HOUSTON LIGHTING & POWER COMPANY
ALLENS CREEK NUCLEAR GENERATING STATION - UNIT NO. 1
PRELIMINARY SAFETY ANALYSIS REPORT
AMENDMENT NO. 59
INSTRUCTION SHEET

This amendment contains information pertaining to the PSAR Update. Each revised page bears the notation Am. No. 59, (6/81) at the bottom of the page. Vertical bars with the number 59 representing Amendment No. 59 have been used in the margins of the revised pages to indicate the location of the revision on the page.

The following page removals and insertions should be made to incorporate Amendment No. 59 into the PSAR.

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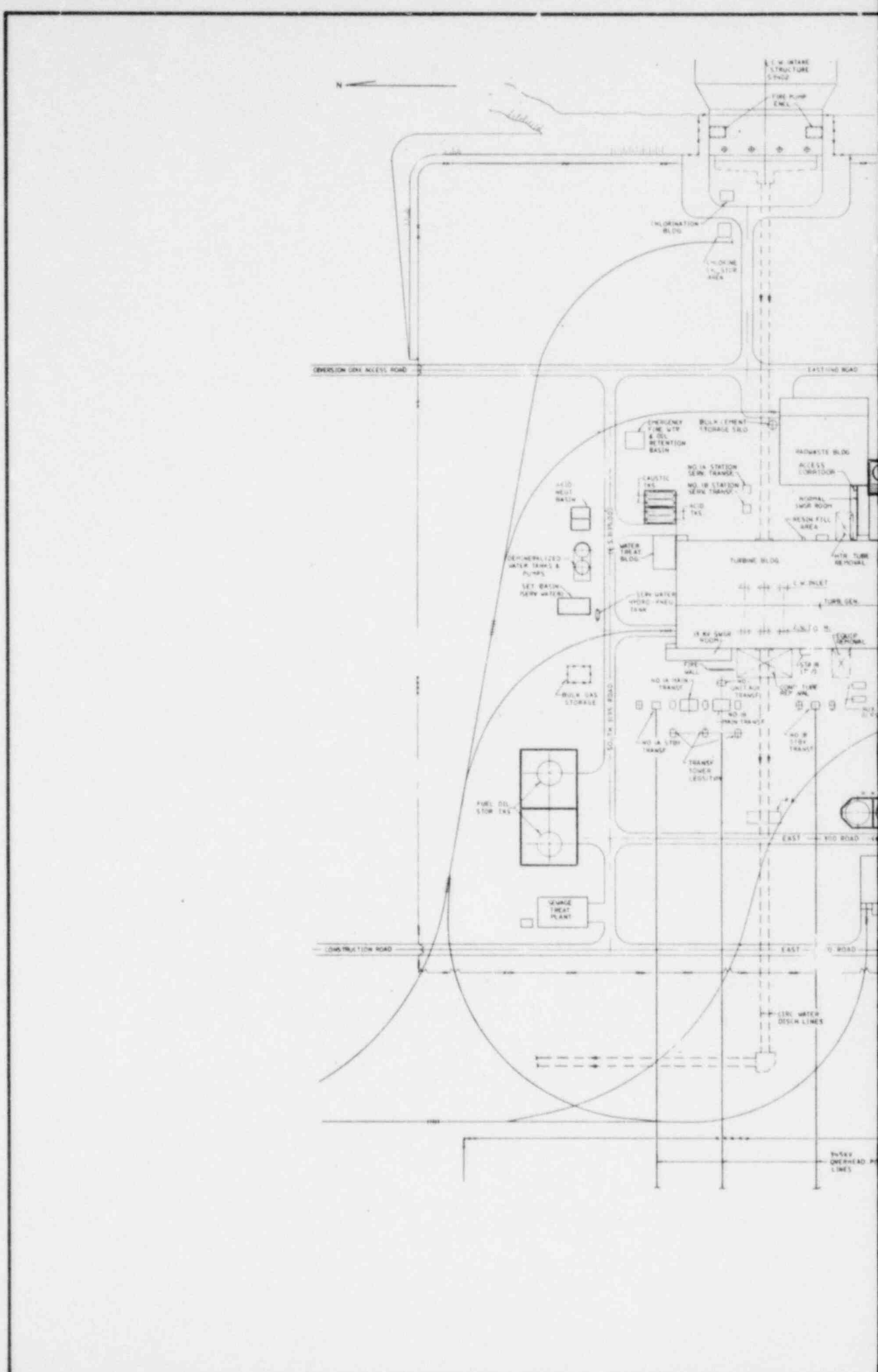
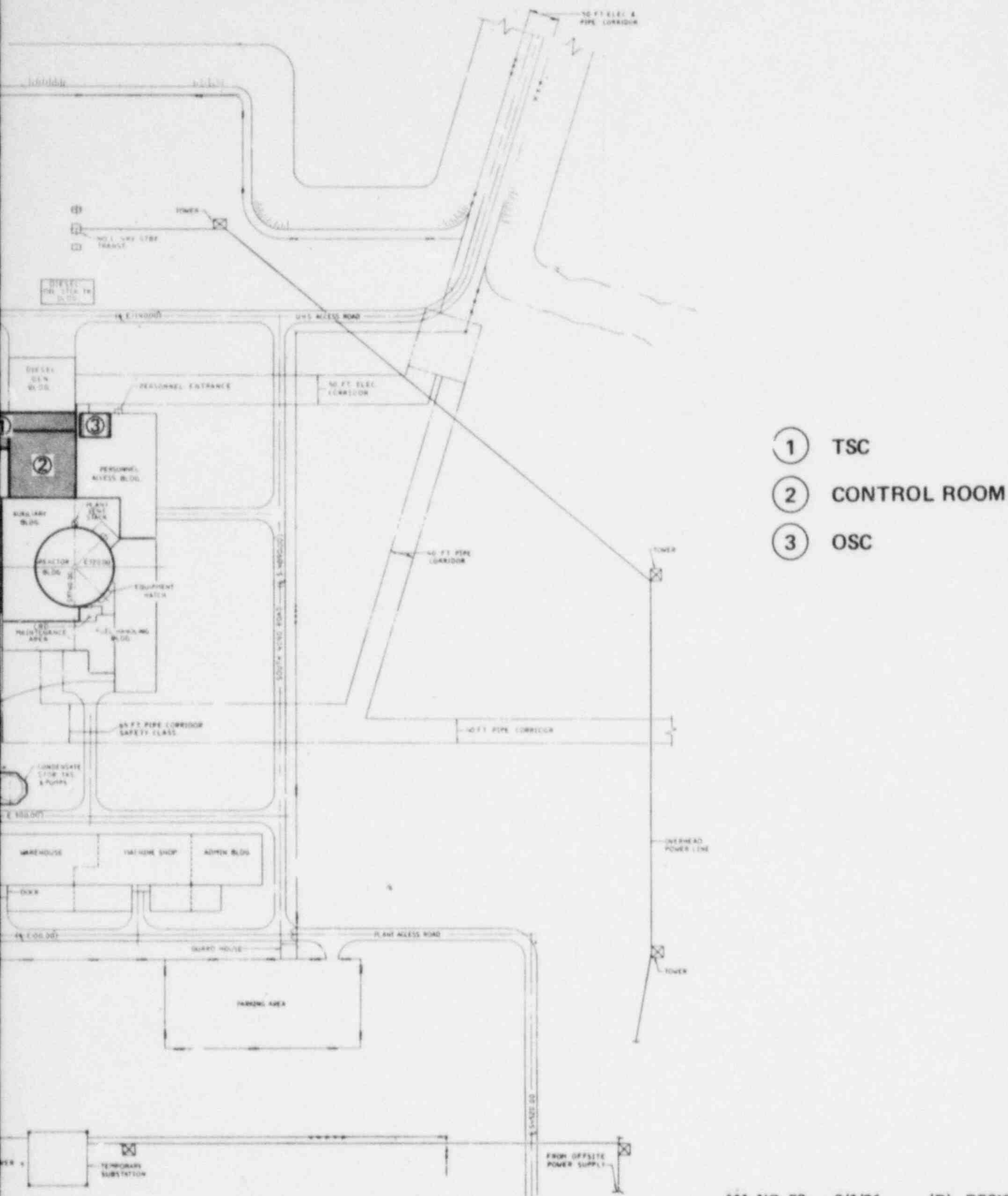

N
C.W. INTAKE STRUCTURE
FIRE PUMP ENCL
CHLORINATION BLDG
CONVERSION DEPT ACCESS ROAD
EASTING ROAD
EMERGENCY FIRE WTR & OIL RETENTION BASIN
BULK CEMENT STORAGE SILO
RADWASTE BLDG
ACCESS CORRIDOR
NO. 1A STATION SERV. TRANSF.
NO. 1B STATION SERV. TRANSF.
CAUSTIC TKS
ACID TKS
ACID NEUT BASIN
NORMAL SWGR ROOM
RESIN FILL AREA
DEMINERALIZED WATER TANKS & PUMPS
WATER TREAT BLDG
TURBINE BLDG
HTR. TUBE REMOVAL
C.W. INLET
TURB GEN
SET BASIN (SERV WATER)
SERV WATER HYDRO-PNEU TANK
EQUIP REMOVAL
FIRE WALL
BULK GAS STORAGE
NO. 1A MAIN TRANSF
NO. 1B MAIN TRANSF
NO. 1A STBY TRANSF
NO. 1B STBY TRANSF
TRANSF TOWER
SOUTH SERV ROAD
FUEL OIL STOR TKS
SEWAGE TREAT PLANT
CONSTRUCTION ROAD
EAST ROAD
CIRC WATER DISCH LINES
OVERHEAD LINES



