Coordinate procurement activities with AEPSC Nuclear Operations, AEPSC engineering and design divisions/departments, Cook Plant Site Purchasing Section, the AEPSC QAD and Cook Nuclear Plant personnel.
- Prepare and issue requests for quotations, contracts, service orders and purchase orders for "N" items.
- Establish a system to implement corrective action as described in the AEPSC General Procedures for the Cook Nuclear Plant.
- Establish a system of document keeping and transmittal. |*
- Establish a system of document control for controlled procedures, instructions, and purchasing documents for "N" items.
- The maintenance and control of selected standard procurement document phrases as identified by the Director - Quality Assurance, or designee.
- Conduct training sessions involving purchasing personnel and others on an annual basis, or more frequently, as required, |* and ascertain that training sessions include complete responsibilities associated with the purchase of safety-related items.

1.7.1.2.6 <u>Organization (Cook Nuclear Plant)</u>

The Plant Manager reports functionally and administratively to the AEPSC Vice President - Nuclear Operations (Manager of Nuclear Operations) and is responsible for the Cook Nuclear Plant activities.
Reporting to the Plant Manager are the following (Figure 1.7-5):

- Assistant Plant Manager - Production
- Assistant Plant Manager - Technical Support
- Assistant Plant Manager - Projects
- Licensing Activity Coordinator
- Safety and Assessment Superintendent

- Radiation Protection Manager (reports functionally to the Plant Manager)
- Nuclear Security Manager (reports functionally to the Plant Manager)

The Cook Nuclear Plant organization, under the Plant Manager, is responsible for the following: |*

- Ensure the safety of all facility employees and the general public relative to general plant safety, as well as radiological safety, by maintaining strict compliance with plant Technical Specifications, procedures and instructions. |*
- Recommend facility engineering modification and initiate and approve plant improvement requisitions.
- Ensure that work practices in all plant departments are consistent with regulatory standards, safety, approved procedures, and plant Technical Specifications.
- Provide membership, as required, on the PNSRC.
- Maintain close working relationships with the NRC, as well as local, state, and federal government regulatory officials regarding conditions which could affect, or are affected, by Cook Nuclear Plant activities. |* |* |*
- Set up plant load schedules and arrange for equipment outages.
- Develop and efficiently implement all site centralized training activities.
- Administer the centralized facility training complex, simulator, and programs ensuring that program development is consistent with the systematic approach to training, maintain INPO accreditations, regulatory and corporate requirements.
- Ensure that human resource activities include employee support programs (i.e., fitness for duty) consistent with INPO/NUMARC guidelines, company policies, and regulatory requirements and standards.
- Administer the NRC approved physical Security Program in compliance with regulatory standards, Modified Amended Security Plan (MASP), and company policy.

- Supervise, plan, and direct the activities related to the maintenance and installation of all Cook Nuclear Plant equipment, structures, grounds, and yards.
- Prepare and maintain records and reports pertinent to equipment maintenance and regulatory agency requirements.
- Administer contracts and schedule outside contractors' work forces.
- Enforce and coordinate Cook Nuclear Plant regulations, procedures, policies, and objectives to assure safety, efficiency, and continuity in the operation of the Cook Nuclear Plant within the limits of the operating license and the Technical Specifications and formulation of related policies and procedures.
- Plan, schedule, and direct activities relating to the operation of the Cook Nuclear Plant and associated switchyards; cooperate in planning and scheduling of work and procedures for refueling and maintenance of the Cook Nuclear Plant; and direct and coordinate fuel loading operations. |*
- Review reports and records, direct general inspection of operating conditions of plant equipment, and investigate any abnormal conditions, making recommendations for repairs. Establish and administer equipment clearance procedures consistent with company, plant, and radiation protection standards; authorize and arrange for equipment outages to meet normal or emergency conditions. Provide the shift operating crews with appropriate procedures and instructions to assist them in operating the Cook Nuclear Plant safely and efficiently. |*
- Approve operator training programs administered by the Cook Nuclear Plant Training Department designed to provide operating personnel with the knowledge and skill required for safe operation of the facility, and for obtaining and holding NRC operator licenses. Coordinate training programs in plant safety and emergency procedures for Cook Nuclear Plant Operating Department personnel to ensure that each shift group will function properly in the event of injury of personnel, fire, nuclear incident, or civil disorder. |*

- Advance planning and overall conduct of scheduled and forced outages, including the scheduling and coordination of all plant activities associated with refueling, preventive maintenance, corrective maintenance, equipment overhaul, Technical Specification surveillance, and design change installations.
- Coordinate all Cook Nuclear Plant activities associated with the initiation, review, approval, engineering, design, production, examination, inspection, test, turnover, and close out of design changes.
- Develop and implement an effective Quality Control (QC) Program. This encompasses, but is not limited to, the planning and directing of quality control activities to assure that industry codes, NRC regulations, and company instructions and policies regarding quality control for Cook Nuclear Plant are implemented, qualified personnel perform the work, and that these activities are properly documented. |*
- Prepare reports of reportable events which are mandated by the NRC and the Technical Specifications.
- Direct the activities of contractor QC/nondestructive examination (NDE) personnel assigned to the Safety and Assessment Department and provide inspections of work performed.
- Prepare statistical reports utilized in NRC Appraisal Meetings and Enforcement Conference.
- Coordinate the efforts of outside agencies, such as American Nuclear Insurers (ANI), INPO, and third-party inspector programs. |*
- Maintain knowledge of developments and changes in NRC requirements, industry standards and codes, regulatory compliance activities, and quality control disciplines and techniques.
- Stop plant operation in the event that conditions are found which are in violation of the Technical Specifications or adverse to quality.
- Maintain and renew accreditation of training programs.

- Qualification and certification of I&M personnel performing inspections or tests of major modifications and non-routine maintenance to the requirements of Regulatory Guide 1.58 and ANSI N45.2.6, except as noted in Appendix B hereto, item 9. |*
- Qualification and certification of I&M NDE personnel to the requirements of the AEP NDE Manual.
- Qualification of I&M personnel performing inspection of normal operating activities to ANSI N18.1.
- Proper certification of contractor inspection, test and examination personnel in accordance with Regulatory Guide 1.58, ANSI N45.2.6, ASME B&PV Code and/or SNT-TC-1A, as applicable.
- Perform peer inspections of work completed by I&M personnel by independent persons qualified to ANSI N18.7.
- Conduct of the Inservice Inspection (ISI) Program.
- Procurement, receiving, quality control receipt inspection, storage, handling, issue, stock level maintenance, and overall control of stores items.
- Provide material service and support in accordance with policies and procedures required by AEP Purchasing and Stores, AEPSC QA, and the NRC, which are administered and enforced in a total effort to ensure safety and plant reliability.
- Plan and direct engineering and technical studies, nuclear fuel management, equipment performance, instrument and control maintenance, on-site computer systems, Shift Technical Advisors, and emergency planning for the Cook Nuclear Plant. These activities support daily on-site operations in a safe, reliable, and efficient manner in accordance with all corporate policies, applicable laws, regulations, licenses, and Technical Specification requirements.
- Implement station performance testing and monitor programs to ensure optimum plant efficiency.
- Direct programs related to on-site fuel management and reactor core physics testing, and ensure satisfactory completion. |*
- Establish testing and preventive maintenance programs related to station instrumentation, electrical systems, and computers.

- Recommend alternatives to Cook Nuclear Plant operation, technical or emergency procedures, and design of equipment to improve safety of operations and overall plant efficiency.
- Implement the corporate Emergency Plan as it pertains to the Cook Nuclear Plant site.
- Provide technical and engineering services in the fields of chemistry, radiation protection, ALARA, and environmental in support of the safe operation of the plant and the health and safety of the employees and the public.
- Plan and schedule the activities of the Technical Physical Science Sections of the Cook Nuclear Plant in support of operations and maintenance.
- Establish chemistry, radiochemistry, and health physics criteria which ensure maximum equipment life, and the protection of the health and safety of the workers and the public.
- Establish sampling and analysis programs which ensure the chemistry, radiochemistry, and health physics criteria are within the established criteria. |*
- Establish and direct investigations, responses, and corrective actions when outside the established criteria.
- Administer and direct the Cook Nuclear Plant's radioactive waste programs, including volume reduction, packaging and shipping.
- Administration of the QA Records Program.
- Maintain the Cook Nuclear Plant Facility Data Base.

1.7.2 QUALITY ASSURANCE PROGRAM

1.7.2.1 SCOPE

Policies that define and establish the Cook Nuclear Plant QA Program are summarized in the individual sections of this document. The program is implemented through procedures and instructions responsive to provisions of the QAPD, and will be carried out for the life of the Cook Nuclear Plant.

Quality assurance controls apply to activities affecting the quality of safety-related structures, systems and components to an extent based on |* the importanceof those structures, systems, components, etc., (items) to safety. Such activities are performed under controlled conditions, including the use of appropriate equipment, environmental conditions, assignment of qualified personnel, and assurance that all applicable prerequisites have been met.

Safety-related items are defined as items:

- Which are associated with the safe shutdown (hot) of the reactor; |* or isolation of the reactor; or maintenance of the integrity of the reactor coolant system pressure boundary.

OR

- Whose failure might cause or increase the severity of a design |* basis accident as described in the Updated FSAR; or lead to a | release of radioactivity in excess of 10CFR100 guidelines.

In general, items are classified as safety-related if they are: Seismic Class I, or Electrical Class 1E; or associated with the Engineered Safety Features Actuation System (ESFAS); or associated with the Reactor Protection System (RPS).

A special QA Program has been implemented for Fire Protection items |* (Section 1.7.19 herein).

The QA Program also includes provision for Radwaste QA in accordance with the requirements of 10CFR71, part H.

QA Program status, scope, adequacy, and compliance with 10CFR50, Appendix B, are regularly reviewed by AEPSC management through reports, meetings, and review of audit results.

The implementation of the QA Program may be accomplished by AEPSC and/or Indiana Michigan Power Company or delegated in whole or in part to other AEP System companies or outside parties. However, AEPSC and/or Indiana Michigan Power Company retain full responsibility for all activities affecting safety-related items. The performance of the delegated organization is evaluated by audit or surveillances on a frequency commensurate with their scope and importance of assigned work.

1.7.2.2 IMPLEMENTATION

1.7.2.2.1

The Chief Executive Officer of AEPSC has stated in a signed, formal "Statement of Policy", that it is the corporate policy to comply with the provisions of applicable codes, standards and regulations pertaining to quality assurance for nuclear power plants as required by the Cook Nuclear Plant operating licenses. |

The statement makes this QAPD and the associated implementing procedures and instructions mandatory, and requires compliance by all responsible organizations and individuals. The statement also identifies the management positions within the companies vested with responsibility and authority for implementing the program and assuring its effectiveness.

1.7.2.2.2

The QA Program at AEPSC and the Cook Nuclear Plant consist of controls exercised by organizations responsible for attaining quality objectives, and by organizations responsible for assurance functions.

The QA Program effectiveness is continually assessed through management review of various reports, NSDRC review of the QA audit program, and |*
shall also be periodically reviewed by independent outside parties as deemed necessary by management.

The QA Program described in this QAPD is intended to apply for the life of the Cook Nuclear Plant.

The QA Program applies to activities affecting the quality of safety-related structures, components, and related consumables during plant operation, maintenance, testing, and all design changes. Safety-related structures, systems and components are identified in the Facility Data Base and other documents which are developed and maintained for the plant.

As deemed necessary by the AEPSC Director - Quality Assurance, or the Plant Manager, applicable portions of the QA Program controls will be applied to nonsafety-related activities associated with the implementation of the QA Program to ensure that commitments are met (e.g., off-site records storage, training services, etc.).

1.7.2.2.3

This QAPD, organized to present the QA Program for the Cook Nuclear Plant in the order of the 18 criteria of 10CFR50, Appendix B, states AEPSC policy for each of the criteria, and describes how the controls pertinent to each are carried out. Any changes made to this QAPD that do not reduce the commitments previously accepted by the NRC must be submitted to the NRC at least annually. Any changes made to this QAPD that do reduce the commitments previously accepted by the NRC must be submitted to the NRC and receive NRC approval prior to implementation. The submittal of the changes described above shall be made in accordance with the requirements of 10CFR50.54.

The program described in this QAPD will not be intentionally changed in any way that would prevent it from meeting the criteria of 10CFR50, Appendix B, and other applicable operating license requirements.

1.7.2.2.4

Documents used for implementing the provisions of this QAPD include the following:

Plant Manager Instructions (PMIs) establish the policy at the plant for compliance with specified criteria, and assign responsibility to the various departments, as required, for implementation. When necessary, Department Head Procedures (DHPs), and in some cases Department Head Instructions (DHIs), have been prepared to describe the detailed activities required to support safe and effective plant operation as per the PMIs.

The PMIs are reviewed by AEPSC QA for concurrence that they will satisfactorily implement regulatory requirements and commitments. They are then reviewed by the PNSRC prior to approval by the Plant Manager.

Safety-related DHPs and DHIs are reviewed by the department head of origination, PNSRC and Plant Manager prior to use.

AEPSC General Procedures (GPs) are utilized to define corporate policies and requirements for quality assurance, and to implement certain corporate QA Program requirements. AEPSC division/department and/or section procedures are also used to implement QA Program requirements.

GPs may also be used to define policies which are nonprocedural in nature.

Nuclear Engineering Procedures (NEPs) establish the policy in AEPSC Nuclear Engineering Department for compliance with the AEPSC GPs and assign responsibility to NED personnel, as required, for implementation. Nuclear Engineering Section Procedures (NESPs) are issued to satisfy a specialized GP requirement.

The NEPs include a procedure for the development and maintenance of procedures. This procedure contain the reviews and approvals necessary to satisfactorily implement regulatory requirements and commitments.

Other procedures used at AEPSC to implement QA Program requirements include Nuclear Design Procedures (NDPs), Nuclear Operations Department Procedures (NODs), and the Civil Engineering Organization and Procedures Manual.

When contractors perform work on-site under their own quality assurance programs, the programs are audited for compliance and consistency with the applicable requirements of the Cook Nuclear Plant's QA Program and the contract, and are approved by AEPSC QA prior to the start of work. Implementation of on-site contractor's QA programs, will be audited to assure that the contractor's programs are effective.

1.7.2.2.5

Provisions of the QA Program for the Cook Nuclear Plant apply to activities affecting the quality of safety-related items. Appendix A to this QAPD lists the Regulatory/Safety Guides and ANSI Standards that identify AEPSC's commitment. Appendix B describes necessary exceptions and clarifications to the requirements of those documents. The scope of the program, and the extent to which its controls are applied, are established as follows: *

a) AEPSC uses the criteria specified in the Cook Nuclear Plant Updated FSAR for identifying structures, systems and components to which the QA Program applies.

b) This identification process results in the Facility Data Base for the Cook Nuclear Plant. This Facility Data Base is controlled by authorized personnel. Facility Data Base items are determined by engineering analysis of the function(s) of plant items in relation to safe operation and shutdown.

c) The extent to which controls specified in the QA Program are applied to Facility Data Base items is determined for each item considering its relative importance to safety. Such determinations are based on data in such documents as the Cook Nuclear Plant Technical Specifications and the Updated FSAR.

1.7.2.2.6

Activities affecting safety-related items are accomplished under controlled conditions. Preparations for such activities include consideration of the following:

a) Assigned personnel are qualified.
b) Work has been planned to applicable engineering and/or Technical Specifications.
c) Specified equipment and/or tools are available.
d) Items are in an acceptable status.
e) Items on which work is to be performed are in the proper condition for the task.
f) Proper instructions/procedures for the work are available for use.
g) Items and facilities that could be damaged by the work have been protected, as required.
h) Provisions have been made for special controls, processes, tests and verification methods.

1.7.2.2.7

Responsibility and authority for planning and implementing indoctrination and training of AEPSC and Cook Nuclear Plant staff personnel are specifically designated, as follows:

a) The training and indoctrination program provides for on-going training and periodic familiarization with the QA Program for the Cook Nuclear Plant.

b) Personnel who perform inspection and examination functions are qualified in accordance with requirements of Regulatory Guide 1.8, ANSI N18.1, Regulatory Guide 1.58, ANSI N45.2.6, the ASME B&PV Code, or SNT-TC-1A, as applicable, and with exceptions as noted in Appendix B hereto. |*

c) AEPSC QAD auditors are qualified in accordance with Regulatory Guide 1.146 and ANSI N45.2.23.

d) Personnel assigned duties such as special cleaning processes, welding, etc., are qualified in accordance with applicable codes, standards, regulatory guides and/or plant procedures.

e) The training, qualification and certification program includes, as applicable, provisions for retraining, reexamination and recertification to ensure that proficiency is maintained.

f) Training, qualification, and certification records including documentation of objectives, waivers/exceptions, attendees and dates of attendance, are maintained at least as long as the personnel involved are performing activities to which the training, qualification and certification is relevant. |*

g) Personnel responsible for performing activities that affect safety-related items are instructed as to the purpose, scope and implementation of the applicable manuals, instructions and procedures.

Management/supervisory personnel receive functional training to the level necessary to plan, coordinate and administer the day-to-day verification activities of the QA Program for which they are responsible.

Training of AEPSC and Cook Nuclear Plant personnel is performed employing the following techniques, as applicable: 1) on the job and

formal training administered by the department or section the individual works for; 2) formal training conducted by qualified instructors from the Cook Nuclear Plant Training Department or other entities (internal and external to the AEP System); and 3) formal, INPO accredited training conducted by the Cook Nuclear Plant Training Department. Records of training sessions for such training are maintained. Where personnel qualifications or certifications are required, these certifications are performed on a scheduled basis (consistent with the appropriate code or standard).

Cook Nuclear Plant employees receive introductory training in quality assurance usually within the first two weeks of employment. In addition, AEPSC personnel receive training prior to being allowed unescorted access to the plant. This training includes management's policy for implementation of the QA Program through Plant Manager and Department Head Instructions and Procedures. These instructions also include a description of the QA Program, the use of instructions and procedures, personnel requirements for procedure compliance and the systems and components controlled by the QA Program.

1.7.3 DESIGN CONTROL

1.7.3.1 SCOPE

Design changes are accomplished in accordance with approved design. Activities to develop such designs are controlled. Depending on the type of design change, these activities include design and field engineering; the performance of physics, seismic, stress, thermal, hydraulic and radiation evaluations; update of the FSAR; review of accident analyses; the development and control of associated computer programs; studies of material compatibility; accessibility for inservice inspection and maintenance; determination of quality standards; and requirement for equipment qualification. The controls apply to preparation and review of design documents, including the correct translation of applicable regulatory requirements and design bases into design, procurement and procedural documents.

1.7.3.2 IMPLEMENTATION

1.7.3.2.1

Design changes are controlled by procedures and instructions and are reviewed as required by 10CFR50.59.

Safety-related design changes are accomplished by one of two separate processes: Minor Modification (MM), or Request for Change (RFC). Those that do not alter the intended function of the item and can be determined by judgement to have a minimal overall impact on the item being modified may be implemented via the MM process. All other safety-related design changes, that are not appropriate for MM processing, are implemented via the RFC process.

In cases where design changes could be deemed to be within the scope of RFCs or MMs solely due to possible insignificant adverse seismic effects, the change may be implemented via the Plant Modification (PM) process.

1.7.3.2.2

RFCs (except those requiring emergency processing), MMs and PMs (having only insignificant seismic effect on safety items) are reviewed and approved prior to implementation, as a minimum, by the cognizant AEPSC engineering section, PNSRC, and Plant Manager.

RFCs and MMs are reviewed to determine their impact on nuclear safety and to determine if the proposed changes involve an unreviewed safety question as defined by 10CFR50.59. If a design change were to involve an unreviewed safety question, it would not be approved for implementation until the required NRC approval was received.

1.7.3.2.3

For RFCs, the Change Control Board established within AEPSC provides an additional review and approval level. The Change Control Board is comprised of members of the Engineering, Design, Nuclear Operations and QA organizations within AEPSC, and is supplemented by other AEPSC organizations or individuals, as required. |* |* |*

The cognizant member of the Change Control Board assigns a lead engineer for each RFC. The lead engineer is responsible for coordinating the RFC activities within AEPSC and maintaining close interface with the Cook Nuclear Plant Project Engineers.

1.7.3.2.4

Proposed RFCs which require emergency processing are originated at the plant, reviewed by the PNSRC, and approved by the Plant Manager. Cook Nuclear Plant management then contacts the AEPSC NOD, and other AEPSC management, as required, describes the change requested, and implements the change only after receiving verbal AEPSC management authorization to proceed. These reviews and approvals are documented and become a part of the RFC Packet. |* |*

1.7.3.2.5

When RFCs or MMs involve design interfaces between internal or external design organizations, or across technical disciplines, these interfaces are controlled. Procedures are used for the review, approval, release, distribution and revision of documents involving design interfaces to ensure that structures, systems and components are compatible geometrically and functionally with processes and the environment. Lines of communication are established for controlling the flow of needed design information across design interfaces, including changes to the |*

information as work progresses. Decisions and problem resolutions involving design interfaces are made by the AEPSC organization having responsibility for engineering direction of the design effort.

1.7.3.2.6

Checks are performed and documented to verify the dimensional accuracy and completeness of design drawings and specifications.

1.7.3.2.7

RFC design document packages are audited by AEPSC QA to assure that the documents have been prepared, verified, reviewed and approved in accordance with company procedures.

1.7.3.2.8

The extent of, and methods for, design verification are documented. The extent of design verification performed is a function of the importance of the item to safety, design complexity, degree of standardization, the state-of-the-art, and similarity with previously proven designs. Methods for design verification include evaluation of the applicability of standardized or previously proven designs, alternate calculations, qualification testing and design reviews. These methods may be used singly or in combination, depending on the needs for the design under consideration. |*

When design verification is done by evaluating standardized or previously proven designs, the applicability of such designs is confirmed. Any differences from the proven design are documented and evaluated for the intended application.

Qualification testing of prototypes, components, or features is used when the ability of an item to perform an essential safety function cannot otherwise be adequately substantiated. This testing is performed before plant equipment installation, where possible, but always before reliance upon the item to perform a safety-related function. Qualification testing is performed under conditions that simulate the most adverse design conditions, considering all relevant operating modes. Test requirements, procedures and results are documented. Results are evaluated to assure that test requirements have been satisfied. Design changes shown to be necessary through testing are made, and any necessary retesting or other verification is performed. Test configurations are clearly documented. |*

Design reviews are performed by multi-organizational or interdisciplinary groups, or by single individuals. Criteria are established to determine when a formal group review is required, and when review by an individual is sufficient.

Procedures require that minor design changes accomplished by the MM process also receive formal design verification. Applicable design verification activities shall be completed prior to declaring the design change, or portion thereof, operational. |*

1.7.3.2.9

Persons representing applicable technical disciplines are assigned to perform design verifications. These persons are qualified by appropriate education or experience, but are not directly responsible for the design. The designer's immediate supervisor may perform the verification, provided that: |*

1) The supervisor is the only technically qualified individual.

or

2) The supervisor has not specified a singular design approach, ruled out design considerations, nor established the design inputs.

and

3) The need is individually documented and approved in advance by the supervisor's management.

and

4) Regularly scheduled QA audits verify conformance to previous items 1 through 3.

Design verification on safety-related design changes shall be completed prior to declaring a design change, or portions thereof, operational. |*

1.7.3.2.10

Cook Nuclear Plant implementation of design changes is accomplished by the Cook Nuclear Plant Project Engineering Department. Material to perform the design change must meet the specifications established for the original system, or as specified by the lead engineer. For those |* design changes where testing after completion is required, the testing documentation is reviewed by the organization performing the test and, when specified, by the AEPSC lead engineer or other cognizant engineer(s). Further, completed design changes are audited by AEPSC QA following installation and testing.

1.7.3.2.11

Changes to design documents, including field changes, are reviewed, approved and controlled in a manner commensurate with that used for the original design. Such changes are evaluated for impact. Information on approved changes is transmitted to all affected organizations.

1.7.3.2.12

Error and deficiencies in, and deviations from, approved design documents are identified and dispositioned in accordance with established design control and/or corrective action procedures. |*

1.7.3.2.13

Established design control procedures provide for: |

1) controlled submission of design changes,
2) engineering evaluation,
3) review for impact on nuclear safety,
4) audit by AEPSC QA,
5) design modification,
6) AEPSC managerial review, and
7) approval and record keeping for the implemented design change.

1.7.4 **PROCUREMENT DOCUMENT CONTROL**

1.7.4.1 **SCOPE**

Procurement documents define the characteristics of item(s) to be procured, identify applicable regulatory and industry codes/standards requirements, and specify supplier QA Program requirements to the extent necessary to assure adequate quality. |*

1.7.4.2 IMPLEMENTATION

1.7.4.2.1

Procurement control is established by instructions and procedures. These documents require that purchase documents be sufficiently detailed to ensure that purchased materials, components and services associated with safety-related structures or systems are: 1) purchased to specification and code requirements equivalent to those of the original equipment or service (except when the Code of Federal Regulations requires upgrading), 2) properly documented to show compliance with the applicable specifications, codes and standards, and 3) purchased from vendors or contractors who have been evaluated and deemed qualified, or by the dedication plan process. |

Procedures establish the review of procurement documents to determine that: quality requirements are correctly stated, inspectable and controllable; there are adequate acceptance criteria; and procurement documents have been prepared, reviewed and approved in accordance with established requirements. |*

The manager of the originating group, with support of the cognizant AEPSC engineering group, is responsible for assuring that applicable requirements are set forth in procurement documents. |

The Cook Nuclear Plant may request assistance of AEPSC cognizant engineers in any procurement activity.

1.7.4.2.2

The Facility Data Base, in conjunction with other sources, is used for equipment safety classification and procurement grade. AEPSC specifications are used to determine requirements, codes or standards that items must fulfill, and define the documentation that must accompany the item to the plant. |

Procurement documents for safety related items and services are reviewed to ensure that: correct classification is made; the requirements are properly stated; and that measures have been, or will be, implemented to assure the requirements are met and adequately provided for.

Purchase requisitions for new safety related items are initiated by the cognizant engineering group which establishes initial requirements.

Replacement/spares are purchased to requirements equivalent to the original unless upgrading is required by Federal Regulations, or deemed necessary by the cognizant engineering group.

1.7.4.2.3

The contents of procurement documents vary according to the item(s) being purchased and its function(s) in the Cook Nuclear Plant. Provisions of this QAPD are considered for application to service contractors also. As applicable, procurement documents include:

a) Scope of work to be performed.

b) Technical requirements, with applicable drawings, specifications, codes and standards identified by title, document number, revision and date, with any required procedures such as special process instructions identified in such a way as to indicate source and need. Imposition of guides/standards on AEPSC/I&M suppliers and subtier suppliers will be on a case-by-case basis depending upon the item or service to be supplied and upon the degree that AEPSC/I&M relies on suppliers to invoke guides/standards. AEPSC/I&M recognizes that certain suppliers have acceptable 10CFR50, Appendix B QA programs, even though, the suppliers are not committed to Regulatory Guides or industry standards (e.g. ANSI N45.2.6). In those cases, in which suppliers are not committed to the same guides/standards as AEPSC/I&M, AEPSC/I&M will assure that (1) the supplier's QA program provides adequate QA controls, regardless of the lack of specific commitment, or (2) controls will be invoked directly by AEPSC/I&M to assure adequate quality of products/services received by suppliers.

c) Regulatory, administrative and reporting requirements.

d) Quality requirements appropriate to the complexity and scope of the work, including necessary tests and inspections.

e) A requirement for a documented QA Program, subject to QA review and written concurrence prior to the start of work.

f) A requirement for the supplier to invoke applicable quality requirements on subtier suppliers.

g) Provisions for access to supplier and subtier suppliers' facilities and records for inspections, surveillances and audits.

h) Identification of documentation to be provided by the supplier, the schedule of submittals and documents requiring AEPSC approval.

1.7.4.2.4

The AEPSC QA Division performs audits of procurement documents to assure that QA Program requirements have been met. These audits are conducted in accordance with AEPSC QA Division procedures.

1.7.4.2.5

Changes to procurement documents are controlled in a manner commensurate with that used for the original documents.

1.7.5 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

1.7.5.1 SCOPE

Activities affecting the quality of safety-related structures, systems and components are accomplished using instructions, procedures and drawings appropriate to the circumstances, including acceptance criteria for determining if an activity has been satisfactorily completed.

1.7.5.2 IMPLEMENTATION

1.7.5.2.1

Instructions and procedures incorporate: 1) a description of the

activity to be accomplished, and 2) appropriate quantitative (such as tolerances and operating limits) and qualitative (such as workmanship and standards) acceptance criteria sufficient to determine that the activity has been satisfactorily accomplished. Hold points for inspection are established when required.

Instructions and procedures pertaining to the specification of, and/or implementation of, the QA Program receive multiple reviews for technical adequacy and inclusion of appropriate quality requirements. Top tier instructions and procedures are reviewed and/or approved by AEPSC QA. Lower tier documents are reviewed and approved, as a minimum, by management/supervisory personnel trained to the level necessary to plan, coordinate and administer those day-to-day verification activities of the QA Program for which they are responsible. |*

Special procedures may be issued for activities which have short-term applicability.

1.7.5.2.2

AEPSC activities relative to the Cook Nuclear Plant are outlined by procedures which provide the controls for the implementation of these activities. AEPSC has two categories of QA Program implementation procedures:

1) General Procedures (GPs) which are applicable to all AEPSC divisions and departments involved with Cook Nuclear Plant.

2) Division/department/section procedures which apply to the specific division, department or section involved.

1.7.5.2.3

Activities at the Cook Nuclear Plant are controlled using plant procedures.

The PMIs have been classified into the following series:

1000 Organization and Responsibilities
2000 Administration - Document Control, Security, Training, Records, Radiation Protection and Fire Protection
3000 Procurement, Receiving, Shipping and Storage
4000 Operations, Fuel Handling, Surveillance Testing
5000 Maintenance, Repair, Modification, Special Processes, EQ and ISI
6000 Technical - Chemistry, Radiological Controls, Performance/Engineering Testing, and Instrument and Control Maintenance and Calibration
7000 Quality Assurance, Quality Control Program and Condition/Problem Reporting

Instructions and procedures identify the regulatory requirements and commitments which pertain to the subject that it will control and establish responsibilities for implementation. Instructions and procedures may either provide the guidance necessary for the development of supple mental instructions and/or procedures to implement their requirements, or provide comprehensive guidance based on the subject matter.

1.7.5.2.4

Cook Nuclear Plant drawings are produced, controlled and distributed under the control of AEPSC and the Cook Nuclear Plant. AEPSC design drawings are produced by, or under the control of, the AEPSC Nuclear |*
Design Group under a set of procedures which direct their development and review. These procedures specify requirements for inclusion of quantitative and qualitative acceptance criteria. Specific drawings are reviewed and approved by the cognizant engineering divisions/department.

AEPSC has stationed an on-site design staff to provide for the revision of certain types of design drawings to reflect as-built conditions.