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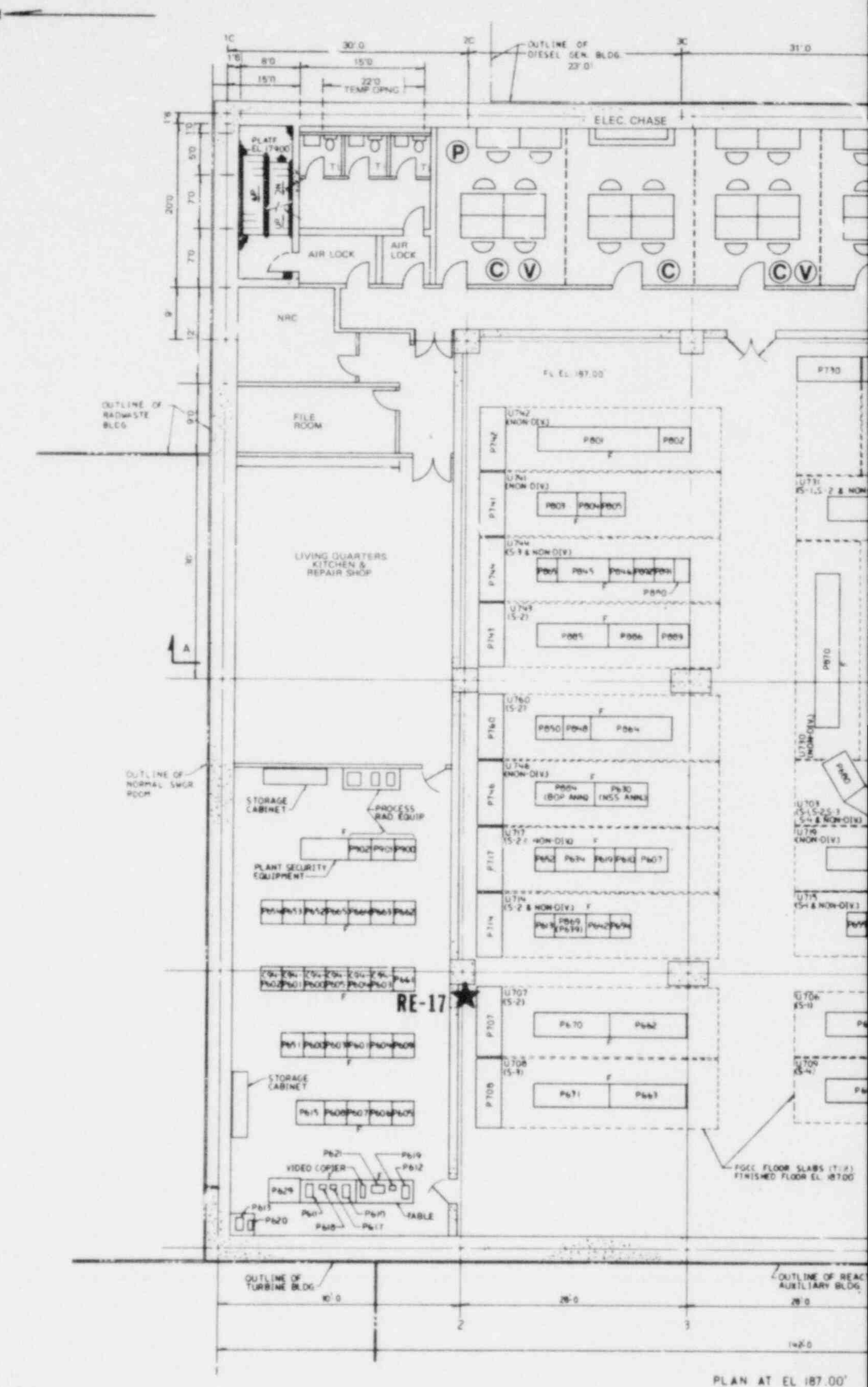
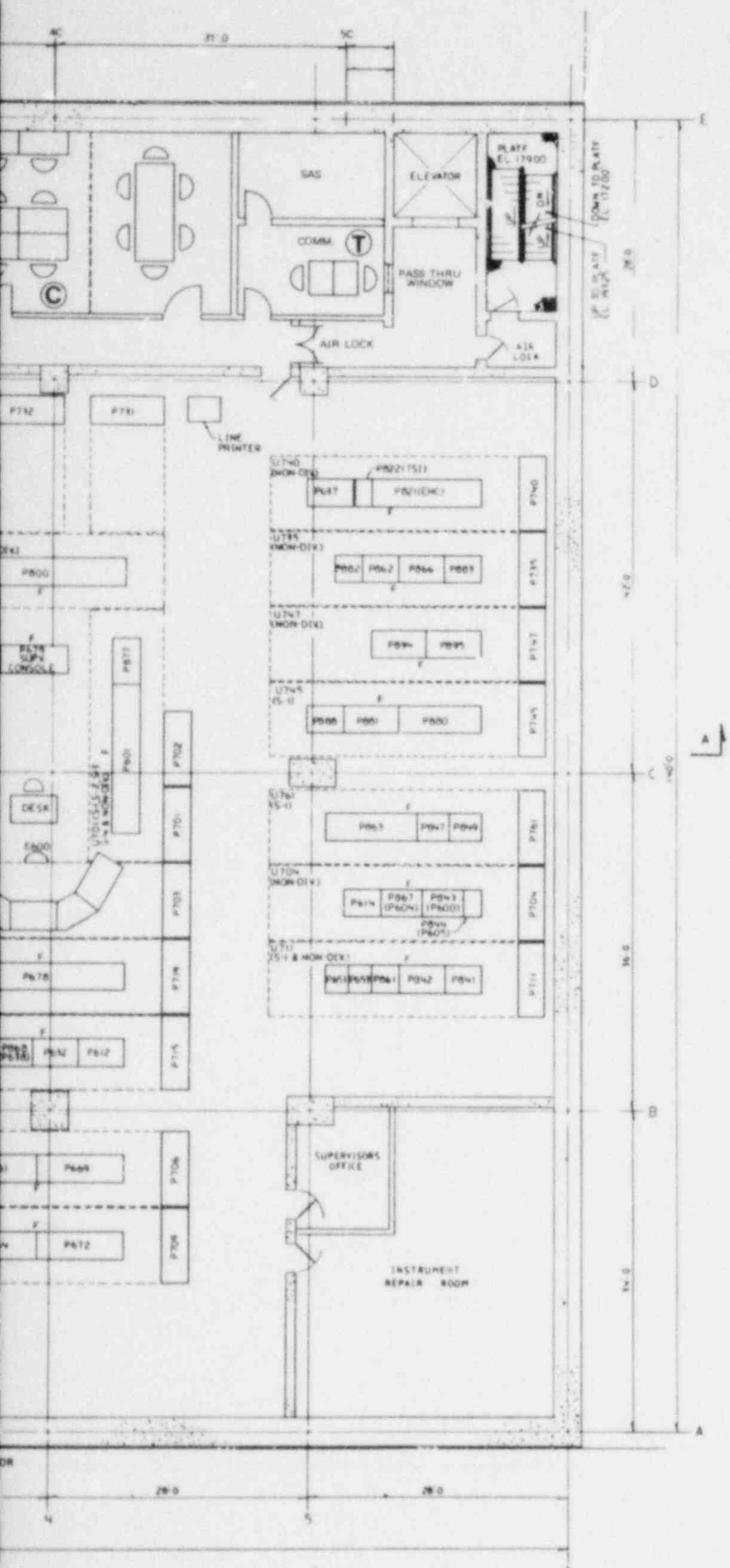
Allens Creek Nuclear Generating Station
Unit 1

ENLARGED PLOT PLAN

FIGURE 1.2–3



NOTE: FIG. 1.2-33 MODIFIED TO SHOW PROPOSED LOCATION OF TSC



SYMBOLS

- (C) CRT
- (V) VIDEO COPIER
- (P) PRINTER
- (T) TELECOPIER

NOTES:

1. FOR GENERAL NOTES AND REFERENCE DWGS SEE SHEET NO. 1

COLUMN DESIGNATIONS ARE TEMPORARY AS SHOWN AND ARE SUBJECT TO CHANGE UPON COMPLETION OF ALL BUILDINGS.

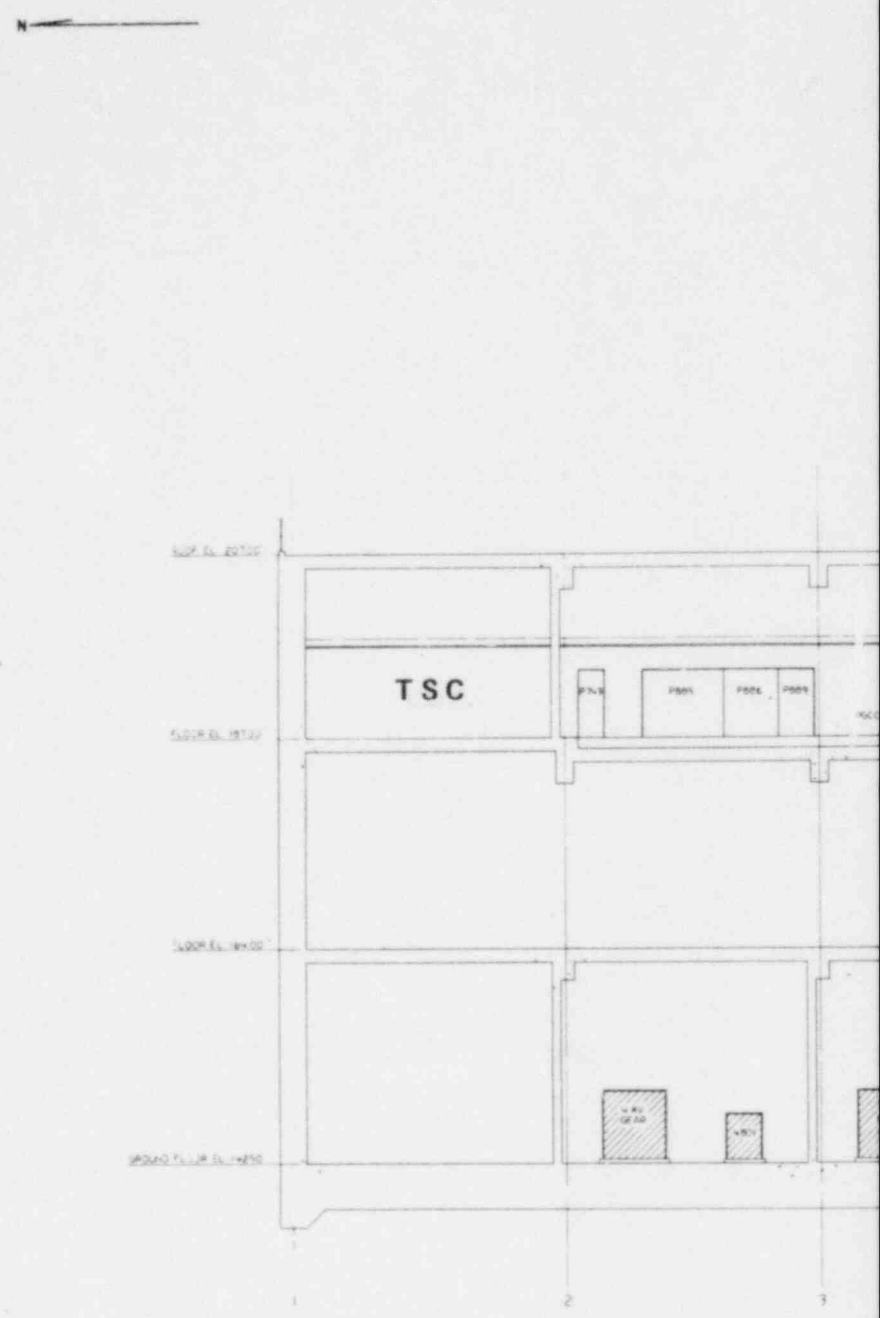
LEGEND: ★ RE - RADIATION ELEMENT (GM TUBES)