1.7.5.2.5

Complex Cook Nuclear Plant procedures are designated as "In Hand" procedures. Examples of "In Hand" procedures are those developed for extensive or complex jobs where reliance on memory cannot be trusted. Further, those procedures which describe a sequence which cannot be altered, or require the documentation of data during the course of the procedure, are considered. "In Hand" procedures are designated as such by double asterisks (**) which precede the procedure number on the cover sheet, all pages and attachments of a procedure and the corresponding index. |*

1.7.6 **DOCUMENT CONTROL**

1.7.6.1 **SCOPE**

Documents controlling activities within the scope defined in 1.7.2 herein are issued and changed according to established procedures. Documents such as instructions, procedures and drawings, including changes thereto, are reviewed for adequacy, approved for release by authorized personnel, and are distributed and used at the location where a prescribed activity is performed. |*

Changes to controlled documents are reviewed and approved by the same organizations that performed the original review and approval, or by other qualified, responsible organizations specifically designated in accordance with the procedures governing these documents. Obsolete or superseded documents are controlled to prevent inadvertent use.

1.7.6.2 **IMPLEMENTATION**

1.7.6.2.1

Controls are established for approval, issue and change of documents in the following categories:

a) Design documents (e.g., calculations, specifications, analyses)

b) Drawings and related documents
c) Procurement documents
d) Instructions and procedures
e) Updated Final Safety Analysis Report (UFSAR)
f) Plant Technical Specifications
g) Safeguards documents

1.7.6.2.2

The review, approval, issuance and change of documents are controlled by:

a) Establishment of criteria to ensure that adequate technical and quality requirements are incorporated.

b) Identification of the organization responsible for review, approval, issue and maintenance.

c) Review of changes to documents by the organization that performed the initial review and approval, or by the organization designated in accordance with the procedure governing the review and approval of specific types of documents.

Maintenance, modification and inspection procedures are audited by AEPSC QA for compliance with established inspection requirements.

1.7.6.2.3

Documents are issued and controlled so that:

a) The documents are available prior to commencing work.
b) Obsolete documents are replaced by current documents in a timely manner.

1.7.6.2.4

Master lists, or equivalent controls, are used to identify the current revision of instructions, procedures, specifications and drawings. These control documents are updated and distributed to designated personnel who are responsible for maintaining current copies of the applicable documents. The distribution of controlled documents is performed under procedures requiring receipt acknowledgement and in accordance with established distribution lists. |*

1.7.6.2.5

In the event a drawing is developed on-site to reflect an as-built configuration, the marked-up drawing is maintained in the Master Plant File and all holders of the drawing are issued appropriate notification to inform them the revision they hold is not current, cannot be used and, if required, reference must be made to the Master Plant File drawing.

1.7.6.2.6

Documents prepared for use in training or for interested parties are appropriately marked to indicate that they are for informational use |*
only and cannot be used to operate or maintain the facility, or to |*
conduct activities affecting the quality of safety-related items. At Cook Nuclear Plant, unless a document is identified as "controlled," it |*
is automatically assumed the document is for informational use only.

1.7.7 CONTROL OF PURCHASED ITEMS AND SERVICES

1.7.7.1 SCOPE

Activities that implement approved procurement requests for items and services are controlled to assure conformance with procurement document requirements. Controls include a system of supplier evaluation and selection audits, acceptance of items and documentation upon

delivery, and periodic assessment of supplier performance. Objective evidence of quality that demonstrates conformance with specified procurement document requirements is available to the Cook Nuclear Plant site prior to use of equipment, material, or services.

1.7.7.2 IMPLEMENTATION

1.7.7.2.1

AEPSC qualifies suppliers and distributors by performing a documented evaluation of their capability to provide items or services specified by procurement documents. Items and services designated as safety-related are purchased from suppliers whose QA programs have been accepted in accordance with AEPSC requirements, or from commercial grade suppliers |* through the AEPSC dedication program. Suppliers of other items/services, such as fire protection, records storage, etc., are also evaluated using different criteria for acceptance.

Qualification of such suppliers is determined by the AEPSC QA Division. In the discharge of this responsibility, the AEPSC QA Division may use | information generated by other utilities. The supplier, or distributor, |* must be approved before procurement can be completed. AEPSC is a member of the Nuclear Procurement Issues Council (NUPIC), participates in joint | supplier audits, and shares audit information consistent with NUPIC |* requirements. The supplier, or distributor, must be acceptable, or |* acceptable subject to follow-up, before a procurement can be approved |* and processed. Additional audits will be conducted, as necessary, to meet requirements. Acceptance is not complete until it has been determined that the suppliers' QA program can meet the requirements for | the item(s)/service(s) offered.

1.7.7.2.2

For items that are not unique to a nuclear power plant ("Commercial Grade") where requirements cannot be imposed in a practical manner at the time of procurement, programs for dedication to safety-related |

standards are established and accomplished by the AEPSC cognizant engineer prior to the item being accepted for safety-related use.

1.7.7.2.3

In-process audits of suppliers' activities during fabrication, inspection, testing and shipment of items are performed when deemed necessary, depending upon supplier qualification status, complexity of the item(s) being furnished, the items' importance to safety, and/or previous supplier history. These audits are performed by AEPSC QA. The cognizant engineer and/or responsible Cook Nuclear Plant personnel may also participate, if deemed necessary. |*

1.7.7.2.4

Spare and replacement parts are procured in such a manner that their performance and quality are at least equivalent to those of the parts that will be replaced.

a) Specifications and codes referenced in procurement documents for spare or replacement items are at least equivalent to those for the original items or to properly reviewed and approved revisions.

b) Parts intended as spares or replacement for "off-the-shelf" items, or other items for which quality requirements were not originally specified, are evaluated for performance at least equivalent to the original.

c) Where quality requirements for the original items cannot be determined, requirements and controls are established by engineering evaluation performed by qualified individuals. The evaluation assures there is no adverse effect on interfaces, safety, interchangeability, fit, form, function, or compliance with applicable regulatory or code requirements. Evaluation results are documented.

d) Any additional or modified design criteria, imposed after previous procurement of the item(s), are identified and incorporated.

1.7.7.2.5

Instructions and procedures address requirements for supplier selection and control, as well as procurement document control. The PMI on receipt inspection of safety-related items addresses the program for inspection of incoming items, including a review of the documentation required under the procurement. Receipt inspection personnel are qualified and certified in accordance with the requirements of ANSI N45.2.6. Provisions for receipt inspection apply regardless of where the procurement originates. Additional inspections may apply if required by the procurement document. |* |*

Where items and/or services are safety-related and procurement is accomplished without assistance of AEPSC, supplier selection is limited to those companies identified as being qualified.

1.7.7.2.6

Items received at the site are tagged with a "HOLD" tag and placed in a designated, controlled area until receipt inspected. During receipt inspection, designated material characteristics and attributes are checked, and documentation is checked against the procurement documents. If found acceptable, the "HOLD" tag is removed and replaced with an "ACCEPTED" tag and the item is placed in a designated area of the storeroom. Item traceability to procurement documents and to end use is maintained through recording of "HOLD" and "ACCEPTED" tag numbers on applicable documents. |*

Nonconforming items, or missing or questionable documentation results in items being placed on "HOLD" and maintained in a designated, controlled area of the storeroom. If the nonconformance cannot be cleared, the item is either scrapped, returned to manufacturer, or dispositioned through engineering analysis. |*

1.7.7.2.7

Contractors providing services (on-site) for safety-related components are required to have either a formal quality assurance program and procedures, or they must abide by the Cook Nuclear Plant QA Program and procedures. Prior to their working at the Cook Nuclear Plant, contractor quality assurance programs must be audited and approved by AEPSC QA. Contractor procedures must be reviewed and approved by the originating/sponsoring department supervisor, PNSRC and the Plant Manager. Further, periodic audits of site contractor activities are conducted under the direction of the AEPSC Quality Assurance Superintendent. |*

1.7.7.2.8

To the extent prescribed in specific procurement documents, suppliers furnish quality records; documentary evidence that material and equipment either conforms to requirements or identifies any requirements that have not been met; and descriptions of those nonconformances from the procurement requirements, which have been dispositioned "Use-as-is" or "repair." This evidence is retained at the Plant, or at the Service Corporation.

To the extent prescribed in specific procurement agreements, suppliers are required to maintain additional (backup) documents in their record system.

In some cases, such as with NSSS, suppliers are designated primary record retention responsibility.

1.7.7.2.9

The capability of suppliers to furnish valid certificates is evaluated during procurement document reviews, annual supplier evaluations, and during audits.

1.7.8 IDENTIFICATION AND CONTROL OF ITEMS

1.7.8.1 SCOPE

Items are identified and controlled to prevent their inadvertent use. Identification of items is maintained either on the items, their storage areas or containers, or on records traceable to the items.

1.7.8.2 IMPLEMENTATION

1.7.8.2.1

Controls are established that provide for the identification and control of items (including partially fabricated assemblies).

1.7.8.2.2

Items are identified by physically marking the item or its container, and by maintaining records traceable to the item. The method of identification is such that the quality of the item is not degraded.

1.7.8.2.3

Items are traceable to applicable drawings, specifications, or other pertinent documents to ensure that only correct and acceptable items are used. Verification of traceability is performed and documented prior to release for fabrication, assembly, or installation. |*

1.7.8.2.4

Requirements for the identification by use of heat number, part number, serial number, etc., are included in AEPSC Specifications (DCCs) and/or the procurement document.

1.7.8.2.5

Separate storage is provided for incorrect or defective items that are on hold and material which has been accepted for use. All safety-related items are appropriately tagged or identified (stamping, etc.) to provide easy identification as to the items' usage status. Records are maintained for the issue of items to provide traceability from storage to end use in the Cook Nuclear Plant. |* |*

1.7.8.2.6

When materials are subdivided, appropriate identification numbers are transferred to each section of the material, or traceability is maintained through documentation.

1.7.9 CONTROL OF SPECIAL PROCESSES

1.7.9.1 SCOPE

Special processes are controlled and accomplished by qualified personnel using approved procedures and equipment in accordance with applicable codes, standards, specifications, criteria and other special requirements. |*

1.7.9.2 IMPLEMENTATION

1.7.9.2.1

Processes subject to special process controls are those for which full verification or characterization by direct inspection is impossible or impractical. Such processes include welding, heat treating, chemical cleaning, application of protective coatings, concrete placement and NDE.

1.7.9.2.2

Special process requirements for chemical cleaning, application of protective coatings and concrete placement are set forth in AEPSC Specifications (DCCs) and/or directives prepared by the responsible AEPSC cognizant engineer. These documents are reviewed and approved by other personnel with the necessary technical competence. AEPSC Specifications are audited by the AEPSC QA Division.

Special process requirements for welding, heat treating and NDE are set forth in AEPSC Specifications, the AEP Welding and NDE Manuals and plant procedures. These specifications and manuals are prepared by, or are reviewed and approved by, the AEPSC Cognizant Engineer - Welding and NDE |
Administrator. The administrative controls portion of the NDE Manual is |
audited by AEPSC QA.

Special process procedures, with the exception of welding and heat treating, are prepared by Cook Nuclear Plant personnel with technical knowledge in the discipline involved. These procedures, which are also |*
reviewed by other personnel with the necessary technical competence, are |*
qualified by testing.

Welding is performed in accordance with procedures contained in the AEP Welding Manual, or in the approved contractor's manual. These |
procedures are qualified in accordance with applicable codes, and Procedure Qualification Records are prepared. Weld Procedure Qualifi

cation. Records are reviewed and approved by the AEPSC Cognizant Engineer - Welding. Weld qualification documentation is retained in the AEP Welding Manual, or the approved contractor's manual.

Contractor welding procedures are qualified by the contractor. These procedures and the qualification documentation are reviewed and approved by the AEPSC Cognizant Engineer - Welding. This documentation is retained by the contractor.

1.7.9.2.3

NDE personnel are qualified and certified by a Cook Nuclear Plant NDE Level III who has been qualified and certified by the designated NDE Administrator. Certification is by examination. Personnel qualification is kept current by re-examination at time intervals specified by the AEP NDE Manual, and in accordance with the ASME Code.

Cook Nuclear Plant welders are qualified by the Maintenance Department utilizing the procedures in the AEP Welding Manual. Supervision of Cook Nuclear Plant welder qualifications is performed by the Maintenance Department. Examination of specimens is performed under the supervision of the Safety and Assessment Department in accordance with the AEP Welding Manual covering welder qualification. Cook Nuclear Plant welder qualification records are maintained for each welder by the Maintenance Department. Contractor and craft welders are qualified by the contractor using procedures approved by the AEPSC Cognizant Engineer - Welding in accordance with AEPSC procedures. Contractor and craft welder qualification records are maintained by the contractor.

1.7.9.2.4

QC/NDE Technicians assigned to the Safety and Assessment Department perform nondestructive testing for work performed by Cook Nuclear Plant

and contractor personnel. These individuals are qualified to either SNT-TC-1A, or ANSI N45.2.6, and records of the qualifications/ certifications are maintained at Cook Nuclear Plant. |*

1.7.9.2.5

For special processes that require qualified equipment, such equipment is qualified in accordance with applicable codes, standards and specifications.

1.7.9.2.6

Special process qualifications are reviewed during regularly scheduled QA audits. Qualification records are maintained in accordance with 1.7.17 herein.

1.7.9.2.7

The documentation resulting from welding and nondestructive testing is reviewed by appropriate personnel.

1.7.10 INSPECTION

1.7.10.1 SCOPE

Activities affecting the quality of safety-related structures, systems and components are inspected to verify their conformance with requirements. These inspections are performed by personnel other than those who perform the activity. Inspections are performed by qualified personnel utilizing written procedures which establish prerequisites and provide documentation for evaluating test and inspection results. Direct inspection, process monitoring, or both, are used as necessary. When applicable, hold points are used to ensure that inspections are accomplished at the correct points in the sequence of activities.

1.7.10.2 IMPLEMENTATION

1.7.10.2.1

Inspections are applied to appropriate activities to assure conformance to specified requirements.

Hold points are provided in the sequence of procedures to allow for the inspection, witnessing, examination, measurement, or review necessary to assure that the critical, or irreversible, elements of an activity are |* being performed as required. Note that hold points may not apply to all procedures, but each must be reviewed for this attribute. |*

Hold points specify exactly what is to be done (e.g., type of inspection or examination, etc.), acceptance criteria, or reference to another procedure, etc., for the satisfactory completion of the hold point.

When included in the sequence of a procedure, the activities required by hold points are completed prior to continuing work beyond that point.

Process monitoring is used in whole, or in part, where direct inspection |* alone is impractical or inadequate.

1.7.10.2.2

Training, qualification and certification programs for personnel who perform inspections are established, implemented and documented in accordance with 1.7.2 herein and as described in Appendix B hereto, item 9b, with exceptions as noted therein.