AM. NO. 59 6/1/81 (D) - DESIGN

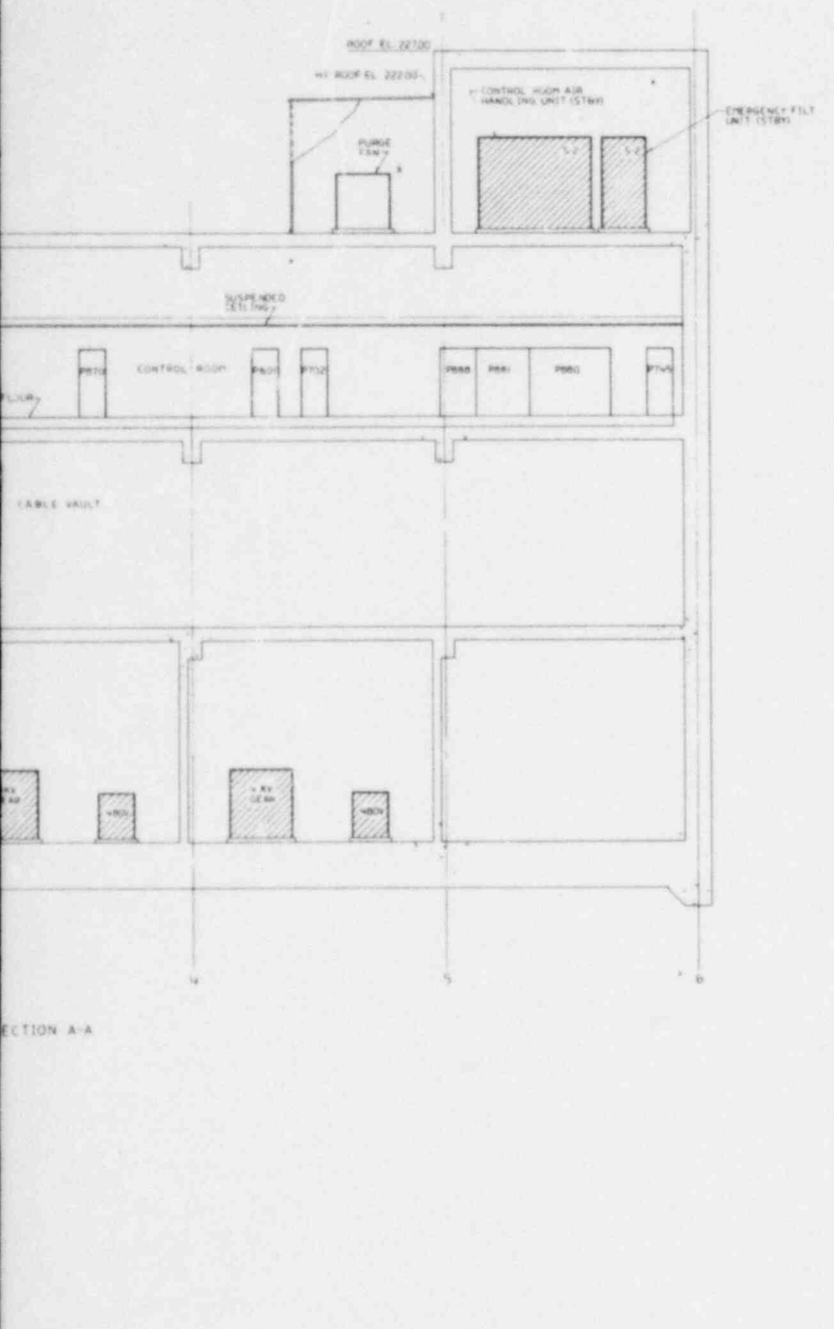
HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

GENERAL ARRANGEMENT - CONTROL BLDG. PLAN AT EL. 187.00'

FIGURE 1.2–33



NOTE: FIG. 1.2-35 MODIFIED TO SHOW PROPOSED LOCATION OF TSC.



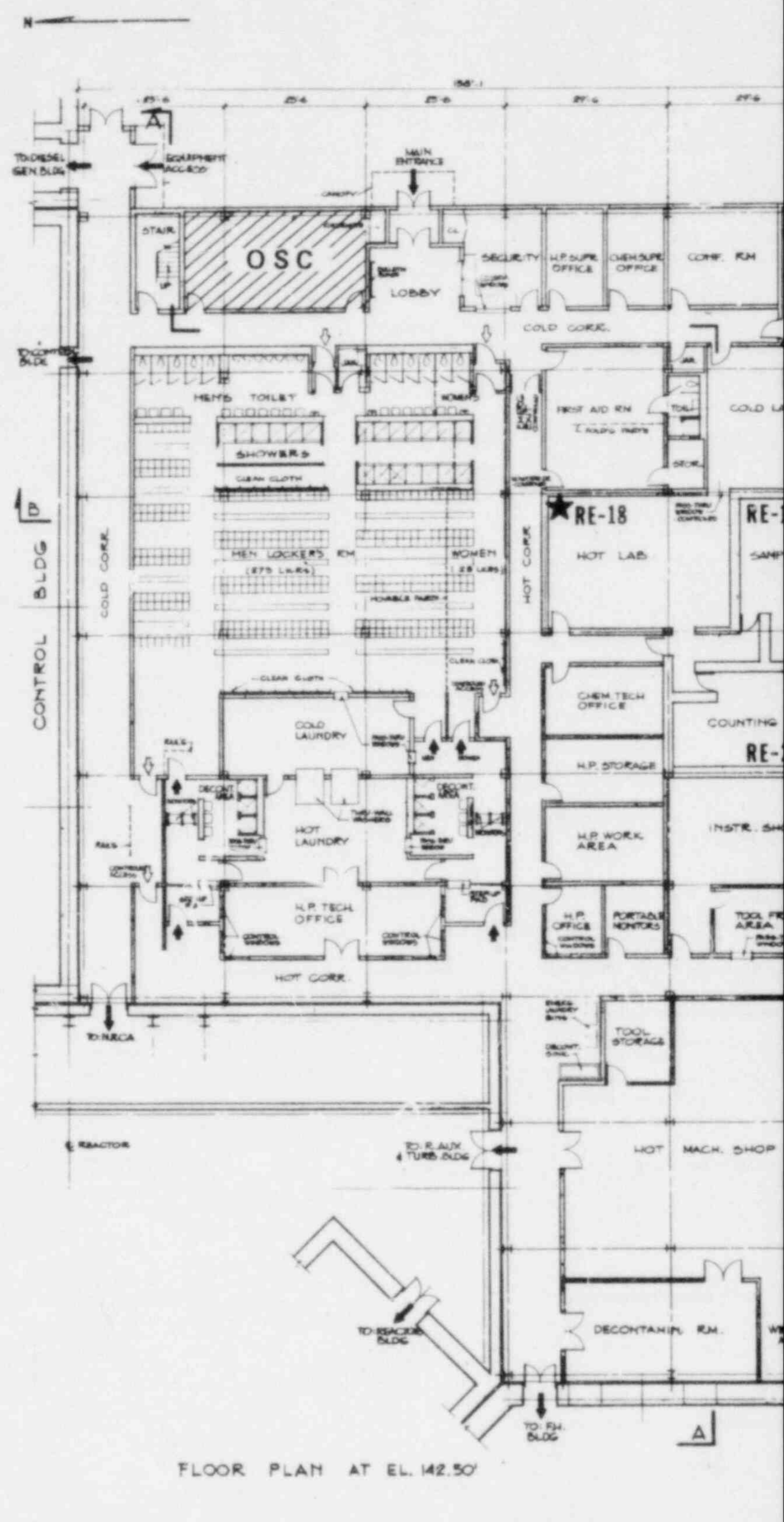
NOTES:
FOR GENERAL NOTES AND REFERENCE DWGS SEE SHEET NO 1

Note:
SIGNATURES ARE TEMPORARY AS SHOWN AND ARE SUBJECT TO CHANGE UPON COMPLETION OF ALL TEDES

AM. NO. 59 6/1/81 (D) - DESIGN

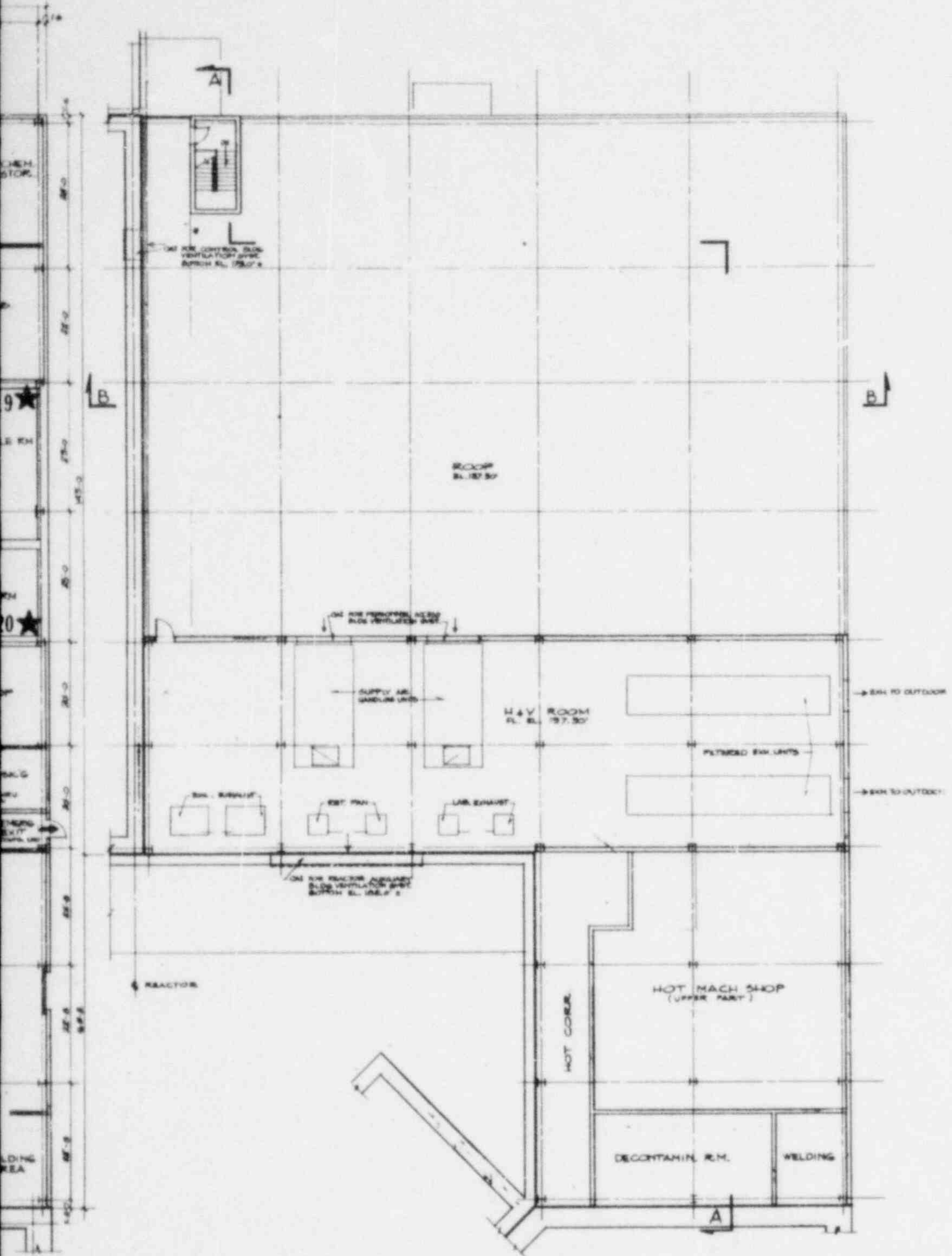
HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

GENERAL ARRANGEMENT
CONTROL BUILDING
SECTION A–A
FIGURE 1.2–35



FLOOR PLAN AT EL. 142.50'

NOTE: FIG. 1.2-38a MODIFIED TO SHOW PROPOSED LOCATION OF OSC.



PLAN AT EL. 157.50'

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★ RE – RADIATION ELEMENT (GM TUBES)

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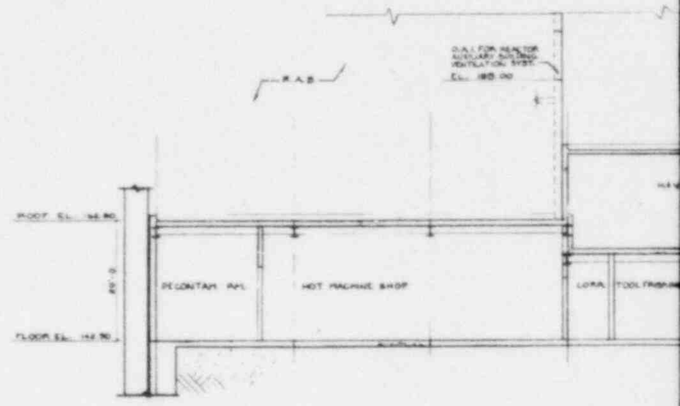
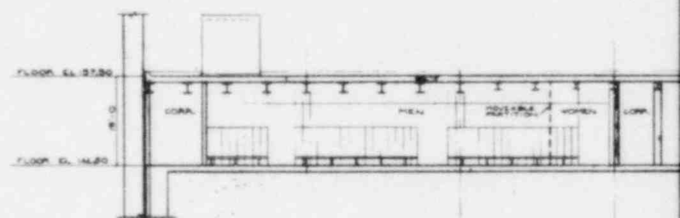
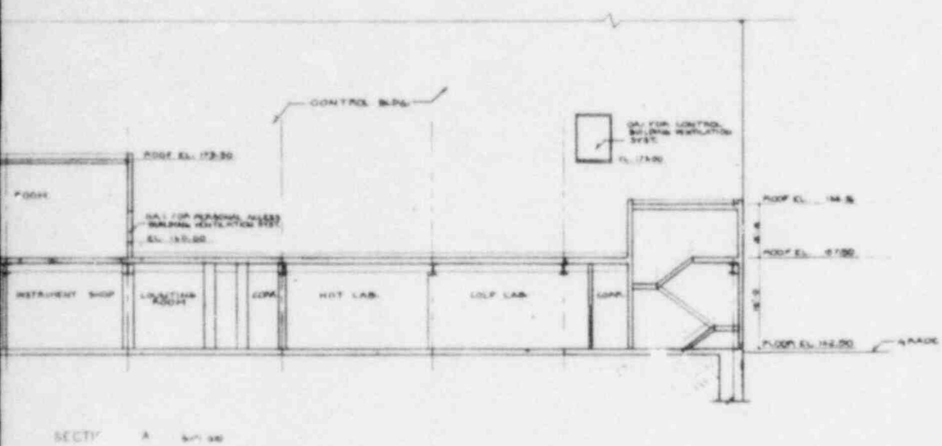
Allens Creek Nuclear Generating Station
Unit 1

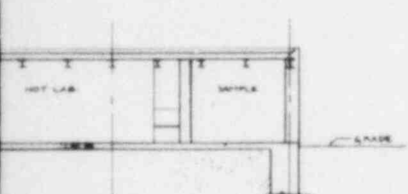
GENERAL ARRANGEMENT
PERSONNEL ACCESS BUILDING
FLOOR PLANS
FIGURE 1.2–38a





SECTION B





NOTE: THESE SECTIONS DO NOT CUT THE PROPOSED LOCATION OF THE OSC.

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HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

GENERAL ARRANGEMENT
PERSONNEL ACCESS BUILDING
SECTIONS
FIGURE 1.2–38b

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2.2.3.2 Gas Pipelines

When relocated, the 24-inch Texas Utilities Company pressurized natural
gas line will pass about 9300 feet northeast of the nearest Category I 38(U)
structure. The six-inch Shell Pipeline Company liquefied petroleum gas
(LPG) line as well as the 8-inch crude line pass about 8000 feet northwest
of the nearest Category I structure (see Figure 2.2-2).

The 24-inch Texas Utilities Company pipeline operates at a maximum flow
rate of 2.5×10^8 ft^3/day at operating pressures ranging from 750 psi
to 900 psi. The design operating pressure of the line is 975 psi. 41
Q312.5
The closest proximity of the line to any Category I plant structure is 9300 Q312.6
feet to the ultimate heat sink structure.

Neither the liquified petroleum gas line (LPG) nor the crude oil line belonging to Shell Oil pass under the lake. The closest proximity of these lines to the lake is approximately 2500 ft from the lines to the dam. The closest proximity between these lines and Category I plant structures is approximately 8000 feet.

The consequences of a rupture in the LPG line at its closest point to plant Category I structures have been evaluated assuming worst atmospheric dispersion as well as formation of a low lying non-dispersive petroleum cloud along the terrain depression outlined by the 140-foot isocline which follows the Allens Creek into the lake.

The 6" LPG has been evaluated under the assumption of the line carrying
pure propane. 59

The hazards to the Allens Creek Plant Category I structures and cooling
lake dike from detonations of gaseous clouds resulting from breaks in 41
proximate natural and liquefied petroleum gas lines have been analyzed Q312.5
using conservative but realistic models. Q312.6

Results of the analyses which are presented in Appendix 2.2-A indicate that
the plant's Category I structures will suffer no adverse effects from the
consequences of any pipeline rupture and subsequent detonation and/or
deflagration of the cloud. 59

In the event that the applicant's analyses do not demonstrate the assurance
recommended by Standard Review Plans 2.2.1-2.2.2 that the postulated
rupture of Shell 6" LPG line need not be considered as a design basis
event, physical changes to the site and/or environs to provide such 51
assurance will be made. No later than the submittal of the application for
an operating license, the applicant will provide for staff review and
approval physical measures to cope with the potential hazard. If necessary,
the applicant will relocate the pipeline if acceptable resolution cannot be 52
demonstrated by analysis or other alternate physical measures.

APPENDIX 2.2A

PIPELINE BREAK EVALUATION

1.0 EVALUATION OF THE BREAK OF 6" LPG PIPELINE

The consequences of a complete severance of the 6 inch LPG line has been evaluated for the case of the line filled with volatile fluid and both with and without consideration of the use of a leak detection system. The choice of the volatile hazardous fluid for analysis is propane. There are two reasons for this choice.

a) The line currently carries batches of gasoline, isopentane, normal butane and isobutane, plus distillate, but could carry propane also, and possibly other hydrocarbons. Pound for pound, propane has effectiveley the same TNT mass equivalency of other flammable hydrocarbons such as butane, isobutane, ethylene, butadiene, propylene, etc. (See Reference 1) 59

b) There are several instances of unconfirmed propane-air cloud deflagrations and at least one detonation, against which it is possible to compare results of the subsequent analyses.

The analysis has been performed on the basis of the following assumptions:

a) Double ended rupture of the line occurs instantaneously and at the closest point to plant Category I structures (8000 feet).

b) The released petroleum liquid-gas mixture escapes from the break at the critical velocity for two phase flow, and at the design pressure of the line, 1,000 psig (a conservative assumption since the operating pressure is only 750 psi).

c) The temperature of the atmosphere is assumed to be 72^{o}F. Higher temperatures would lead to higher vaporization of escaping propane, but the flow rate would be less due to the higher quality at the exit plane.

d) Five percentile meteorology is assumed, which is equivalent to a Pasquill F inversion with wind speed of 0.8 mps in the direction of the plant structures.

e) A leak detection system is available which is capable of detecting a leak of approximately 20 bbls/ hr (or about 5% of operating flow) in 3 minutes after the leak occurs. After detection, line shutdown and isolation will be taken in 5 minutes.

Normally the pumps at the Sheridan Station, located approximately 34.8 miles upstream from the Allens Creek crossing would be stopped and the line block valve at John Sue Junction would be closed. John Sue Junction is located approximately 15 miles downstream from the Allens Creek crossing. Both actions would be accomplished by operator intervention rather than automatically.

Moreover, there are block valves straddling the Allens Creek crossing vicinity as shown in Figure 2.2-2. The upstream valve "A" is located approximately at Station 4120+99 West of hwy 36 just past the western fork of Allens Creek. The other valve "B" is located at Station 3326+00, west of the Brazos River and approximately 20,000 feet downstream of the first valve. These valves are currently manual operated, but can be upgraded to close remotely when a signal is given by the operator.

1.1 CALCULATION OF FLOW RATE OUT OF THE BREAK

Two kinds of break require examination:

a) leaks larger than 5% of operating flow requiring 8 minutes to isolate the section containing the break, and

b) leaks smaller than 5% of operating flow.

The large break scenario is examined first.

The propane in the line will, upon the instant of the break, decompress isenthalpically to a saturation pressure of 125 psia immediately because of the very large speed of sound in the liquid. A decompression wave will travel very rapidly away from the break leaving the fluid behind at the saturation pressure. Since propane would issue from the break to atmosphere at 72^{o}F, approximately 1/3 of it would quickly vaporize, cooling the remainder to its boiling point of about - 44^{o}F. Hence the process of decompression is described by the throttling process shown in Figure 1.1. From that figure the exit plane quality, x, of the fluid can be estimated from: 59

$$v = v_f + x\, v_{fg}$$

where v=2.4 ft^3/lb, v_f=.0275 ft^3/lb, v_g=6.6 ft^3/lb, $v_{fg}=v_g-v_f$

Hence $x = 0.36$

To estimate the flow rate out of the break, Fauske's equation Reference 2 for critical two-phase mass velocity is used:

$$G_{Crit} = (-g\sigma/(k_1 dv_g/dp + k_2 dx/dp + k_3 dv_f/dp))^{1/2}$$

where $k_1 = (1-x+\sigma x)x$

$$k_2 = v_g\,(1+2\,\sigma x-2x) + v_f(2\sigma x-2\sigma-2\sigma^2 x+\sigma^2)$$

$$k_3 = (1-x(\sigma-2)-x^2(\sigma-1))$$

and $\sigma = (v_g/v_f)^{1/2}$

Initially the break discharges approximately 568 lbm/sec. This discharge rate decays as the pressure in the line drops, until such time that inertial flow is established .

During the first 8 minutes following the break, it is very conservatively assumed that the 568 lbm/sec remain constant. At that time the pump pressurized side of the break is assumed to be about 200 psi while the pressure on the other portion of the line is assumed to have fallen to 100 psi.

Inertial flow conservatively computed with single phase rather than two phase friction factors, is predicted to peak at 70 lbm/sec from the pump pressurized side of the break and 30 lbm/sec from the other side.