1.7.10.2.3

Inspection requirements are specified in procedures, instructions, drawings, or checklists as applicable. They provide for the following, |* as appropriate:

a) Identification of applicable revisions of required instructions, drawings and specifications. |*

b) Identification of characteristics and activities to be inspected. |*

c) Inspection methods. |*

d) Specification of measuring and test equipment having the necessary accuracy. |*

e) Identification of personnel responsible for performing the inspection. |*

f) Acceptance and rejection criteria. |*

g) Recording of the inspection results and the identification of the inspector. |*

1.7.10.2.4

Inspections are conducted using the following programs:

a) <u>Work Activities Performed by I&M Personnel.</u> Work functions associated with normal operation of the plant, routine maintenance, calibrations, etc., are routinely assigned to plant personnel. I&M personnel who inspect this work are qualified in accordance with Regulatory Guide 1.8 and ANSI N18.1, and are periodically trained in their skill area using INPO "accredited" training. As a result of the qualifications and training which I&M personnel receive, a peer inspection system is used. Peer inspection personnel are independent in that they do not perform, |*
or directly supervise, the work being inspected, but may be from |*
the same work group. Cook Nuclear Plant Safety and Assessment Department personnel qualified in accordance with Regulatory Guide 1.8 and ANSI N18.1 will ensure (through surveillance) that

inspections have been correctly implemented and make routine reports to management.

b) Work Activities Performed by Contractors. Major modifications, non-routine maintenance, and/or other services on safety-related items are generally performed by contractors who are required to comply with the applicable requirements of Regulatory Guide 1.33 and ANSI N45.2. Inspections of these work activities are performed by inspectors qualified and certified in accordance with Regulatory Guide 1.58 and ANSI N45.2.6. A peer inspection program is not used for work activities performed by these personnel. Contractor inspection personnel are required to be qualified and certified in accordance with Regulatory Guide 1.58 and ANSI N45.2.6. I&M Cook Nuclear Plant Quality Control personnel who are also qualified and certified in accordance with Regulatory Guide 1.58 and ANSI N45.2.6 may perform inspections and/or surveillance of these activities. |*

1.7.10.2.5

Inspections associated with the packaging and shipment of radioactive waste and materials are conducted using the following program:

a) NRC Licensed Packagings - Inspections of NRC licensed radioactive material packagings shall be performed by individuals independent from the work being performed. The independent inspectors shall be Indiana Michigan Power personnel, qualified in accordance with Regulatory Guide 1.8 and ANSI N18.1, as a minimum. Additionally, the inspector shall be familiar with the activities being performed. |*

b) Non-NRC Licensed Packagings and Containers - Inspections of non-NRC licensed radioactive material packagings and containers (shipping and/or burial) shall be performed by Indiana Michigan |*

Power personnel, qualified in accordance with Regulatory Guide 1.8 and ANSI N18.1, as a minimum.

c) Transportation Vehicles - Inspection of transportation vehicles being shipped as "exclusive use", shall be performed by Indiana Michigan Power personnel, qualified in accordance with Regulatory Guide 1.8 and ANSI N18.1, as a minimum.

d) Other inspections and Verification - Inspections and verifications of other activities associated with the packaging and shipment of radioactive materials and waste shall be performed by Indiana and |*
Michigan Power personnel, qualified in accordance with Regulatory Guide 1.8 and ANSI N18.1, as a minimum. |*

1.7.10.2.6

Inspections are performed, documented, and the results evaluated by designated personnel in order to ensure that the results substantiate the acceptability of the item or work. Evaluation and review results are documented.

1.7.11 TEST CONTROL

1.7.11.1 SCOPE

Testing is performed in accordance with established programs to demonstrate that structures, systems and components will perform satisfactorily in service. The testing is performed by qualified personnel in accordance with written procedures that incorporate specified requirements and acceptance criteria. Types of tests are:

Scheduled

Surveillance, preventive maintenance, post-design, qualification.

Unscheduled

Pre- and post-maintenance.

Test parameters (including any prerequisites), instrumentation requirements, and environmental conditions are specified in test procedures. Test results are documented and evaluated. |*

1.7.11.2 **IMPLEMENTATION**

1.7.11.2.1

Tests are performed in accordance with programs, procedures and criteria that designate when tests are required and how they are to be performed. Such testing includes the following:

a) Qualification tests, as applicable, to verify design adequacy.

b) Acceptance tests of equipment and components to assure their operation prior to delivery or installation.

c) Post-design tests to assure proper and safe operation of systems and equipment prior to unrestricted operation.

d) Surveillance tests to assure continuing proper and safe operation of systems and equipment. The PMI on surveillance testing controls the periodic testing of equipment and systems to fulfill the surveillance requirements established by the Technical Specifications. Controls have been established to identify uncompleted surveillance testing to assure it is rescheduled for completion to meet Technical Specification frequency requirements.

Data taken during surveillance testing is reviewed by appropriate management personnel to assure that acceptance criteria is fulfilled, or corrective action is taken to correct deficiencies.

e) Maintenance tests after preventive or corrective maintenance.

1.7.11.2.2

Test procedures, as required, provide mandatory hold points for witness or review. |*

1.7.11.2.3

Testing is accomplished after installation, maintenance, or repair, by surveillance test procedures, or performance tests, which must be satisfactorily completed prior to determining the equipment is in an operable status. All data resulting from these tests is retained at the Cook Nuclear Plant after review by appropriate management personnel. |*

1.7.12 CONTROL OF MEASURING AND TEST EQUIPMENT

1.7.12.1 SCOPE

Measuring and testing equipment used in activities affecting the quality of safety-related systems, components and structures are properly identified, controlled, calibrated and adjusted at specified intervals to maintain accuracy within necessary limits.

1.7.12.2 IMPLEMENTATION

1.7.12.2.1

Established procedures and instructions are used for calibration and control of measuring and test equipment utilized in the measurement, inspection and monitoring of structures, systems and components. These procedures and instructions describe calibration techniques and frequencies, and maintenance and control of the equipment.

AEPSC QA periodically assesses the effectiveness of the calibration program via the QA audit program.

1.7.12.2.2

Measuring and test equipment is uniquely identified and is traceable to its calibration source.

1.7.12.2.3

A system has been established for attaching, or affixing labels, to measuring and test equipment to display the date calibrated and the next calibration due date, or a control system is used that identifies to potential users any equipment beyond the calibration due date. |* |*

1.7.12.2.4

Measuring and test equipment is calibrated at specified intervals. These intervals are based on the frequency of use, stability characteristics and other conditions that could adversely affect the required measurement accuracy. Calibration standards are traceable to nationally recognized standards; or where such standards do not exist, provisions are established to document the basis for calibration. |*

The primary standards used to calibrate secondary standards have, except in certain instances, an accuracy of at least four (4) times the required accuracy of the secondary standard. In those cases where the four (4) times accuracy cannot be achieved, the basis for acceptance is documented and is authorized by the responsible manager. The secondary standards have an accuracy that assures equipment being calibrated will be within required tolerances. The basis for acceptance is documented and authorized by the responsible manager.

1.7.12.2.5

Cook Nuclear Plant procedures define the requirements for the control of standards, test equipment and process equipment.

1.7.12.2.6

When measuring and testing equipment used for inspection and testing is found to be outside of required accuracy limits at the time of calibration, evaluations are conducted to determine the validity of the results obtained since the most recent calibration. Retests, or reinspections, are performed on suspect items. The results of evaluations are documented. |* |*

1.7.13 HANDLING, STORAGE, AND SHIPPING

1.7.13.1 SCOPE

Activities with the potential for causing contamination or deterioration, by environmental conditions such as temperature or humidity that could adversely affect the ability of an item to perform its safety-related functions and activities necessary to prevent damage or loss, are identified and controlled. These activities are cleaning, packaging, preserving, handling, shipping and storing. Controls are effected through the use of appropriate procedures and instructions. |*

1.7.13.2 IMPLEMENTATION

1.7.13.2.1

Procedures are used to control the cleaning, handling, storing, packaging, preserving and shipping of materials, components and systems in accordance with designated procurement requirements. These procedures include, but are not limited to, the following functions:

a) Cleaning - to assure that required cleanliness levels are achieved and maintained.

b) Packaging and preservation - to provide adequate protection against damage or deterioration. When necessary, these procedures provide for special environments, such as inert gas atmosphere, specific moisture content levels and temperature levels. |*

c) Handling - to preclude damage or safety hazards.

d) Storing - to minimize the possibility of loss, damage or deterioration of items in storage, including consumables such as chemicals, reagents and lubricants. |*

1.7.13.2.2

Controls have been established for limited shelf life items such as "O" rings, epoxy, lubricants, solvents and chemicals to assure they are correctly identified, stored and controlled to prevent shelf life expired materials from being used in the Cook Nuclear Plant. Controls are established in plant procedures.

1.7.13.2.3

Packaging and shipping requirements are provided to vendors with the AEPSC Specifications (DCCs) which are a part of the purchase order, or are otherwise specified on the procurement order. Controls for receipt inspection, damaged items and special handling requirements at the Cook Nuclear Plant are established by plant procedures. Special controls are provided to assure that stainless steel components and materials are handled with approved lifting slings. |

1.7.13.2.4

Storage and surveillance requirements have been established to assure segregation of storage. Special controls have been implemented for critical, high value, or perishable items. Routine surveillance is conducted on stored material to provide inspection for damage, rotation

of stored pumps and motors, inspection for protection of exposed surfaces and cleanliness of the storage area.

1.7.13.2.5

Special handling procedures have been implemented for the processing of nuclear fuel during refueling outages. These procedures minimize the risk of damage to the new and spent fuel and the possible release of radioactive material when placing the spent fuel into the spent fuel pool.

1.7.14 INSPECTION, TEST, AND OPERATING STATUS

1.7.14.1 SCOPE

Operating status of structures, systems and components is indicated by tagging of valves and switches, or by other specified means, in such a manner as to prevent inadvertent operation. The status of inspections and tests performed on individual items is clearly indicated by markings and/or logging under strict procedural controls to prevent inadvertent bypassing of such inspections and tests.

1.7.14.2 IMPLEMENTATION

1.7.14.2.1

For design change activities, including item fabrication, installation and test, a program exists which specifies the degree of control required for the identification of inspection and test status of structures, systems and components.

Physical identification is used to the extent practical to indicate the status of items requiring inspections, tests, or examinations. Procedures exist which provide for the use of calibration and rejection stickers, tags, stamps and other forms of identification to indicate test and inspection status. The Clearance Permit System uses various tags to identify equipment and system operability status. Another

program establishes a tagging system for lifted leads, etc. For those items requiring calibration, the program provides for physical indication of calibration status by calibration stickers, or a control system is used. |*

1.7.14.2.2

Application and removal of inspection and welding stamps, and of such status indicators as tags, markings, labels, etc., is controlled by plant procedures. |*

The inspection status of materials received at the Cook Nuclear Plant is identified in accordance with established instructions. The status is identified as Hold, Hold for Quality Control Clearance, Reject, or Accept.

The inspection status of work in progress is controlled by the use of hold points in procedures. Plant Quality Control, or departmental ANSI N18.1 qualified personnel (reference 1.7.10.2.4 herein), inspect an activity at various stages and sign off the procedural inspection steps. |*

The status of welding is controlled through the use of a weld data block which identifies the inspection and NDE status of each weld.

1.7.14.2.3

Required surveillance test procedures are defined in PMIs. These instructions provide for documenting bypassed tests and rescheduling of the test. |*

The status of testing after minor maintenance is recorded as part of the Job Order. The status of testing after major maintenance is included as part of the procedure, and includes the performance of functional testing and approval of data by supervisory personnel.

Testing, inspection and other operations important to safety are conducted in accordance with properly reviewed and approved procedures. The PMI for plant procedures requires that procedures be followed as written. Alteration to the sequence of a procedure can only be accomplished by a procedure change which is subject to the same controls as the original review and approval. When an immediate procedure change is required to continue in-process work or testing and the required complete review and approval process cannot be accomplished, an "On The Spot" change is processed in accordance with the PMI on plant procedures. |*

1.7.14.2.4

Nonconforming, inoperable, or malfunctioning structures, systems and components are clearly identified by tags, stickers, stamps, etc., and documented to prevent inadvertent use.

1.7.15 NONCONFORMING ITEMS

1.7.15.1 SCOPE

Materials, parts, or components that do not conform to requirements are controlled in order to prevent their inadvertent use. Nonconforming items are identified, documented, segregated when practical and dispositioned. Affected organizations are notified of nonconformances.

1.7.15.2 IMPLEMENTATION

1.7.15.2.1

Items, services, or activities that are deficient in characteristic, documentation, or procedure, which render the quality unacceptable or indeterminate, are identified as nonconforming and any further use is |*

controlled. Nonconformances are documented and dispositioned, and notification is made to affected organizations. Personnel authorized to disposition, conditionally release and close out nonconformances are designated.

The Job Order System and/or the Condition/Problem Reports (refer to 1.7.16 herein) are used at Cook Nuclear Plant to identify nonconforming items and initiate corrective action for items which are installed or have been released to the Cook Nuclear Plant. Systems, components, or materials which require repair or inspection are controlled under the Job Order System. In addition, the various procedures identified in 1.7.14 herein provide for identification, segregation and documentation of nonconforming items.

1.7.15.2.2

Nonconforming items are identified by marking, tagging, segregating, or by documented administrative controls. Documentation describes the nonconformance, the disposition of the nonconformance and the inspection requirements. It also includes signature approval of the disposition.

Completed Job Orders are reviewed by the supervisor responsible for accomplishing the work, and the supervisor of the department/section |*
that originated the Job Order. The QA Division periodically audits the Job Order System, and on a sample basis, Job Orders.

1.7.15.2.3

Items that have been repaired or reworked are inspected and tested in accordance with the original inspection and test requirements, or |*
alternatives, that have been documented. |*

Items that have the disposition of "repair" or "use-as-is" require documentation justifying acceptability. The changes are recorded to denote the as-built condition.

When required by established procedures, surveillance or operability tests are conducted on an item after rework, repair or replacement.

1.7.15.2.4

Disposition of conditionally released items are closed out before the items are relied upon to perform safety-related functions.

1.7.16 CORRECTIVE ACTION

1.7.16.1 SCOPE

Conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are identified promptly and corrected as soon as practical. |*

For significant conditions adverse to quality, the cause of the condition is determined, corrective action is taken to correct the immediate problem, and preventive action is implemented to prevent recurrence. In these cases, the condition, cause and corrective action taken is documented and reported to appropriate levels of management.

1.7.16.2 IMPLEMENTATION

1.7.16.2.1

Procedures are established that describe the plant and AEPSC corrective action programs. These procedures are reviewed and concurred with by the AEPSC QA Division.

1.7.16.2.2

Condition/Problem Reports provide the mechanism for plant and AEPSC personnel to notify management of conditions adverse to quality. Condition/Problem Reports are also used to report violations to codes, regulations and the Technical Specifications. Investigations of reported conditions adverse to quality are assigned by management. The Condition/Problem Report is used to document the investigation of a problem; and to identify the need for a design change to correct system or equipment deficiencies, or to identify the need for the initiation of Job Orders to correct minor deficiencies. Further, Condition/Problem Reports are used to identify those actions necessary to prevent recurrence of the reported condition.

Significant problems, which are so designated on Condition/Problem Reports, are reviewed by the PNSRC for evaluation of actions taken, or |*
being taken, to correct the deficiency and prevent recurrence. |*

The AEPSC NSDRC is responsible for assuring that independent reviews of violations (as specified in the Technical Specifications) are performed. These violations are considered significant problems which are documented on Condition/Problem Reports. The reviews will provide an independent evaluation of the reported problems and corrective actions.

The AEPSC QA Division periodically audits the corrective action systems for compliance and effectiveness.

1.7.17. QUALITY ASSURANCE RECORDS

1.7.17.1 SCOPE

Records that furnish evidence of activities affecting the quality of safety-related structures, systems and components are maintained. They are accurate, complete, legible and are protected against damage, deterioration, or loss. They are identifiable and retrievable.

1.7.17.2 IMPLEMENTATION

1.7.17.2.1

Documents that furnish evidence of activities affecting the quality of safety-related items are generated and controlled in accordance with the procedure that governs those activities. Upon completion, these documents are considered records. These records include:

a) Results of reviews, inspections, surveillances, tests, audits and material analyses.
b) Qualification of personnel, procedures and equipment.
c) Operation logs.
d) Maintenance and modification procedures and related inspection results.
e) Reportable occurrences.
f) Records required by the plant Technical Specifications.
g) Problem Reports.
h) Other documentation such as drawings, specifications, dedication plans, procurement documents, calibration procedures and reports.
i) Radiographs.