The full break scenario is depicted in histogram (See Figure 1.2) wherein the solid line represents the flow rates chosen for the analyses, and the dashed line represents what is physically expected.

1.2 CALCULATION OF DETONABLE CLOUD SIZE

To estimate the atmospheric-'ly dispersed propane and air cloud under Pasquill F conditions with ... fps wind speed, a constant average source of propane of approximately 100 lbm/sec has been assumed to continue to release from the break indefinitely. At a specific volume of propane of approximately 9.0 ft^3/lbm, this corresponds to a source strength of 900-1000 ft^3/sec of propane. Conservatively assuming 1000 ft^3/sec, and further assuming that vertical dispersion is depressed to account for the heavier than air character of the ensuing cloud, the resulting cloud is depicted in Figure 1.3. 59

The centerline (directly downwind) concentration of propane is determined by:

$$\chi_{cl} = \frac{Q}{\pi \sigma_y \quad \sigma_z \; u}$$

where Q = 1000 ft^3(STP) of propane per second

u = 2.6 ft/sec

and σ_y, σ_z are the plume dispersion standard deviations obtained from reference 3

The off-centerline concentrations are determined by

$$\chi = \chi_{cl} \exp \left\{ -1/2 \left[(y/\sigma_y)^2 + (z/\sigma_z)^2 \right] \right\}$$

It is very important to determine which are the proper dispersion standard deviations to be used in solving for the equilibrium concentrations.

Dispersion standard deviations for instantaneous "puff" releases and continuous constant releases are considerably different; the former being much smaller than the latter.

The reason for this is that the continuous release deviations account for a part of averaging process caused by wind vagaries resulting in meandering of the plume about its axis. The σ_y and σ_z then account for this effect. For a puff release this meandering averaging process is absent and the puff disperses more slowly. The case of a pipeline break is neither a puff release nor a continuous constant release, but rather a continuous release at an initially rapid decreasing rate, followed by an almost constant release.

The meandering process would then be in effect and the proper dispersion standard deviations are the continuous release standard deviations.

Table 1-1 lists pertinent centerline concentrations at several distances from the break, and also distances from the cloud axis at which the propane concentration falls to flammable and detonable limits.

Since lower volumetric flow rate result in smaller cloud formation due to increased atmospheric dispersion, the small break scenario is bounded by that previously described for the large break incident.

1.3 CALCULATION OF EFFECTS OF DETONATION

The capability of the Allens Creek plant to withstand potential explosions is evaluated by determining the total quantity of detonable or deflagrable mixture contained in the detonable or deflagrable cloud.

The detonable volume is calculated from Figure 1.3 to be $1.155 \times 10^7 ft^3$; whereas, the deflagrable volume is $1.95 \times 10^7 ft^3$. The total mass of propane contained in the larger deflagrable volume is estimated to be 9.14×10^4 lbm. One can further estimate that the quantity of propane contained in the over rich cloud region is as a minimum 2.14×10^4 lbm. Moreover, more than 90% of the mass of propane is dispersed past the lower flammable limit. 59

Thus to achieve the clouds depicted in Figure 1.3 in excess of 2×10^6 lbm of propane must be released. This is a larger quantity of propane that can escape while the line is not isolated (8 minutes) and following isolation of the line. For instance if the line is isolated by stopping the pumps at the Sheridan Station and blocking the valve at John Sue Junction, at most 1.8×10^6 lbm of propane can be ejected.

This fact demonstrates that the cloud depicted in Figure 1.3 is a very conservative upper bound of real possible clouds, and that therefore if the plant can withstand the consequences of its detonation or deflagration, it will be safe against any real explosion.

The detonation is conservatively assumed to be at the extreme edge of the cloud closest to the plant. Thus detonation parameters are evaluated at a distance of 3700 ft. At this distance, detonation of the entire detonable cloud, calculated from Figure 1.3 assuming no isolation of the line would result in the following parameters at the plant safety related structures:

Peak - Overpressure	0.6 psi
Peak Dynamic Pressure	0.009 psi
Peak Reflected Overpressure	1.22 psi
Positive Phase Duration	0.281 msec.

Peak Acceleration (Horizontal or Vertical)	0.048 g
Peak Velocity	0.064
Peak Displacement	0.0057 inches

The plant can withstand these blast induced loadings. The parameters have been conservatively evaluated using the methods described below.

From an enthalpy of detonation release of 260 Kcal/lb of propane air mixtures of 4.9 percent (enthalpy of detonation is insensitive to mixture ratios between 4-5 percent), and a volume of 1.155×10^7 cubic feet, it is possible to compute the total energy released in a hypothetical detonation of the entire detonable cloud, assuming that the whole cloud is at a mixture averaging 4.9 percent. From the total weight of the mixture (944,382 lbs) one computes the total enthalpy released (2.455×10^8 Kcal), divides it by 500 Kcal/lb TNT to derive the equivalent weight of TNT. A 50% yield is conservatively used to compute the TNT explosive equivalent yield, which is then that of 127 tons of TNT. Since work by Iotti et al, shows that indeed the yield of a gaseous detonation is lower than that of TNT. Reference 4 / compares overpressures calculated by assuming gaseous point sources 5 / to overpressures obtained by Kingery 6 / for the same yield, and those measured by Kogarko et al, 7 / for the given gaseous detonation. This comparison shows that Kingery's result would have been comparable to those of References 5 and 7, if a TNT yield of 50 percent had been employed. Reference 8 / cites a yield of 7.5 percent. Thus a conservative estimate of the TNT equivalent of the detonation of the entire cloud can be obtained by using 50 percent yield and use of Kingery charts. 59

It must be pointed out that the method chosen to compute the equivalent TNT yield of the detonation differs from that recommended in Reference 1.

If one had used the methodology of Reference 1, one would have first computed the weight of propane in the detonable cloud (6.056×10^4 lbs), multiplied it by 2.4 to compute the equivalent weight of TNT (72.67 tons).

As can be seen, the method chosen is more conservative. As previously described the small break event is bounded by the large break analyzed above, thus the plant is safe against detonation of atmospherically dispersed clouds resulting from any size in the pipeline.

1.4 GRAVITY SLUMPING EFFECT OF LPG CLOUDS

Since there is a potential for propane cloud to flow along Allens Creek, spillover the 140-foot isocline near the site and reach the plant safety related structures in sufficient concentrations to pose a hazard, analyses have been performed. Therefore the following sections have discussed the different effects of the formation of low-lying propane clouds which may be governed by gravitationally induced flow with limited vertical dispersion.

1.5 CLOUD FORMATION AND DISPERSION MODEL

This section examines the gravity slumping model associated with the propane cloud resulting from 6 LPG inch pipeline break. For boiloff rate $B = 33,800\ BTU/ft^2hr$ it is possible to compute initial radii and height of the pure propane clouds formed at the break as a function of the initial release rate, W assumed to be a constant

$$r_{max} = \left[\frac{\rho_f W(h_g - h_f)}{\pi B}\right]^{\frac{1}{2}} \quad (ft) \qquad (1)$$

and

$$h_{init} = \frac{Br_{max}}{\rho_g u(h_g - h_f)} \quad (ft) \qquad (2)$$

wherein B is the boiloff rate, h_g and h_f are the vapor and liquid enthalpies of propane at the chosen temperature, ρ_g and ρ_f are the vapor and liquid density of propane at that temperature, and u is the wind velocity. 59

The gravity spread of the cloud is then computed by

$$\frac{dR(t)}{dt} = \left\{ 2g \left[\left(\frac{\rho_c(t) - \rho_a}{\rho_a}\right) h(t) + S\,\Delta R \right] \right\}^{\frac{1}{2}} \qquad (3)$$

where $\rho_c(t)$ is the time varying cloud density, ρ_a is the density of ambient air, g is the acceleration of gravity, S is the slope of the ground in ft/ft, and h(t) is the time varying cloud height.

R(t) is the radius of the cloud at time t, or alternatively the distance travelled by the cloud front at time t after the cloud makes contact with banks confining it at an arbitrary distance W, where W can be taken as the half width of the channel.

Air is entrained at the surface of the cloud as it spreads. If the entrainment velocity is defined as

$$V_c = \frac{r}{R}\frac{dR}{dt}$$ while the cloud slumps radially (4)

and

$$V_c = \frac{dR}{dt} - u$$ while it slumps along the wind direction, where u is the wind velocity (5)

the volumetric entrainment of air during radial spreading is given by

$$dQ(t) = \alpha V_c \; 2\pi r dr \qquad (6)$$

while during the subsequent phase of slumping along a channel of width 2W assumed to be in the same direction as the wind direction, it is given by

$$dQ(t) = \alpha V_c \quad 2Wdr \tag{7}$$

Integration of equations (6) and (7) using equations (4) and (5) yield

$$Q(t) = \frac{2\pi\alpha}{3} R^2(t) \frac{dr(t)}{dt} \qquad \text{for radial spreading} \tag{8}$$

and

$$Q(t) = \left\{\frac{\pi}{3} W^2 - \gamma W \left[R(t)-W\right]\right\} \alpha \left[\frac{dR}{dt} - u\right] \tag{9}$$ 59

for initial radial spreading to a radius W followed by axial spreading down wind in a constant width channel (channel width = 2W).

In equation (9) the parameter γ is used to distinguish between triangularly shaped and rectangularly shaped channels (γ has value of unity for triangular channels and 2 for rectangular channels).

No entrainment is considered for the upstream of the radial spread.

With an entrainment coefficient α = 0.1 (see Reference 9) the conservation of mass and energy equations are solved as a function of time numerically. The initial motion of the cloud is radial, and the entrainment velocity goes from a maximum at the cloud edge, to zero at the center. When the cloud edge reaches the banks, further motion is only down the channel, and the velocity from that point is maintained uniform at the computed value for any time for all channel sections downstream of a distance equal to half the channel width. The velocity is decreased to zero from that distance to the break location. The cloud properties at any point in time are computed as if the cloud were of homogeneous composition. The results are shown in Figures 1.4 and 1.5. Figure 1.4 details two cases. They are (1) an initial source of 100 lbm/sec. lasting 995 seconds followed by a flow of 30 lbm/sec. and (2) a constant flow of 100 lbm/sec. contained indefinitely. Figure 1.5 assumes an initial source of 568 lbm/sec. for the first 480 seconds, followed by a flow of 100 lbm/sec for another 515 seconds and a flow of 30 lbm/sec. for another 2750 seconds.

From the results of Figure 1.5 it is obvious that initial cloud heights, regardless of wind velocity are so high that atmospheric dispersion would be sure to occur.

Moreover, the 568 lbm/sec discharge has been conservatively assumed for 480 sec, whereas the real discharge will decay from the initial 568 lbm/sec to 100 lb/sec. Hence, the more realistic slumping results will be somewhere within the results presented in Figures 1.4 and 1.5. Figure 1.4 illustrates the effect of variability of source on cloud extent.

For constant flow rates there exists a wind velocity under which very long clouds could be predicted. The variability in flow from the break combined with the presumed constancy in wind velocity during the intervals of time considered remove the possibility of these very long clouds. In fact for constant channel width (300 ft), the longest distance down the channel forecasted for the postulated full break is between 8500 and 9500 ft.