1.7.17.2.2

Instructions and procedures establish the requirements for the identification and preparation of records for systems and equipment under the QA Program, and provide the controls for retention of these records. |*

Criteria for the storage location of quality related records, and a retention schedule for these records, has been established.

File Indexes have been established to provide direction for filing, and to provide for the retrievability of the records.

Controls have been established for limiting access to the Plant Master File to prevent unauthorized entry, unauthorized removal, and for use of the records under emergency conditions. The Accounting Supervisor is responsible for the control and operation of the Plant Master File Room.

1.7.17.2.3

Within AEPSC, each department/division manager is responsible for the identification, collection, maintenance and storage of records generated by their department/division. Procedures ensure the maintenance of records sufficient to furnish objective evidence that activities affecting quality are in compliance with the established QA Program.

1.7.17.2.4

When a document becomes a record, it is designated as permanent, or nonpermanent, and then transmitted to file. Nonpermanent records have specified retention times. Permanent records are maintained for the life of the plant or equipment, as applicable.

1.7.17.2.5

Only authorized personnel may issue corrections or supplements to records.

1.7.17.2.6

Traceability between the record and the item or activity to which it applies is provided.

1.7.17.2.7

Except for records that can only be stored as originals, such as radiographs and some strip charts, or micrographs thereof, records are stored in remote, dual facilities to prevent damage, deterioration, or loss due to natural or unnatural causes. When only the single original can be retained, special fire-rated facilities are used.

1.7.18 AUDITS

1.7.18.1 SCOPE

A comprehensive system of audits is carried out to provide independent evaluation of compliance with, and the effectiveness of, the QA Program, |*
including those elements of the program implemented by suppliers and contractors. Audits are performed in accordance with written procedures or checklists by qualified personnel not having direct responsibility in the areas audited. Audit results are documented and reviewed by |*
management. Follow-up action is taken where indicated.

1.7.18.2 IMPLEMENTATION

1.7.18.2.1 <u>AEPSC QA Division Responsibilities</u>

The basic responsibility for the assessment of the QA Program is vested in the AEPSC QAD. The AEPSC QAD is primarily responsible for ensuring that proper QA programs are established and for verification of their implementation. These responsibilities are discharged in cooperation with the AEPSC and Cook Nuclear Plant management and their staffs. |*

1.7.18.2.2

Internal audits are performed in accordance with established schedules that reflect the status and importance of safety to the activities being performed. All areas where the requirements of 10CFR50, Appendix B apply are audited within a period of one to two years.

1.7.18.2.3

The AEPSC QAD conducts audits to verify the adequacy and implementation of the QA Program at the Cook Nuclear Plant and within the AEP System. QA audit reports are distributed to appropriate Cook Nuclear Plant management and the NSDRC (all audits). |*

1.7.18.2.4

The independent off-site review and audit organization is the AEPSC NSDRC. This committee is composed of AEPSC, I&M and Cook Nuclear Plant management members. An NSDRC Manual has been developed for this committee which contains the NSDRC Charter and procedures. The NSDRC conducts periodic audits of Cook Nuclear Plant operations pursuant to established criteria (Technical Specifications, etc.).

NSDRC audit reports are submitted for review to the NSDRC membership, the Chairman of the NSDRC, and the AEPSC Senior Executive Vice President - Engineering and Construction. Problem Reports provide for the recording of actions taken to correct deficiencies found during these audits. |*

1.7.18.2.5

The Cook Nuclear Plant on-site review group is the PNSRC. This committee reviews plant operations as a routine evaluation and serves to advise the Plant Manager on matters related to nuclear safety. The composition of the committee is defined in the Technical Specifications.

The PNSRC also reviews instructions, procedures, and design changes for safety-related systems prior to approval by the Plant Manager. In addition, this committee serves to conduct investigations of violations to Technical Specifications, and reviews significant Problem Reports to determine if appropriate action has been taken. |*

1.7.18.2.6

Audits of suppliers and contractors are scheduled based on the status of safety importance of the activities being performed, and are initiated early enough to assure effective quality assurance during design, procurement, manufacturing, construction, installation, inspection and testing.

Principal contractors are required to audit their suppliers systematically in accordance with the criteria established within their quality assurance programs.

1.7.18.2.7

Regularly scheduled audits are supplemented by "special audits" when significant changes are made in the QA Program, when it is suspected that quality is in jeopardy, or when an independent assessment of program effectiveness is considered necessary.

1.7.18.2.8

Audits include an objective evaluation of practices, procedures, instructions, activities and items related to quality; and a review of documents and records to confirm that the QA Program is effective and properly implemented. |*

1.7.18.2.9

Audit procedures and the scope, plans, checklists and results of individual audits are documented.

1.7.18.2.10

Personnel selected for auditing assignments have experience, or are |*
given training commensurate with the needs of the audit, and have no |*
direct responsibilities in the areas audited.

1.7.18.2.11

Management of the audited organization identifies and takes appropriate action to correct observed deficiencies and to prevent recurrence. Follow-up is performed by the auditing organization to ensure that the appropriate actions were taken. Such follow-up includes reaudits, when necessary. |*

1.7.18.2.12

The adequacy of the QA Program is regularly assessed by AEPSC management. The following activities constitute formal elements of that assessment:

a) Audit reports, including follow-up on corrective action accomplishment and effectiveness, are distributed to appropriate levels of management.

b) Individuals independent from the QA organization, but knowledgeable in auditing and quality assurance, periodically review the effectiveness of the QA Programs. Conclusions and recommendations are reported to the AEPSC Vice President - Nuclear Operations.

1.7.19 FIRE PROTECTION QA PROGRAM

1.7.19.1 Introduction

The Cook Nuclear Plant Fire Protection QA Program has been developed using the guidance of NRC Branch Technical Position (APCSB) 9.5-1, Appendix A, Section C, "Quality Assurance Program," and NRC clarification "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance," dated June 14, 1977. As such, the Fire Protection QA Program is part of the overall QA Program for the plant. The Fire Protection QA Program encompasses design, procurement, fabrication, construction, surveillance, inspection, operation, maintenance, modification, and audits.

Implementation and assessment of the Fire Protection QA Program is the responsibility of each involved AEPSC and Indiana Michigan Power Company organization.

1.7.19.2 Organization

The Fire Protection QA Program is under the management control of AEPSC. This control consists of:

1) Verifying the effectiveness of the Fire Protection QA Program through review, surveillance, and audits.

2) Directing formulation, implementation, and assessment of the Fire Protection QA Program by procedural controls. |*

3) Assuring the QA program is acceptable to the management responsible for fire protection.

The Plant Manager has delegated responsibility to various Cook Nuclear Plant departments for the following fire protection activities:

a) Maintenance of fire protection systems. |*
b) Testing of fire protection equipment. |*
c) Fire safety inspections. |*
d) Fire fighting procedures. |*
e) Fire drills. |*
f) Emergency remote shutdown procedures. |*
g) Emergency repair procedures (10CFR50, Appendix R). |*

The Fire Protection QA Program at the Cook Nuclear Plant also provides for inspection of fire hazards, explosion hazards, and training of fire brigade and responding fire departments.

The Assistant Shift Supervisor on duty, or designee, is designated as the Fire Brigade Leader and coordinates the fire fighting efforts of shift personnel and the Fire Brigade.

1.7.19.3 Design Control and Procurement Document Control

Quality standards are specified in the design documents such as appropriate fire protection codes and standards, and, as necessary, deviations and changes from these quality standards are controlled. |*

The Cook Nuclear Plant design was reviewed by qualified personnel to ensure inclusion of appropriate fire protection requirements. These reviews include items such as:

1) Verification as to the adequacy of electrical isolation and cable separation criteria.

2) Verification of appropriate requirements for room isolation (sealing penetrations, floors and other fire barriers).

3) Determination for increase in fire loadings.

4) Determination for the need of additional fire detection and suppression equipment.

Procurement of fire protection equipment and related items are subject to the requirements of the fire protection procurement documents. A review of these documents is performed to assure fire protection requirements and quality requirements are correctly stated, verifiable, and controllable, and that there is adequate acceptance and rejection criteria. Procurement documents must be prepared, reviewed, and approved according to QA Program requirements.

Design and procurement document changes, including field changes and design deviations, are controlled by procedure. |*

1.7.19.4 Instructions, Procedures and Drawings

Inspections, tests, administrative controls, fire drills and training that assist in implementing the fire protection program are prescribed by approved instructions or procedures.

Indoctrination and training programs for fire prevention and fire fighting are implemented in accordance with approved procedures. Activities associated with the fire protection systems and fire protection related systems are prescribed and accomplished in accordance with documented instructions, procedures, and drawings. Instructions and procedures for design, installation, inspection, tests, maintenance, modification and administrative controls are reviewed through audits to |* assure that the fire protection program is maintained.

Operation and maintenance information has been provided to the plant in the form of System Descriptions and equipment supplier information.

1.7.19.5 Control of Purchased Items and Services

Measures are established to assure that purchased items and services conform to procurement documents. These measures include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor, inspections at suppliers, or receipt inspection.

Source or receipt inspection is provided, as a minimum, for those items where quality cannot be verified after installation.

1.7.19.6 Inspection

A program for independent inspection of the fire protection activities has been established and implemented.

These inspections are performed by personnel other than those responsible for implementation of the activity. The inspections include:

a) Inspection of installation, maintenance and modification of fire protection systems and equipment.

b) Inspections of penetration seals and fire retardant coating installations to verify the activity is satisfactorily completed in accordance with installation specifications.

c) Inspections of cable routing to verify conformance with design requirements as specified in AEPSC Specifications and/or plant procedures.

d) Inspections to verify that appropriate requirements for fire barriers are satisfied following installation, modification, repair or replacement activities.

e) Measures to assure that inspection personnel are independent from the individuals performing the activity being inspected and are knowledgeable in the design and installation requirements for fire protection. |*

f) Inspection procedures, instructions or checklists for required inspections.

g) Periodic inspections of fire protection systems, emergency breathing and auxiliary equipment.

h) Periodic inspections of materials subject to degradation, such as fire stops, seals and fire retardant coating as required by Technical Specifications or manufacturer's recommendations. |*

1.7.19.7 Test and Test Control

a) Installation testing - Following installation, modification, repair, or replacement, sufficient testing is performed to demonstrate that the fire protection systems and equipment will perform satisfactorily. Written test procedures for installation tests incorporate the requirements and acceptance limits contained in applicable design documents.

b) Periodic testing - Periodic testing occurs to document that fire protection equipment functions in accordance with its design. |*

c) Programs have been established to verify the testing of fire protection systems, and to verify that test personnel are effectively trained. |*

d) Test results are documented, evaluated, and their acceptability determined by a qualified responsible individual or group.

1.7.19.8 Inspection, Test and Operating Status

The inspection, test and operating status for plant Technical Specification fire protection systems are performed as described in 1.7.14 herein.

1.7.19.9 Nonconforming Items

Technical Specification fire protection equipment nonconformances are identified and dispositioned as described in 1.7.15 herein.

1.7.19.10 Corrective Action

The corrective action mechanism described in 1.7.16 herein applies to the Technical Specification fire protection equipment.

1.7.19.11 Records

Records generated to support the fire protection program are controlled as described in 1.7.17 herein.

1.7.19.12 Audits

Audits are conducted and documented to verify compliance with the Fire Protection QA Program as described in 1.7.18 herein.

Audits are periodically performed to verify compliance with the administrative controls and implementation of fire protection quality assurance criteria. The audits are performed in accordance with pre-established written procedures or checklists. Audit results are documented and reviewed by management having responsibility in the area audited. Follow-up action is taken by responsible management to correct the deficiencies revealed by the audit. |*

AMERICAN ELECTRIC POWER SERVICE CORPORATION
SUPPORT ORGANIZATION FOR THE COOK NUCLEAR PLANT

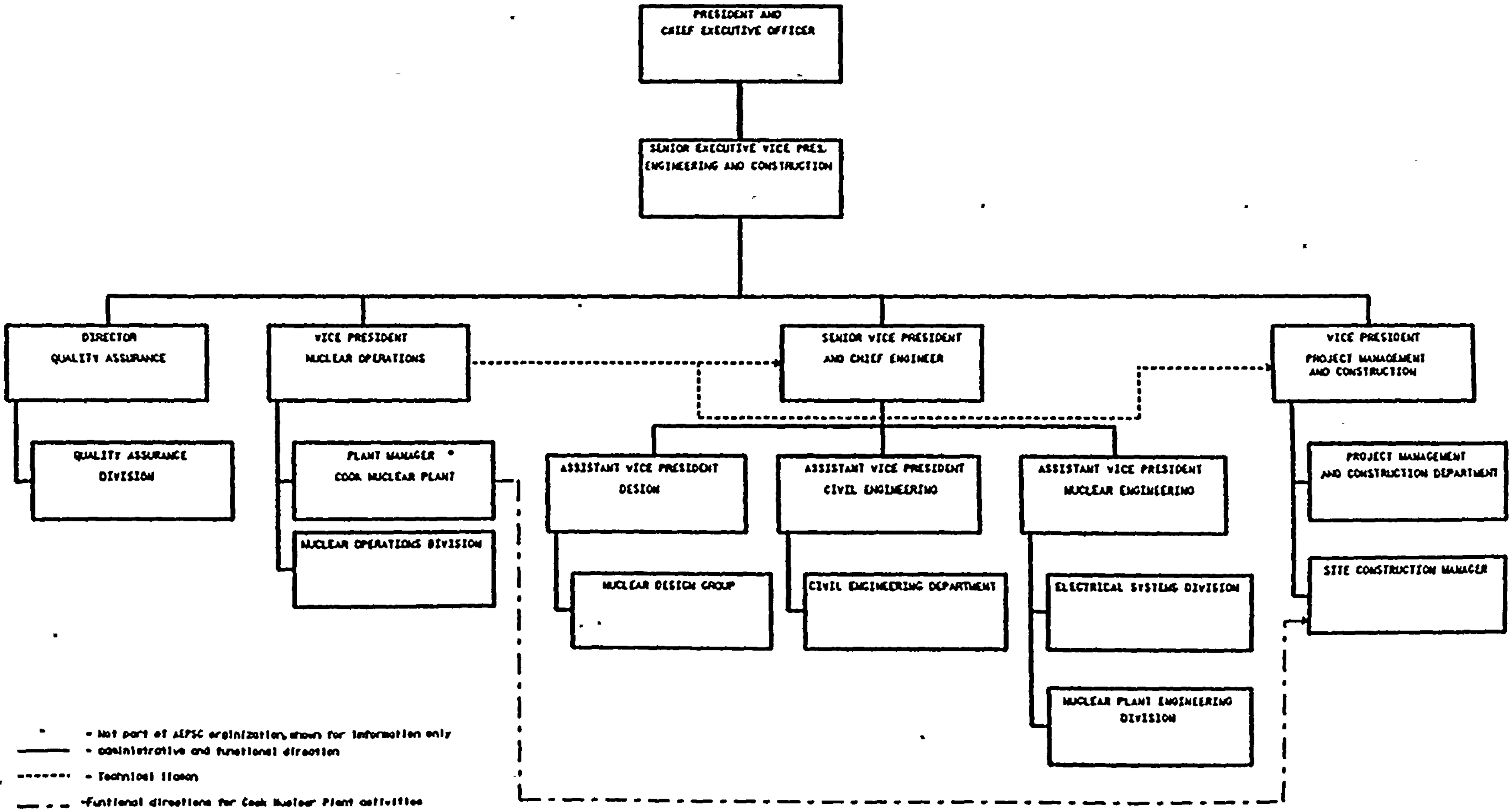
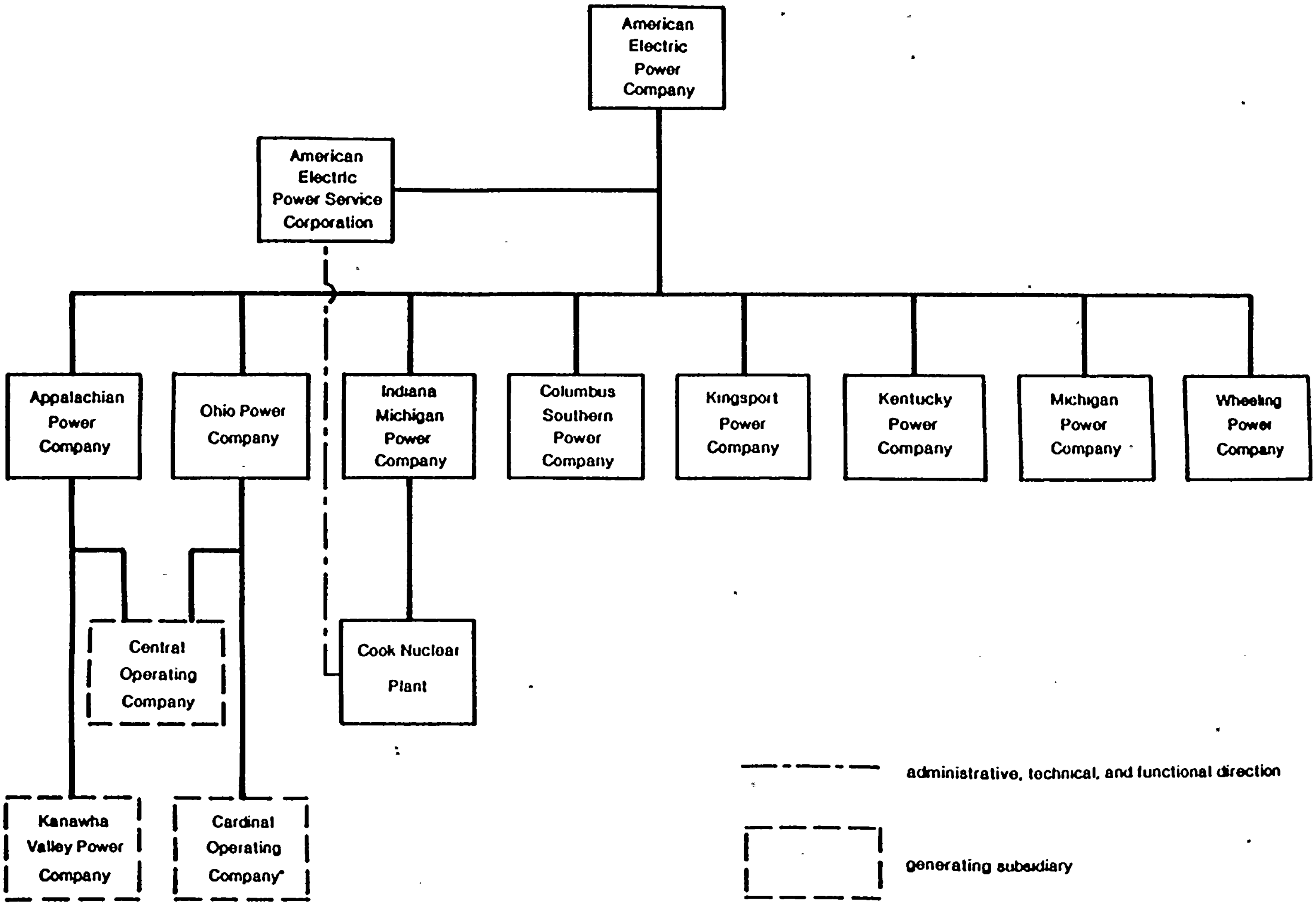


Figure No. 1.7-1

American Electric Power Company

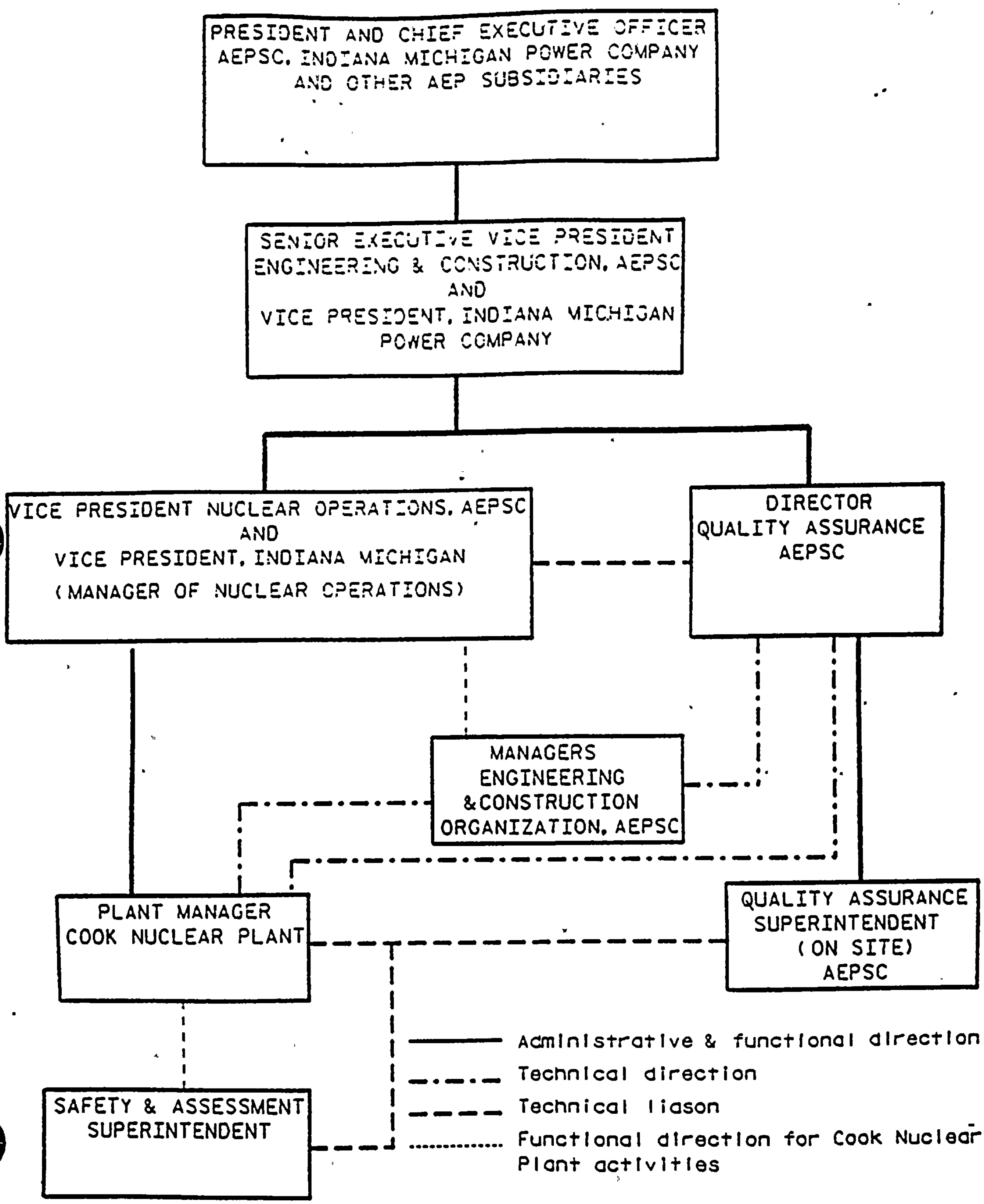


*jointly owned with Buckeye Power, Inc.

Figure No. 1.7-2

Figure No. 1.7-3

ORGANIZATIONAL RELATIONSHIPS WITHIN THE AMERICAN ELECTRIC POWER SYSTEM PERTAINING TO QUALITY ASSURANCE & QUALITY CONTROL SUPPORT OF THE COOK NUCLEAR PLANT



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AEPSC QUALITY ASSURANCE DIVISION ORGANIZATION

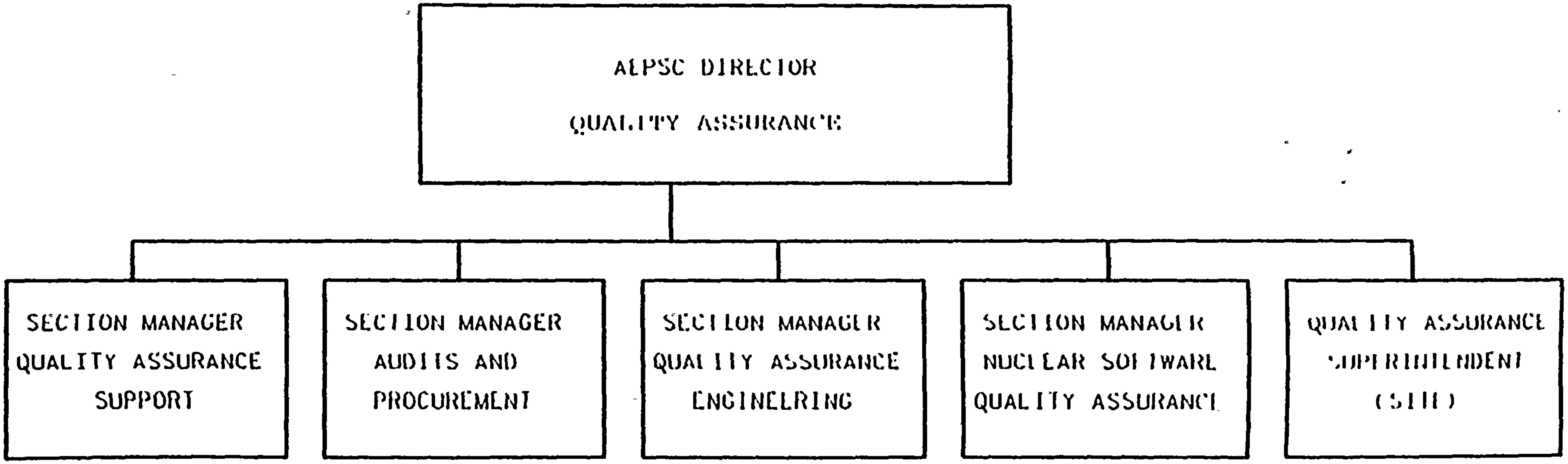

Figure No. 1.7-4

Indiana Michigan Power Company
Organization for the Cook Nuclear Plant

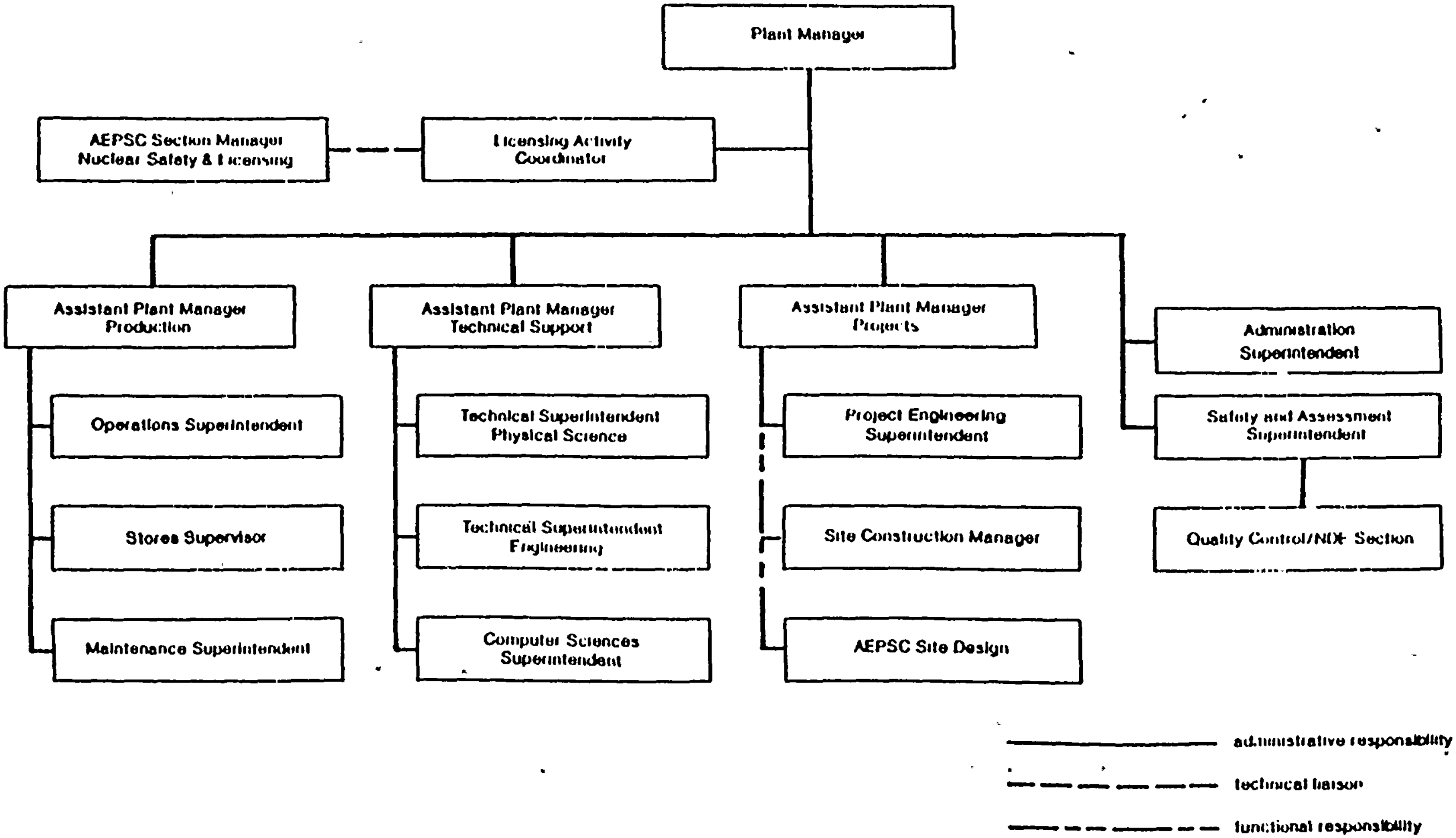

Figure No. 1.7-5

APPENDIX A

REGULATORY AND SAFETY GUIDES/ANSI STANDARDS

1.	Reg. Guide 1.8 (9/75)	-	Personnel Selection and Training
	ANSI N18.1 (1971)	-	Selection and Training of Nuclear Power Plant Personnel
2.	Reg. Guide 1.14 (8/75)	-	Reactor Coolant Pump Flywheel Integrity
3.	Reg. Guide 1.16 (8/75)	-	Reporting of Operating Information, Appendix A - Technical Specifications
4.	Safety Guide 30 (8/72)	-	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment
	ANSI N45.2.4 (1972)	-	Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations
5.	Reg. Guide 1.33 (02/78)	-	Quality Assurance Program Requirements (Operation)
	ANSI N18.7 (1976) (ANS 3.2 1976)	-	Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants
	ANSI N45.2 (1977)	-	Quality Assurance Program Requirements for Nuclear Facilities

6.	Reg. Guide 1.37 (3/73)	-	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants
	ANSI N45.2.1 (1973)	-	Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants
7.	Reg. Guide 1.38 (10/76)	-	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants
	ANSI N45.2.2 (1972)	-	Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants (During the Construction Phase)
8.	Reg. Guide 1.39 (10/76)	-	Housekeeping Requirements for Water-Cooled Nuclear Power Plants
	ANSI N45.2.3 (1973)	-	Housekeeping During the Construction Phase of Nuclear Power Plants
9.	Reg. Guide 1.54 (6/73)	-	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants
	ANSI N101.4 (1972)	-	Quality Assurance for Protective Coatings Applied to Nuclear Facilities
10.	Reg. Guide 1.58 (9/80)	-	Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel |*