In reality the Allens Creek depression has channel width at the 140 feet isocline ranging from approximately 300 feet to more than 550 feet approximately 4000 feet down channel from the pipeline crossing the Creek as shown in Figure 1.6. Figure 1.7 shows the distance which the cloud can reach for wind speeds of 6 fps as a function of channel width for flow rates, but of the break corresponding to those used in Figure 1.5. Note that the final cloud height is always sufficient to exceed the 145 foot isocline, hence the effective channel can be considerably larger than 600 feet after the cloud travels more than 2000 feet down the channel.

To compensate for the widening of the channel as the cloud height exceeds the 145 foot isocline, a rectangular channel of cross-sectional area roughly corresponding to the average cross-sectional area of the Allens Creek channel from its lowest point to the 145 foot isocline (depth of channel approximately equal to 25 feet so that cloud travel is overestimated) can be used to estimate the farthest reach of the cloud. Between Sections 14E and 11E of Figure 1.6 representing approximately 5000 feet of down channel distance the cross-sectional area up to the 145 foot isocline varies between 9000 square feet to 11200 square feet at Sections 14E and 13E to even larger values down channel. For instance, at 11E it is more than 17500 square feet. Thus, an average value can be taken to be linear between 9000 and 18000 ft^2 or 13500 ft^2. For the depth of 25 feet, the equivalent width of the rectangular channel is 540 ft. From Figures 1.5 and 1.7, therefore it is expected that the cloud resulting from the full break in the pipeline will not reach further than 6000 feet down the channel. 59

To assess the hazards to the plant from the homogeneous cloud resulting from the gravity slumpimg calculation, the total quantity of propane released to the point wherein the cloud reaches a homogeneous average detonable concentration of 2.8% is calculated. This represents 3.23×10^5 lbm of propane, the detonation of which by the methodology of Reference (1) would correspond to the detonation of 0.39 kT of TNT, which is roughly three times what had been conservatively estimated for the detonation of the atmospherically dispersed cloud. This detonation would pose no safety hazard to the plant.

The preceding gravity slumping analyses have assumed a considerable amount of an entrainment in the propane cloud. Experiments (See References 10 and 11) conducted with fluids of density differences not comparable to the problem at hand; i.e, $\rho_2/\rho_1 \ll 1$; whereas, $\rho_{C_3H_8}/\rho_{air} \approx 2.0$ initially and $\rho_{cloud}/\rho_{air} > 1$ always, indicate that actual entrainment of the lighter fluid may be considerably less. Therefore, the gravity flow of propane down the Allens Creek has been also addressed under a condition in which air entrainment is assumed to be negligible.

For this condition, the characteristic of the propane flow down the Allens Creek are determined from the following equations which have been adapted from those of Reference (11) for 1-D flow in a constant channel

$$r_m = 1.26 \left(g' \frac{Q}{w}\right)^{1/3} t \qquad \text{for } 0 \le t \le t^* \qquad (10)$$

$$h(r_m) = 0.794 \left(\frac{Q^2}{g' w^2}\right)^{1/3} \qquad \text{for } 0 \le t \le t^* \qquad (11)$$

$$t^* = 0.4 \left(\frac{Q}{g' w c^3}\right)^{1/3} \qquad (12)$$

$$r_m = 1.05 \left(\frac{Q^2 g'}{w^2 c}\right)^{1/5} t^{4/5} \qquad \text{for } t \ge t^* \qquad (13)$$

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$$h(r_m) = .954 \left(\frac{Q^3 c}{g' w^3}\right)^{1/5} t^{1/5} \qquad \text{for } t \ge t^* \qquad (14)$$

Therein w is the channel width, Q is the volumetric flow rate and t^* is the time at which flow changes from inviscid (i.e., no frictional effects), to viscous. The initial inviscid flow is due to the fact that the initial Froude number is larger than unity; i.e., the flow is supercritical, but approaches critical very quickly.

c is the frictional loss which can be expressed as

$$c = g \left(\frac{1.486}{n} h^{1/6}\right)^{-2} \qquad n = 0.075 \text{ for rough river beds} \qquad (15)$$

Also

$$g' = g \left(\frac{\rho - \rho_a}{\rho_a}\right) \qquad (16)$$

In equation (15) it is assumed that the width of the channel is sufficiently larger than the depth of the flow, h, so that the hydraulic radius is approximately equal to the depth, h.

For Allens Creek, the channel shape between the locations of the pipeline cross-over and the plant vicinity can be taken as square bottomed with an average width of 50 feet for a depth of 6 to 7 feet.

In the preceding equations, r_m is the distance downchannel of the front of the cloud at time, t.

Solution of the above sets of equations yields the following cloud front, distance and average depth as a function of time as shown in Figure 1.8.

It should be noted that to achieve the distances and depths given by Figure 1.8, it is necessary for the source of propane to continue for times longer than the time of flow when the line is isolated.

Two cases are then possible,

a) If the line is isolated the maximum depth will occur at a distance of 9330 feet downstream of the break location and will equal 6.3 feet. Therefore, the depth will decrease as the cloud continues slumping.

b) The line is not isolated and flow continues indefinitely. Equations 59
(10) through (14) are applicable only until steady uniform flow is achieved. The depth for such flow can be computed from Bresse's equation (See Reference 12)

$$\frac{dh}{dr_m} = \frac{i - b_1 v^2/R}{1 - \frac{v^2 w}{Ag}} \tag{17}$$

where i is the slope of the channel, A its cross-sectional area, R its hydraulic radius, v the flow velocity, and

$$b_1 = \frac{RC}{gh} \tag{18}$$

All other terms are as previously defined.

Equation (18) is entirely analogous to the equations of Reference (10) for flows with entrainment, which are reproduced below

$$\frac{dh}{dr_m} = \frac{(2 - \frac{1}{2} S_1 R_i)\ E - S_2 R_i \tan \alpha + C}{1 - S_1 R_i} \tag{19}$$

$$\frac{h}{3R_i} \frac{dR_i}{dr_m} = \frac{(1 + \frac{1}{2} S_1 R_i)E - S_2 R_i \tan \alpha + C}{1 - S_1 R_i} \tag{20}$$

where the Richardson number, R_i, is defined as

$$R_i = gh \frac{(\rho - \rho_a) \cos \alpha}{\rho v^2} \tag{21}$$

S_1 and S_2 are shape coefficients, E is the entrainment coefficient, and α is the angle of the sloping channel.

For steady uniform flow, and no entrainment, it is necessary that

$$R_i \tan \alpha = C \tag{22}$$

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where S_2 is taken as equal to 1.0 (i.e., no vertical acceleration of the fluid).

Equation (22) when solved yields a depth of 9.53 feet for a channel width of 50 feet.

Consideration of entrainment, by solving equations (19) and (20) would increase depth in the initial region of the cloud, where the Richardson number is less then unity. That number becomes rapidly larger than unity however, so that the effects of entrainment can be considered negligible (i.e., $E \rightarrow 0$ as $R_i \geq 1.0$) (See Reference 10).

The steady uniform flow depth of 9.53 feet, according to equations (10) through (14) would be reached at a distance of approximately 50,000 feet downstream of the break (well into the lake) and a time of 35000 seconds.

Thus it can be concluded that the depth of the cloud in the proximity of the Allens Creek plant will be in all likelihood of the order of 7 feet or less (less if the line is isolated) and the width of the cloud will be about 50 feet.

The cloud will be pure propane vapor (non-detonable). To determine the potential effects on the plant, the plant safety related structures of which are located 1800 feet away from the channel at its closest proximity, the amount of propane mixing with air and thus leaving the channel bed and being transported toward the plant by atmospheric dispersion, is determined from

$$\frac{k_m L}{c D_{AB}} = 0.36 \left(\frac{\mu}{\rho D_{AB}}\right)^{1/3} \left(\frac{\rho v L}{\mu}\right)^{0.8} \tag{23}$$

where k_m is the mass transfer coefficient, L the length from the break location, c the molar density of air (2.589×10^{-3} # mole/ft^3), v the cloud velocity which is about 2.2 fps, ρ the air density = 0.075 lb/ft^3, at 70°F, μ the air viscosity 1.05×10^{-5} lb/ft sec, and

$$D_{AB} = 2.745 \times 10^{-4} \left(\frac{T}{T_{CA}T_{CB}}\right)^{1.823} (P_{CA}P_{CB})^{1/3} (T_{CA}T_{CB})^{5/12} \cdot \quad (24)$$

$$\cdot \left(\frac{1}{M_A} + \frac{1}{M_B}\right)^{1/2} \ cm^2 \ sec.$$

where M_A = 44.09 lb

M_b = 28.97 lb

T_{CA} = 370°K

P_A = 42 atm

P_b = 36.4 atm 59

T_{CB} = 132°K

The heat transfer from the ground has been computed to be insufficient to heat the cloud too much above its -44°F temperature, as long as the source of propane is continuous. Hence the value of the mass transfer coefficient, k_m, and diffusivity D_{AB}, have been computed at -44°F, for air at 70°F. The length L has been chosen as 12,000 feet, this being the distance from the break to the plant along the channel.

Longer distances would result in slightly lower values of k_m.

The computed values of D_{AB} and k_m are

$$D_{AB} = 2.0 \times 10^{-4} \ ft^2 \ sec^{-1}$$

$$k_m = 2.8 \times 10^{-4} \ lb/ft^2 sec$$

Assuming an infinitely long line source of strength, $Q = 2.8 \times 10^{-4}$ lb/ft^{2}sec or 0.179 ft^2 /sec (using a width of 50 ft and a propane density of 0.107 lb/ft^3 at 70°F), the dimension of the detonable and the deflagrable cloud downwind of the source have been computed for a wind speed of 2.6 fps. The results are shown in Figure 1.9.

Assuming that the actual length of the cloud parallel to the plant and channel is no longer than the 26,000 ft separating the pipeline crossing location and the lake exit of the Allens Creek, the maximum detonable volume is computed to be 1.23×10^7 ft^3. This volume contains 64,410 lbs. of propane at an average concentration of 4.9 percent. Its detonation is equivalent to that of 1.55×10^5 lbs. of TNT.

Assuming that the center of the detonation is 400 feet from the channel, or 1400 feet from the plant nearest safety related structure, the peak overpressure to which the plant would be subjected is a tolerable pressure of 1.o psi.

It is therefore concluded that the propane flow down the Allens Creek would not present an unacceptable hazard to the plant safety related structures.

1.6 EVALUATION OF THE HAZARD OF THE DEFLAGRATION FROM LPG PIPELINE BREAK

A vapor cloud of flammable concentration may burn (deflagrate) or detonate (if within the detonable limits) or both types of combustion may occur in case of transition from deflagration to detonation. Volumetric explosions may also occur particularly if partial confinement of the cloud exists. More likely, pockets of gas in a cloud may explode volumetrically if heated to the autoignition limit by radiated heat or shock waves.

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In general, the deflagration, detonation, and volumetric explosions are the common modes of combustion to be considered. In our case, due to the unconfined nature of the cloud, the latter mode cannot occur, and it is only necessary to address detonations and deflagrations.