ANSI N45.2.6 (1978)	-	Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Power Plants
11. Reg. Guide 1.63 (7/78)	-	Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants
12. Reg. Guide 1.64 (10/73)	-	Quality Assurance Requirements for the Design of Nuclear Power Plants
ANSI N45.2.11 (1974)	-	Quality Assurance Requirements for the Design of Nuclear Power Plants
13. Reg. Guide 1.74 (2/74)	-	Quality Assurance Terms and Definitions
ANSI N45.2.10 (1973)	-	Quality Assurance Terms and Definitions
14. Reg. Guide 1.88 (10/76)	-	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records
ANSI N45.2.9 (1974)	-	Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants
15. Reg. Guide 1.94 (4/76)	-	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants

	ANSI N45.2.5 (1974)	- Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants
16.	Reg. Guide 1.108 (8/77)	- Periodic Testing of Diesel Generator Units used as Onsite Electric Power Systems at Nuclear Power Plants
17.	Reg. Guide 1.123 (7/77)	- Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants
	ANSI N45.2.13 (1976)	- Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants
18.	Reg. Guide 1.144 (1/79)	- Auditing of Quality Assurance Programs for Nuclear Power Plants
	ANSI N45.2.12 (1977)	- Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants
19.	Reg. Guide 1.146 (8/80)	- Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants
	ANSI N45.2.23 (1978)	- Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants

20. ANSI N45.2.8 (1975) - Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants

21. ANSI N45.4 (1972) - Leakage-Rate Testing of Containment Structures for Nuclear Reactors

APPENDIX B

AEPSC/I&M EXCEPTIONS TO OPERATING PHASE STANDARDS AND REGULATORY GUIDES

1. GENERAL

   Requirement

   Certain Regulatory Guides invoke, or imply, Regulatory Guides and standards in addition to the standard each primarily endorses. |*

   Certain ANSI Standards invoke, or imply, additional standards. |*

   Exception/Interpretation

   The AEPSC/I&M commitment refers to the Regulatory Guides and ANSI Standards specifically identified in Appendix A. Additional Regulatory Guides, ANSI Standards and similar documents implied, or referenced, in those specifically identified are not part of this commitment. |*

2. N18.7, General

   Exception/Interpretation

   AEPSC and I&M have established both an on-site and off-site standing committee for independent review activities; together they form the independent review body. |*

   The standard numeric and qualification requirement may not be met by each group individually. Procedures will be established to specify how each group will be involved in review activities. This exception/interpretation is consistent with the plant's Technical Specifications.

2a. Sec. 4.3.1

Requirement

"Personnel assigned responsibility for independent reviews shall be specified in both number and technical disciplines, and shall collectively have the experience and competence required to review problems in the following areas: ...."

Exception/Interpretation

AEPSC Nuclear Safety and Design Review Committee (NSDRC) and Plant Nuclear Safety Review Committee (PNSRC) will not have members specified by number, nor by technical disciplines, and its members may not have the experience and competence required to review problems in all areas listed in this section. This exception/interpretation is consistent with the plant's Technical Specifications. |*

The NSDRC and PNSRC will not specifically include a member qualified in nondestructive testing, but will use qualified technical consultants to perform this and other functions as determined necessary by the respective committee chairman. |*

2b. Sec. 4.3.2.1

Requirement

"When a standing committee is responsible for the independent review program, it shall be composed of no less than five persons of whom no more than a minority are members of the on-site operating organization. Competent alternates are permitted if designated in advance. The use of alternates shall be restricted to legitimate absences of principals."

Exception/Interpretation

See Item 2a.

2c. Sec. 4.3.3.1

Requirement

"... recommendations ... shall be disseminated promptly to appropriate members of management having responsibility in the area reviewed."

Exception/Interpretation

Recommendations made as a result of review will generally be conveyed to the on-site, or off-site, standing committee. Procedures will be maintained specifying how recommendations are to be considered. |*

2d. Sec. 4.3.4

Requirement

"The following subjects shall be reviewed by the independent review body: ...."

Exception/Interpretation

Subjects requiring review will be as specified in the plant Technical Specifications.

2e. Sec. 4.3.4(3)

Requirement

"Changes in the Technical Specifications or License Amendments relating to nuclear safety are to be reviewed by the independent review body prior to implementation, except in those cases where the change is identical to a previously reviewed proposed change."

Exception/Interpretation

Although the usual practice is to meet this requirement, exceptions are made to NSDRC review and approval prior to implementation in rare cases with the permission of the NSDRC Chairman and Secretary. PNSRC review and approval is always done prior to implementation of Technical Specification changes.

2f. Sec. 4.4

Requirement

"The on-site operating organization shall provide, as part of the normal duties of plant supervisory personnel ...."

Exception/Interpretation

Some of the responsibilities of the on-site operating organization described in Section 4.4 may be carried out by the PNSRC and/or NSDRC as described in plant Technical Specifications.

2g. Sec. 5.2.2

Requirement

"Temporary changes, which clearly do not change the intent of the approved procedure, shall as a minimum be approved by two members of the plant staff knowledgeable in the areas affected by the procedures. At least one of these individuals shall be the supervisor in charge of the shift and hold a senior operator's license on the unit affected."

Exception/Interpretation

I&M considers that this requirement applies only to procedures identified in plant Technical Specifications. Temporary changes to these procedures shall be approved as described in plant Technical Specifications.

2h. Sec. 5.2.6

Requirement

"In cases where required documentary evidence is not available, the associated equipment or materials must be considered nonconforming in accordance with Section 5.2.14. Until suitable documentary evidence is available to show the equipment or material is in conformance, affected systems shall be considered to be inoperable and reliance shall not be placed on such systems to fulfill their intended safety functions."

Exception/Interpretation

I&M initiates appropriate corrective action when it is discovered that documentary evidence does not exist for a test or inspection which is a requirement to verify equipment acceptability. This action includes a technical evaluation of the equipment's operability status.

2i. Sec. 5.2.8

Requirement

"A surveillance testing and inspection program ... shall include the establishment of a master surveillance schedule reflecting the status of all planned in-plant surveillance tests and inspections."

Exception/Interpretation

Separate master schedules may exist for different programs, such as |*
ISI, pump and valve testing, and Technical Specification surveillance |*
testing.

2j. Sec. 5.2.13.1

Requirement

"To the extent necessary, procurement documents shall require suppliers to provide a Quality Assurance Program consistent with the pertinent requirements of ANSI N45.2 - 1977."

Exception/Interpretation

To the extent necessary, procurement documents require that the supplier has a documented Quality Assurance Program consistent with the pertinent requirements of 10CFR50, Appendix B; ANSI N45.2; or other nationally recognized codes and standards.

2k. Sec. 5.2.13.2

Requirement

ANSI N18.7 and N45.2.13 specify that where required by code, regulation, or contract, documentary evidence that items conform to procurement requirements shall be available at the nuclear power plant site prior to installation or use of such items.

Exception/Interpretation

The required documentary evidence is available at the site prior to use, but not necessarily prior to installation. This allows installation to proceed while any missing documents are being obtained, but precludes dependence on the item for safety purposes.

21. Sec. 5.2.15

Requirement

"Plant procedures shall be reviewed by an individual knowledgeable in the area affected by the procedure no less frequently than every two years to determine if changes are necessary or desirable."

Exception/Interpretation

Biennial reviews are not performed in that I&M has programmatic control requirements in place that make the biennial review process redundant from a regulatory perspective. These programmatic controls were effected in an effort to ensure that plant instructions and procedures are reviewed for possible revision when pertinent source material is revised, therefore maintaining the procedures current. We believe that this approach, in addition to an annual random sampling |*
of procedures, better addresses the intent of the biennial review process and is more acceptable from both a technical and practical perspective than a static two-year review process.

2m. Sec. 5.2.16

Requirement

Records shall be made, and equipment suitably marked, to indicate |*
calibration status.

Exception/Interpretation

See Item 6b.

2n. Sec. 5.3.5(4)

Requirement

This section requires that where sections of documents such as vendor manuals, operating and maintenance instructions, or drawings are |*
incorporated directly, or by reference into a maintenance procedure, |*
they shall receive the same level of review and approval as operating procedures.

Exception/Interpretation

Such documents are reviewed by appropriately qualified personnel prior to use to ensure that, when used as instructions, they provide proper and adequate information to ensure the required quality of work. Maintenance procedures which reference these documents receive the same level of review and approval as operating procedures.

3. N45.2.1,

3a. Sec. 3

Requirement

N45.2.1 establishes criteria for classifying items into "cleanliness levels," and requires that items be so classified. |*

Exception/Interpretation

Instead of using the cleanliness level classification system of N45.2.1, the required cleanliness for specific items and activities is addressed on a case-by-case basis.

Cleanliness is maintained, consistent with the work being performed, so as to prevent the introduction of foreign material. As a minimum, cleanliness inspections are performed prior to closure of "nuclear" systems and equipment. Such inspections are documented.

3b. Sec. 5

Requirement

"Fitting and tack-welded joints (which will not be immediately sealed by welding) shall be wrapped with polyethylene or other nonhalogenated plastic film until the welds can be completed."

Exception/Interpretation

I&M sometimes uses other nonhalogenated material, compatible with the parent material, since plastic film is subject to damage and does not always provide adequate protection.

4. N45.2.2, General

Requirement

N45.2.2 establishes requirements and criteria for classifying safety related items into protection levels.

Exception/Interpretation

Instead of classifying safety related items into protection levels, controls over the packaging, shipping, handling and storage of such items are established on a case-by-case basis with due regard for the item's complexity, use and sensitivity to damage. Prior to installation or use, the items are inspected and serviced, as |*
necessary, to assure that no damage or deterioration exists which |*
could affect their function.

4a. Sec. 3.9 and Appendix A3.9

Requirement

"The item and the outside of containers shall be marked."
(Further criteria for marking and tagging are given in the Appendix.)

Exception/Interpretation

These requirements were originally written for items packaged and shipped to construction projects. Full compliance is not always

necessary in the case of items shipped to operating plants and may, in some cases, increase the probability of damage to the item. The requirements are implemented to the extent necessary to assure traceability and integrity of the item.

4b. Sec. 5.2.2

Requirement

"Receiving inspections shall be performed in an area equivalent to the level of storage."

Exception/Interpretation

Receiving inspection area environmental controls may be less stringent than storage environmental requirements for an item. However, such inspections are performed in a manner and in an environment which do not endanger the required quality of the item.

4c. Sec. 6.2.4

Requirement

"The use or storage of food, drinks and salt tablet dispensers in any storage area shall not be permitted."

Exception/Interpretation

Packaged food for emergency or extended overtime use may be stored in material stock rooms. The packaging assures that materials are not contaminated. Food will not be "used" in these areas.

4d. Sec. 6.3.4

Requirement

"All items and their containers shall be plainly marked so that they are easily identified without excessive handling or unnecessary opening of crates and boxes."

Exception/Interpretation

See N45.2.2, Section 3.9 (Exception 4b.).

4e. Sec. 6.4.1

Requirement

"Inspections and examinations shall be performed and documented on a periodic basis to assure that the integrity of the item and its container ... is being maintained."

Exception/Interpretation

The requirement implies that all inspections and examinations of items in storage are to be performed on the same schedule. Instead, the inspections and examinations are performed in accordance with material storage procedures which identify the characteristics to be inspected and include the required frequencies. These procedures are based on technical considerations which recognize that inspections and frequencies needed vary from item to item.

5. N45.2.3,

5a. Sec. 2.1

Requirement

Cleanliness requirements for housekeeping activities shall be established on the basis of five zone designations.

Exception/Interpretation

Instead of the five-level zone designation system referenced in ANSI N45.2.3, I&M bases its controls over housekeeping activities on a consideration of what is necessary and appropriate for the activity involved. The controls are effected through procedures or instructions. Factors considered in developing the procedures and instructions include cleanliness control, personnel safety, fire prevention and protection, radiation control and security. The procedures and instructions make use of standard janitorial and work practices to the extent possible. However, in preparing these procedures, consideration is also given to the recommendations of Section 2.1 of ANSI N45.2.3.

6. <u>N45.2.4</u>,

6a. Sec. 2.2

Requirement

Section 2.2 establishes prerequisites which must be met before the installation, inspections and testing of instrumentation and electrical equipment may proceed. These prerequisites include personnel qualification, control of design, conforming and protected materials and availability of specified documents.

Exception/Interpretation

During the operations phase, this requirement is considered to be applicable to modifications and initial start-up of electrical equipment. For routine or periodic inspection and testing, the prerequisite conditions will be achieved, as necessary. |*

6b. Sec. 6.2.1

Requirement

"Items requiring calibration shall be tagged or labeled on completion, indicating date of calibration and identity of person that performed calibration."

Exception/Interpretation

Frequently, physical size and/or location of installed plant instrumentation precludes attachment of calibration labels or tags. Instead, each instrument is uniquely identified and is traceable to its calibration record.

A scheduled calibration program assures that each instrument's calibration is current.

7. <u>N45.2.5</u>,

7a. Sec. 2.5.2

Requirement

"When discrepancies, malfunctions or inaccuracies in inspection and testing equipment are found during calibration, all items inspected

with that equipment since the last previous calibration shall be considered unacceptable until an evaluation has been made by the responsible authority and appropriate action taken." |*

Exception/Interpretation

I&M uses the requirements of N18.7, Section 5.2.16, rather than N45.2.5, Section 2.5.2. The N18.7 requirements are more applicable to an operating plant.

7b. Sec. 5.4

Requirement

"Hand torque wrenches used for inspection shall be controlled and must be calibrated at least weekly and more often if deemed necessary. Impact torque wrenches used for inspection must be calibrated at least twice daily."

Exception/Interpretation

Torque wrenches are controlled as measuring and test equipment in accordance with ANSI N18.7, Section 5.2.16. Calibration intervals are based on use and calibration history rather than as per N45.2.5.

8. N45.2.6, Sec. 1.2

Requirement

"The requirements of this standard apply to personnel who perform inspections, examinations and tests during fabrication prior to or during receipt of items at the construction site, during construction, during preoperational and start-up testing and during operational phases of nuclear power plants."

Exception/Interpretation

Personnel participating in testing who take data or make observations, where special training is not required to perform this function, need not be qualified in accordance with ANSI N45.2.6, but need only be trained to the extent necessary to perform the assigned function. |*

9. Reg. Guide 1.58 - General

Requirement

Qualification of nuclear power plant inspection, examination and testing personnel.

9a. C.2.a(7)

Requirement

Regulatory Guide 1.58 endorses the guidelines of SNT-TC-1A as an acceptable method of training and certifying personnel conducting leak tests.

Exception/Interpretation

I&M takes the position that the "Level" designation guidelines as recommended in SNT-TC-1A, paragraph 4 do not necessarily assure adequate leak test capability. I&M maintains that departmental supervisors are best able to judge whether engineers and other personnel are qualified to direct and/or perform leak tests. Therefore, I&M does not implement the recommended "Level" designation guidelines.

It is I&M's opinion that the training guidelines of SNT-TC-1A, Table I-G, paragraph 5.2 specifically are oriented towards the basic physics involved in leak testing, and further, towards individuals who are not graduate engineers. I&M maintains that it meets the essence of these training guidelines. The preparation of leak test procedures and the conduct of leak tests at Cook Nuclear Plant is under the direct supervision of Performance Engineers who hold engineering degrees from accredited engineering schools. The basic physics of leak testing have been incorporated into the applicable test procedures. The review and approval of the data obtained from leak tests is performed by department supervisors who are also graduate engineers.