For detonations, all of the thermodynamic properties, detonation velocities and flow properties behind the detonation front are calculated from standard thermodynamic equilibrium calculations.

If U_1 and U_2 are defined as velocity of the unburned gas and burned gas with respect to the stationary detonation wave, then with respect to a stationary observer U is the detonation velocity and $W = U_1 - U_2$ is the velocity of the gas behind the detonation wave front.

Further, the thermodynamic states behind the detonation front are described by the Hugoniot equation

$$\Delta E = E_2 - E_1 = \frac{1}{2} (P_2 + P_1) (v_1 - v_2) \qquad (25)$$

Wherein p and v are the pressure and specific volume and E the energy and 1 and 2 denote the unburned and burned states, respectively.

The actual detonation involves a passage through a family of Hugoniot curve which proceed from the curve corresponding to no chemical reaction wherein $\Delta E = E_2 - E_1 = C_v(T_2 - T_1)$ (C_v is the average specific heat at constant volume between temperatures T_1 and T_2 before and after the passage of the wave front) to the curve corresponding to complete chemical reaction in which case

$$E_2 - E_1 = C_v (T_2 - T_1) - \Delta E_c \qquad (26)$$

wherein ΔE_c is the energy released in the combustion process.

Figure 1.10 shows the Hugoniot curves. The Hugoniot curve for the complete reaction is distinguished by two branches. The branch from A to V is the "detonation" branch and the one from B to C is the "deflagration" branch. It has been shown that the only possible stable detonation is the detonation proceeding at the minimum detonation velocity (see Reference 16). This is the Chapman Juguet detonation and the resulting detonation overpressure for stoichiometric propane air mixture has been determined to be $p_2/p_1 = 17.8$ from an initial state of 14.7 psi and $460^{o}F$.

That value is closely correspondent to that presented by J H Lee (see Reference 17). Point A corresponds to the overpressure resulting from an adiabatic constant volume explosion. The value obtained of $p_2/p_1 = 8.5$ differs slightly from that quoted in various references, including Reference (1), which reports 8.34. The difference is due to the initial state assumed for these calculations which was $0^{o}C$, whereas others generally used $25^{o}C$.

The intersection of the tangent line with the Hugoniot curve for no reaction is the Von Neumann spike which precedes the C - J detonation pressure. Its computed value is approximately $p_2/p_1 = 28$. 59

The detonation velocity which is given by

$$U_1 = D = v_1 \sqrt{\frac{P_2 - P_1}{v_1 - v_2}} \qquad (27)$$

is computed to be D = 5330 fps or 1610 mps, which is in reasonable agreement with literature data.

In the detonation branch the flow velocity of burned gas, W, is in the same direction as the detonation wave. In fact, it can be shown that (see Reference 18)

$$D = W + C \qquad (28)$$

where C is the velocity of sound in the burned medium.

Since the detonation wave travels supersonically with respect to the unburned medium, no disturbance precedes it, hence the cloud size remains at its initial size as the detonation propagates; i.e, the gas expansion occurs afterwards and has the effect of a rarefaction wave which tries to overtake the detonation wave at sonic velocity in the burned medium. This observation has been made by others (see Reference 17, page 12).

Thus, the effects of the detonation blast of the wave of the propane gas clouds has already been properly addressed in the preceding subsections.

The portion of the curve between A and B corresponds to no real physical state. The portion of the Hugoniot curve between B and C represents a decrease in pressure and increase in volume, corresponding to a rarefaction.

The burned gas flow velocity, W, therefore is always negative; i.e., the burned gas no longer moves in the same direction as the wave, but away from it.

The consequence of this is that in this deflagrative process a pre-compression wave is sent out into the explosive mixture to push that unburned gas with a velocity just sufficient to ensure that the gas may come to rest when it is swept over and burned by the deflagration front.

Physically no deflagration occurs past the point C (Reference 16) which is known as a Chapman Jouguet deflagration point. In fact, weak deflagrations (essentially constant pressure explosions) are those to be considered for vapor clouds explosions, and are the ones that have been observed in experiments.

Whereas the C - J detonation velocity represents the minimum of all detonation velocities, the C - J deflagration velocity is in fact the largest possible deflagration velocity, which is calculable from equation (27). 59

For the propane-air mixture studied (stoichimetric) the C - J deflagration velocity was computed to be 168 fps.

For weak deflagration, however, the deflagration velocity is the same as the laminar burning velocity, 1.5 - 2.0 fps, hence thousand times less than detonation velocities.

However, the spatial deflagration velocity, which would be that seen by a stationary observer is higher than the deflagration velocity computed by equation (27). This is due to the displacement of the unburned gases ahead of the propagating flame caused by the specific volume increase across the flame front. Since this increase is about eight-fold, the spatial deflagration velocity is roughly eight times that computed by equation (27). Hence, for a C - J deflagration spatial velocities of the order of 1300-1400 fps could occur. In actual tests however, (see References 1 and 19) the spatial flame velocity has been measured to be of the order of 30-40 fps, for mixtures difficult to detonate (like propane air) and ten times larger for mixtures rich in oxygen.

The effect of oxygen richness is not surprising since lack of inerts such as nitrogen has the effect of raising the Hugoniot curve for complete reaction to higher values. For instance, the C - J detonation point for a stoichiometric propane-oxygen mixture would correspond to overpressure almost exactly double that occurring for the detonation of propane-air mixtures, with detonation velocities 30 percent higher. Similarly, the C - J deflagration velocities for oxygen rich mixtures is expected to be higher than that for propane-air mixtures.

Hence, C - J deflagrations of any kind exhibit velocities in excess of 1300 fps. The fact that no such spatial burning velocity has been observed confirms that C - J deflagrations do not occur, but that only weak deflagrations; i.e., basically constant pressure burning occur.

For these kind of deflagrations, the precompression sent into the unburned medium and surrounding air is in the nature of basically an acoustic wave, which does not steepen into an air shock unless spatial flame velocities of 300 fps or more are achieved, as shown on Figure 1.11 which is taken from Reference (19).

Likewise, there is no overpressure of great significance within the deflagrating cloud. For typical spatial burning velocities of 30 fps or less, overpressure of about 1 psi will occur just ahead of the flame front. Hence, it is concluded that blast damages are insignificant for deflagrative burning of propane air clouds either near or far from the cloud. Hence, for a deflagration the possible damage is limited to temperature.

In a detonation event, after passage of the initial detonation blast, the compressed products expand. Under the assumption that this expansion is isentropic, the final volume will be approximately 9.7 times initial volume (assuming $k = 1.25$) (for an oxygen rich detonation these rates would be more than double). This expansion in turn can generate a second shock in the air ahead of the expanding products of the detonation, which follows the initial shock caused by hydrodynamic coupling of the detonation wave and the air.

This second shock is one order of magnitude smaller than the first air shock (see Reference 20) and exhibits the same decay with distance from the center of detonation as the first and much stronger shock. 59

Since the Allens Creek plant has been shown to withstand the first shock overpressures, it is also safe against the weaker second shock caused by the expanding detonation products.

During the preceding it has been tacitly assunmed that the flammable cloud is entirely formed of a stoichiometric mixture of propane and air (or oxygen). This was of course the case for all experiments conducted (see References 17, 18, 19 and 20). In fact, the propane air cloud computed to occur as a result of atmosphere dispersion exhibits a range of mixtures which are only stoichiometric in a region which at ground level is centered about the 1300 ft and 3500 ft from the origin of the cloud. At closer distances to the origin of the cloud, ground level concentrations are over rich and farther away they are leaner than stoichiometric.

Whereas the detonable limits for propane air mixtures are 7.0 and 2.8 percent by volume of propane, the limits of deflagration, are 9.0 and 2.2 percent.

The volume of flammable mixture contained within the vapor cloud is computed to be 1.96×10^7 ft^3, whereas the detonable volume is only 1.155×10^7 ft^3.

Hence, 40 percent of the total flammable volume has concentrations below 2.8 percent, or below 0.7 times the stoichiometric concentration of 4.12 percent.

The expansion of a mixture of less than 67 percent stoichiometric concentration (as well as that of concentrations about 120-130 percent of stoichiometric mixtures is at least 30 percent less than the expansion of stoichiometric mixtures. This results basically from the lower flame temperature. The expansion given on Figure 1.10 as point B is thus applicable only to stoichiometric concentrations. Conservatively therefore, the final volume of the deflagrated vapor cloud can be estimated by expanding the volume of

2.8 percent or larger propane concentration to a final volume eight times larger, and the volume of lower concentration to a volume 5.5 times larger. The resultant final volume would be 1.35×10^8 ft^3.

Assuming that the original cloud can be represented as a hemi-ellipsoid of 4564 ft, 143 ft, and 28.6 ft dimensions in the downwind, crosswind and vertical direction, centered at 2734 ft, from the break, and that the deflagration is centered at that point, the final dimension of the product cloud downward will be either 6716 ft from the break.

The closest distance from the 6 inch pipeline to a plant safety related structure in Allens Creek is 8000 feet. Hence, it is concluded that the deflagration of the "worst" cloud ensuing from a catastrophic break in such a line poses no hazard to the plant.

Finally we have to examine the deflagration of the homogeneous cloud calculated by gravity slumping. Again assuming an expansion of 5.5 times for the lower temperatures resulting from the low concentrations applicable to the cloud, the length of the cloud will have increased 1.75 times. Hence, the burned cloud can reach a distance which is approximately 2250 feet farthe down channel than its farthest reaches when unburned. 59

Since the cloud slumps by gravity, it follows the channel. 6000 feet down channel puts the furthest reaches of the cloud 3000 feet from the plant safety related structures and 2400 feet from the switchyard. Thus the burned cloud cannot reach plant safety related structures although it comes close to the switchyard, and the plant is safe.

1.7 GRAVITY SLUMPING EFFECT OF LPG CLOUD FORMED FROM SMALL BREAK OF THE PIPELINE

The following scenarios to be examined are the gravity slumping of cloud formed from small leaks or breaks in the line. Analyses of the Ruff Creek incident (see Reference 21) wherein the cloud resulting from a 20.5 lb/sec constant propane source was subjected to gravity slumping, indicated that the distances from the break which homogeneous clouds of 2.4% concentration are computed to reach are in general lesser than those calculated for larger flows out of the break, although wind conditions can make a cloud resulting from a smaller flow go farther than that resulting from a higher flow, there will be another wind condition at which the latter will go even farther.

This is shown by Figure 1.12 which compares cloud distances and final heights for breaks resulting in constant 100 lbm/sec and 20.5 lbm/sec in like channels.

Since even for small breaks the source of propane is expected to vary in time as the pressure in the line drops, a similar behavior of distance vs. wind velocity as exhibited in Figure 1.5 is expected for the smaller breaks as for the large breaks. Thus, a small break presents less hazard from the gravity slumping standpoint than the large break.