I&M does recognize the need to assure that individuals involved in leak tests are fully cognizant of leak test procedural requirements and thoroughly familiar with the test equipment involved. Plant

performance engineers receive routine, informal orientation on testing programs to ensure that these individuals fully understand the requirements of performing a leak test. |*

9b. C5, C6, C7, C8, C10

Exception/Interpretation

I&M takes the position that the classification of inspection, examination and test personnel (inspection personnel) into "Levels" based on the requirements stated in Section 3.0 of ANSI N45.2.6 does not necessarily assure adequate inspection capability. I&M maintains that departmental and first line supervisors are best able to judge the inspection capability of the personnel under their supervision, and that "Level" classification would require an overly burdensome |* administrative work load, could inhibit inspection activities, and |* provides no assurance of inspection capabilities. Therefore, I&M does not implement the "Level" classification concept for inspection, |* examination and test personnel.

The methodology under which inspections, examinations and tests are conducted at the Cook Nuclear Plant requires the involvement of first line supervisors, engineering personnel, departmental supervisors and plant management. In essence, the last seven (7) project functions shown in Table 1 to ANSI N45.2.6 are assigned to supervisory and engineering personnel, and not to personnel of the inspector category. |* These management supervisory and engineering personnel, as a minimum, meet the educational and experience requirements of "Level II and Level III" personnel, as required, to meet the criteria of ANSI 18.1 which exceeds those of ANSI N45.2.6. In I&M's opinion, no useful purpose is served by classification of management, supervisory and engineering personnel into "Levels."

Therefore, I&M takes the following positions relative to regulatory positions C5, 6, 7, 8 and 10 of Regulatory Guide 1.58.

C-5 Based on the discussion in 9b, this position is not applicable to the Cook Nuclear Plant.

C-6 Replacement personnel for Cook Nuclear Plant management, supervisory and engineering positions subject to ANSI 18.1 will meet the educational and experience requirements of ANSI 18.1 and therefore, those of ANSI N45.2.6. |*

Replacement inspection personnel will, as a minimum, meet the educational and experience requirements of ANSI N45.2.6, Section 3.5.1 - "Level I." |*

C-7 I&M, as a general practice, complies with the training recommendations as set forth in this regulatory position.

C-8 All I&M inspection, examination and test personnel are instructed in the normal course of employee training in radiation protection and the means to minimize radiation dose exposure.

C-10 I&M maintains documentation to show that inspection personnel meet the minimum requirements of "Level I," and that management, supervisory and engineering personnel meet the minimum requirements of ANSI 18.1. |*

10. <u>N45.2.8</u>,

10a. Sec. 2.9e

<u>Requirement</u>

Section 2.9e of N45.2.8 lists documents relating to the specific stage of installation activity which are to be available at the construction site.

Exception/Interpretation

All of the documents listed are not necessarily required at the construction site for installation and testing. AEPSC and I&M assure that they are available to the site, as necessary. |*

10b. Sec. 2.9e

Requirement

Evidence that engineering or design changes are documented and approved shall be available at the construction site prior to installation.

Exception/Interpretation

Equipment may be installed before final approval of engineering or design changes. However, the system is not placed into service until such changes are documented and approved.

10c. Sec. 4.5.1

Requirement

"Installed systems and components shall be cleaned, flushed and conditioned according to the requirements of ANSI N45.2.1. Special consideration shall be given to the following requirements: ...." (Requirements are given for chemical conditioning, flushing and process controls.)

Exception/Interpretation

Systems and components are cleaned, flushed and conditioned as determined on a case-by-case basis. Measures are taken to help preclude the need for cleaning, flushing and conditioning through good practices during maintenance or modification activities.

11. N45.2.9

11a. Sec. 5.4, Item 2

Requirement

Records shall not be stored loosely. "They shall be firmly attached in binders or placed in folders or envelopes for storage on shelving in containers." Steel file cabinets are preferred.

Exception/Interpretation

Records are suitably stored in steel file cabinets, or on shelving in containers. Methods other than binders, folders, or envelopes (for example, dividers) may be used to organize the records for storage. |*

11b. Sec. 6.2

Requirement

"A list shall be maintained designating those personnel who shall have access to the files".

Exception/Interpretation

Rules are established governing access to and control of files as provided for in ANSI N45.2.9, Section 5.3, Item 5. These rules do not always include a requirement for a list of personnel who are authorized access. It should be noted that duplicate files and/or microforms may exist for general use.

11c. Sec. 5.6

Requirement

When a single records storage facility is maintained, at least the following features should be considered in its construction: etc.

Exception/Interpretation

The Cook Nuclear Plant Master File Room and other off-site record storage facilities comply with the requirements of NUREG-0800 (7/81), Section 17.1.17.4.

12. Reg. Guide 1.144/ANSI N45.2.12

12a. Sec. C3a(2)

Requirement

Applicable elements of an organization's Quality Assurance Program for "design and construction phase activities should be audited at least annually or at least once within the life of the activity, whichever is shorter."

Exception/Interpretation

Since most modifications are straight forward, they are not audited individually. Instead, selected controls over modifications are audited periodically.

12b. Sec. C3b(1)

Requirement

This section identifies procurement contracts which are exempted from being audited.

Exception/Interpretation

In addition to the exemptions of Reg. Guide 1.144, AEPSC/I&M considers that the National Institute of Standards and Technology, or other State and Federal Agencies which may provide services to AEPSC/I&M, are not required to be audited.

12c. Sec. 4.5.1

Requirement

Responses to adverse audit findings, giving results of the review and investigation, shall clearly state the corrective action taken or planned to prevent recurrence. "In the event that corrective action cannot be completed within thirty days, the audited organization's response shall include a scheduled date for the corrective action."

Exception/Interpretation

AEPSC/I&M take the position that certain circumstances warrant more than thirty (30) days to completely investigate the cause and/or total impact of an adverse finding. For these circumstances, an initial thirty (30) day response will be provided which addresses a schedule for known corrective actions, the reason why additional investigation time is needed, and a schedule for completion of the investigation. These initial responses require the approval of the Director - Quality Assurance.

13. N45.2.13,

13a. Sec. 3.2.2

Requirement

N45.2.13 requires that technical requirements be specified in procurement documents by reference to technical requirement documents. Technical requirement documents are to be prepared, reviewed and released under the requirements established by ANSI N45.2.11.

Exception/Interpretation

For replacement parts and materials, AEPSC/I&M follow ANSI N18.7, Section 5.2.13, Subitem 1, which states: "Where the original item or part is found to be commercially 'off the shelf' or without specifically identified QA requirements, spare and replacement parts may be similarly procured, but care shall be exercised to ensure at least equivalent performance."

13b. Sec. 3.2.3

Requirement

"Procurement documents shall require that the supplier have a documented Quality Assurance Program that implements parts or all of ANSI N45.2 as well as applicable Quality Assurance Program requirements of other nationally recognized codes and standards."

Exception/Interpretation
Refer to Item 2j.

13c. Sec. 3.3(a)
Requirement
Reviews of procurement documents shall be performed prior to release for bid and contract award.

Exception/Interpretation
Documents may be released for bid or contract award before completing the necessary reviews. However, these reviews are completed before the item or service is put into service, or before work has progressed beyond the point where it would be impractical to reverse the action taken.

13d. Sec. 3.3(b)
Requirement
Review of changes to procurement documents shall be performed prior to release for bid and contract award.

Exception/Interpretation
This requirement applies only to quality related changes (i.e., changes to the procurement document provisions identified in ANSI N18.7, Section 5.2.13.1, Subitems 1 through 5). The timing of reviews will be the same as for review of the original procurement documents.

13e. Sec. 10.1
Requirement
"Where required by code, regulation, or contract requirement, documentary evidence that items conform to procurement documents shall be available at the nuclear power plant site prior to installation or use of such items, regardless of acceptance methods."

Exception/Interpretation
Refer to Item 2j.

Requirement
"Post-installation test requirements and acceptance documentation shall be mutually established by the purchaser and supplier."

Exception/Interpretation
In exercising its ultimate responsibility for its Quality Assurance Program, AEPSC/I&M establishes post-installation test requirements giving due consideration to supplier recommendations.

14. Reg. Guide 1.146/ANSI N45.2.23 and ANSI N45.2.2.12

14a. ANSI N45.2.23, Sec. 1.1

Requirement
This standard provides requirements and guidance for the qualification of audit team leaders, henceforth identified as "lead auditors." |*

14b. ANSI N45.2.12, Sec. 4.2.2

Requirement
A lead auditor shall be appointed team leader.

Exception/Interpretation
The AEPSC audit program is directed by the AEPSC Director - Quality Assurance and is administered by designated QA Division section managers/supervisor who are certified lead auditors.

Audits are, in most cases, conducted by individual auditors, not by "audit teams." These auditors are certified in accordance with established procedures and are assigned by the responsible QA section manager/supervisor based on their demonstrated audit capability and general knowledge of the audit subject. In certain cases, this results in an individual other than a "lead auditor" conducting the actual audit function. |*

Attachment 1 to AEP:NRC:0847V

Specific Changes to the QAPD

Section(s)	Proposed Changes(s)
Policy Statement 1.7.1.2.1 1.7.1.2.2 1.7.1.2.3 1.7.1.2.5 1.7.2.2.1 Figure No. 1.7-1 Figure No. 1.7-3	Title changed to reflect new corporate management structure; i.e., from "Chairman of the Board and Chief Executive Officer" to "President and Chief Executive Officer."
1.7.1.2.5 1.7.2.1 1.7.2.2.5 1.7.6.2.1	Clarification: Added the word "Updated"; i.e., "Updated FSAR..."
1.7.1.2.2	Fourth paragraph clarification: Added the word "Program"; i.e., "Cook Nuclear Plant Quality Control Program..."
1.7.1.2.5 Quality Assurance Division	Second paragraph, fifteenth item clarification: Included the words "audits or"; i.e., "Conduct in-process QA audits or surveillances..."
1.7.1.2.5 QA Auditor Qualification and Certification Program	Deleted the words "Orientation, Training" from the subheading; added the word "Auditor." Also deleted reference to QA providing a general training program to AEPSC personnel engaged in activities affecting the quality of safety-related item. This responsibility has been reassigned to the Nuclear Operations Division and is now addressed under those responsibilities (page 1.7-18)
1.7.1.2.5 Nuclear Operations Division	Deleted reference to Nuclear Operations Division middle management. This allows for lower level reorganization, but maintains delineated divisional responsibilities.

Section(s)	Proposed Changes(s)
1.7.1.2.5 (cont.) Nuclear Operations Division	Included additional item (#42) to reflect current Nuclear Operations Division responsibilities regarding development and implementation of training for AEPSC personnel engaged in activities affecting the quality of safety-related items.
1.7.1.2.5 AEPSC Engineering and Design	Included additional paragraph regarding AEPSC Electrical Engineering and System Planning Departments' responsibilities with regard to Cook Nuclear plant.
1.7.1.2.5 Civil Engineering Department	Deleted reference to Civil Engineering Department middle management. Delineated Department responsibilities are maintained. Also added the words "technical aspects for the" to further clarify Civil Engineering Department's responsibilities with regard to Cook Nuclear Plant.
1.7.1.2.5 Design Department	Deleted reference to Design Department middle management. In addition, Item (#7) was rewritten to clarify the Design Department's responsibilities with regard to the NSDRC and the NSDRC subcommittees.
1.7.1.2.5 Nuclear Engineering Department	Deleted reference to Nuclear Engineering Department middle management. In addition, included "nuclear steam supply" in Item (#2) to further clarify the Nuclear Engineering Department's responsibilities with regard to Cook Nuclear Plant.

Section(s)	Proposed Changes(s)
1.7.1.2.5 Project Management and Construction Department	Deleted reference to Project Management and Construction Department middle management. Also added item regarding NSDRC participation.
1.7.1.2.5 Purchasing and Stores Department	Included additional items (i.e., #2 and #9) to reflect the Purchasing and Stores Department's current responsibilities with regard to the Cook Nuclear Plant. In addition, for clarification in Item #4, deleted reference to "I&M Purchasing Department" and replaced it with "Cook Plant Site Purchasing Section". Also deleted reference to notification of suppliers, I&M Purchasing Department and/or Cook Nuclear Plant Stores Supervisor with regard to supplier acceptability status.
1.7.2.2.4	Included additional paragraphs for further clarification of Nuclear Engineering Procedures (NEPs)
1.7.2.2.5	First paragraph: Deleted third sentence, "Imposition of these guides/standard..." to eliminate confusion or misinterpretation by the reader/reviewer. This statement now expanded and appears in Section 1.7.4.2.3.
1.7.3.2.1 1.7.3.2.2	Changes to these are included to accommodate more efficient handling of design changes that would have an insignificant seismic effect on safety related items. Specifically, changes that would introduce only insignificant seismic effects (e.g., placing a non-safety related instrument on a safety related control panel) would be processed via that Plant Modification (PM) process. The PM process has adequate controls to assure that the changes are limited to

Section(s)	Proposed Changes(s)
1.7.3.2.1 (cont.) 1.7.3.2.2	insignificant effects. The PM process also provides for more expedient and efficient processing, thus allowing our engineers more time to focus their attention on more significant safety issues. Editorial change; clarified the previously used word "mechanism."
1.7.3.2.13	First paragraph: 3) Included the phrase "...or by the dedication plan process" to further define the implementation of procurement control instructions and procedures. In addition, the third paragraph was rewritten to further clarify responsibilities.
1.7.4.2.1	First paragraph: Deleted "as the source document..." in the first sentence and "... for the Cook Nuclear Plant (DCC Specifications)" in the second sentence, for grammatical clarification. In addition, the second paragraph was rewritten to further clarify procurement of safety related items or services.
1.7.4.2.2	Item b): Statement added to accurately depict how imposition is implemented.
1.7.4.2.3	Second paragraph (last sentence): rewritten for grammatical clarification.
1.7.7.2.1	Deleted the words "and upgrading"
1.7.7.2.2	

Section(s)	Proposed Changes(s)
1.7.7.2.8	Rewritten to further clarify supplier record retention responsibility.
1.7.7.2.9	Rewritten to clarify evaluation of supplier's capability to furnish valid certificates.
1.7.9.2.2 1.7.9.2.3	Deleted reference to "AEPSC Staff Engineer - Chief Metallurgist (NDE Administrator)"; replaced with "AEPSC Cognizant Engineer - Welding and NDE Administrator" to reflect current position title.
1.7.9.2.2	Fourth paragraph, first sentence: For further clarification, added the phrase "or in the approved contractor's manual."
1.7.9.2.3	Second paragraph: Replaced "AEPSC Welding Specification" with "AEP Welding Manual." In addition, included the phrase "in accordance with AEPSC procedures."
1.7.13.2.3	For further clarification, included the phrase, "or are otherwise specified on the procurement order."
1.7.17.2.1	Item h: Included "dedication plans." Item i: Deleted the phrase "(which are also classified as safety related item)."
1.7.18.2.4	Second paragraph: Included "NSDRC membership."
Figure No. 1.7-4	Deleted reference to Quality Assurance Division support personnel on the organization flow chart.

Section(s)	Proposed Changes(s)
Appendix B	Item 12: Included reference to "ANSI N45.2.12."
Appendix B	Item 12b: Editorial change; corrected name of agency.
Appendix B	Item 12c: Added Section 4.5.1 regarding responses to adverse audit finding. This issue was previously deemed acceptable by the NRC when submitted in AEP submittal letter AEP:NRC:0847T dated October 30, 1990.

The entire QAPD was repaginated. Proposed changes in content are reflected by margin bars in the margin to the right of the text.

Please note: A margin bar followed by a asterisk (i.e., /*) reflects an editorial change, (e.g., grammar; punctuation) only.