1.8 GRAVITY SLUMPING EFFECT OF THE EQUILIBRIUM ATMOSPHERICALLY CLOUD

The gravity slumping of the equilibrium atmospherically dispersed cloud depicted in Figure 1.3 has been evaluated by considering that the continuously fed cloud begins its gravity slumping from an elongated shape cloud which has an initial length equal to approximately 2500 feet, a width of 150 feet, and a height of 15 feet; instead of the pancake shape having the r_{max} and h_{in} assumed previously. This represents the over rich region of the cloud which, when slumping under its own weight or because of ground slope, can entrain surrounding portions of the cloud which are already flammable or below flammable, and thus dilute its concentration by mixing to levels which are flammable.

In turn, the region which is flammable initially also slumps and entrains non-flammable mixture thus diluting its concentration.

The velocity of the slumping, initially over rich volume of the cloud in the absence of significant slope in the ground is computed by assuming an average concentration of approximately 40% propane. The density of this mixture is about 12% higher than the surrounding mixture which is taken to be at an average concentration of 5%. The velocity of the front of this cloud is computed to be approximately 10.3 fps. The amount of mixture at 5% concentration entrained per second at this velocity is approximately 1.8×10^5 ft^3/sec. 59

The velocity of the 5% volume of the cloud which is slumping is only 5 fps, and the entrainment of non-flammable mixture at this velocity with a width of 300 feet is computed to be about 1.2×10^6 ft^3/sec.

Of course the front velocity of the over rich region of the cloud slows down from the initial 10 fps to 5 fps as the mixing drops its concentration to a concentration comparable to that of the initially flammable cloud, while the latter front velocity drops to essentially zero when its concentration drops to very low values.

Assuming a linear variation in front velocity from the time at which the slumping begins to when it ends, the over rich cloud which had an original volume of 1.93×10^6 ft^3 will have entrained essentially all of the 5% concentration gas that hasn't been diluted by its own slumping in approximately 2 minutes, to form a volume of approximately 1.35×10^7 ft^3 having a concentration which averages about 10% by volume, meanwhile, at the same time, the average concentration in the previously flammable volume will have dropped to less than 0.7%. The cloud will have advanced approximately 1000 feet down the channel. The fraction now computed to be an average 10% concentration will further slump and advance with a velocity varying from 6 fps to zero while entraining the surrounding less rich mixture. Assuming an average surface area of entrainment of 5.4×10^5 ft^2 (interface between 10% region and surrounding less rich region), the concentration will be dropped to about 5% in approximately 80 to 90 seconds, during which the flammable portion of the cloud will have advanced another 500 feet toward the plant.

Thus the slumping of the equilibrium cloud formed by atmospheric dispersion has the effect of moving the flammable and detonable region closer to the plant by about 1500 feet.

The preceding approximate calculation inherently makes the same assumptions that are made in Section 1.2.

As such the cloud resulting from the slumping of the one depicted in Figure 1.3 is itself a very conservative upper bound of physically possible clouds. The detonation of such cloud, now located 1500 feet closer to the plant, would still pose no hazard.

More realistically, vapor clouds of smaller dimension can be expected. For instance, isolation of the line at the Sheridan Station and John Sue Junction limits the maximum vapor cloud size to sixty percent of that shown in Figure 1.3. If such a realistic cloud were to move 1500 feet closer to the plant, then deflagrate, the farthest reaches of the burn marks would extend to a distance of 7150 feet from the break location, and hence pose no undue hazard to the plant safety related structures. 59

1.9 ATMOSPHERIC DISPERSION OF CLOUD INITIALLY MOVING BY GRAVITY

The scenario to be examined is that of gravity slumping followed by atmospheric dispersion.

Except for the analyses performed for the low wind velocities, it was shown that the cloud velocity during gravity slumping falls below the wind speed rather soon. From that point on, the large clouds can only be obtained by assuming that atmospheric dispersion does not occur. This would, of course only happen if the air flow is laminar (i.e., no disturbance), a condition which is extremely unlikely.

Hence, at the point in which the cloud speed falls to the level of the wind speed, the more realistic assumption can be made that atmospheric dispersion begins. The concentration downwind can then be obtained from the equation for a continuous line source

$$\chi = \frac{1}{2}\sqrt{\frac{2}{\pi}} \quad \frac{Q}{L\bar{u}\,\sigma_z} \exp\left(-\frac{z^2}{2\,\sigma_z^2}\right)\left[\operatorname{erf}\frac{(L/2 - y)}{\sqrt{2}\;\sigma_y} + \operatorname{erf}\frac{(L/2 + y)}{\sqrt{2}\;\sigma_y}\right] \tag{29}$$

in which Q is the source strength in ft^3/sec, $\bar{u}$ the wind speed, L the width of the source which is taken to be that of the cloud spread across the wind at the point in time at which the cloud speed equals the wind speed.

For points along the wind direction, y = 0, and equation (29) for ground level reduces to

$$\chi = \sqrt{\frac{1}{2}} \quad \frac{2}{\pi} \quad \frac{Q}{L\bar{u}\;\sigma_z} \quad \operatorname{erf} \frac{L/2}{\sqrt{2\,\sigma_y}} \tag{30}$$

We have calculated from the gravity slumping model the time at which the cloud advancing velocity falls below the wind velocity. We have also calculated the location and the concentration (which is of course homogeneous). To determine the atmospheric dispersion of the cloud from that point on we resort to a line source having a length, L, located perpendicular to the wind direction, and located at a fictitious (virtual) distance, upwind from the point at which the cloud and wind velocity have been compared to be equal, which is equal to that which would result in a centerline concentration equal to the homogeneous cloud concentration computed by the gravity slumping model. This is shown in Figure 1.13.

The location of the virtual line source is chosen upwind of the point at which the cloud and wind speed become equal, by solving the following equation

$$\left(\bar{u} = dR/dt\right) = \sqrt{\frac{2}{\pi}}\ \frac{Q}{\bar{u}L\ (\bar{u} = dR/dt)}\sigma_z\ \mathrm{erf}\frac{L/2}{\sqrt{2}\,\sigma_y} \tag{31}$$

where $\chi_{(u=dR/dt)}$ is the computed concentration, in volume fraction, at the time in which the wind velocity equals the cloud velocity. In equations (29), (30), and (31) the downwind distance does not appear explicitly, but it is found by trial and error solution for σ_z and σ_y, the dispersion parameters which are functions of that distance. For a Pasquill F meteorological condition, assumed throughout this study, values of the dispersion parameters σ_z and σ_y with downwind distance are taken for two cases. In case A, $\sigma_y = 2\sigma_z$; and in case B, $\sigma_y = 5\sigma_z$. 59

The very same equation (29) has also been used to back calculate the width of a source and its distance upwind that would produce concentrations of 2.4% of propane at a point on the ground located downwind on a line bisecting the line source. The results for the Allens Creek plant are shown in Figure 1.14 for different wind speeds.

Figure 1.14 can be used for other source strengths if it is remembered, that in equation (29) $\frac{Q}{\bar{u}}$ is a single parameter. Hence, the effect of a smaller source is like that of increasing velocity; i.e., a source of 100 lb/sec with a wind of 2.6 fps will give the same results as a source of 20 lb/sec in a 0.5 fps wind.

Figure 1.14, when used for Allens Creek, and a break giving 100 lb/sec. of propane, is limited to cases in which the wind velocities are in excess of 4.3 fps. This is because the equilibrium cloud velocity for such a constant break flow, never falls below 4.3 fps. This is illustrated in Figure 1.4, where for wind speeds exactly matching the cloud advancing speed (i.e., no entrainment) the cloud is seen to advance to very large distances. For wind speeds below 4.3 fps, it is assumed that no atmospheric dispersion occurs. For wind speeds equal to or in excess of 4.3 fps, it is assumed that gravity slumping ceases and atmospheric dispersion takes over.

For breaks producing lower flow rates (such as for instance, a break resulting in 20.5 lb/sec.), the speed of wind at which atmospheric dispersion can be assumed to begin is lower. As shown in Figure 1.12 for such a break, the equilibrium cloud speed is about 2.6 fps. Hence, Figure 1.14 for such a break should be used for wind speeds in excess of 2.6 fps. Since Figure 1.14 has been developed for a flow rate of 100 lb/sec. and the flow rate in question is only 20.5 lb/sec, direct use of Figure 10 for such a flow, requires the use of a corrected wind speed of = $\frac{100}{20.5}$ 2.6 = 12.7 fps.

Returning to the case of Allens Creek with a break resulting in 100 lb/sec. flow of propane, it is noted that during gravity slumping, the speed of the cloud falls to or below a wind speed of 4.3 fps very soon (i.e., at distances from the break of less than 400-500 ft.). At such point the homogeneous cloud concentration is below 20-30%. A line source producing such concentration would have to be located about 1500 ft upwind from that point, i.e, 1000 ft. further upwind of the break location.

From Figure 1.14, therefore, one would compute that concentration of 2.4% of propane would be achieved about 3000 ft. from the virtual line source location (wind speed $>$ 4.3 fps) for the case of $\sigma_z = .2\ \sigma_y$, and hence, 4000 ft. from the break location.

Thus, the resulting cloud would be less severe than that examined in Figure 1.3.

1.10 POTENTIAL MISSILES FROM DETONATION

It is difficult to assess the potential hazard to plant critical structures resulting from missiles generated by the detonation either at the initial crater or propelled by the blast wave. Since the LPG line crosses the plant vicinity in an open area, there is little likelihood that substantial missiles may be generated other than from the place where the detonation is postulated to occur. 59

Work by Ahlers 23 / on observed maximum debris distance and equivalent yield, which is reproduced in Figure 1.15 shows that the range of missiles from the 390 tons detonation would be on the average of 6000 feet, with source having a range of up to 12,000 feet. Hence the detonation of the vapor clouds could result in missiles reaching the plant.

The missiles which travel the larger distances, however, are expected to be the smaller since air drag will affect the larger missiles proportionally more. Studies on several detonations 24, 25 / have shown that the size distribution of ejecta (missiles) follows an exponential law.

$$\delta = kr^{-3.5}$$

where δ is the areal density and r the distance from the detonation center.

To ascertain what the probability is of a missile from even the impossible maximum detonation of the propane (390 tons) hitting critical plant structures, we assume that the total weight of missiles is proportional to the volume of the crater which would be created by the detonation, had it been a TNT detonation near the surface. The crater in turn is proportional to the detonation yield. Roughly the volume of the crater can be estimated by the scaling law.

$$\text{Diameter (depth) of crater} = \left(\begin{array}{l}\text{Diameter (depth) of crater}\\ \text{for a 1 KT explosion}\end{array}\right) \times W^{1/3}$$

and the knowledge that a 1 KT surface detonation in dry soil results in diameters of 180 feet and depths of 35 feet.

Hence the total mass of ejecta at 95 lb/ ft^3 will be:

$$95 \text{ lb/ft}^3 \left[2\pi /3 \, (180 \times .73)^2/4\right] \, (35 \times .73) \text{ ft}^3 = 2.19 \times 10^7 \text{ lb}$$

Assuming none of the mass falls back within the crater, then the total mass must be given by

$$M = \int_{80} 2\pi r \delta \, dr = \int_{80} 2\pi\kappa \; r^{-2.5} dr = .00786\kappa$$

$$\kappa = 2.79 \times 10^9 \text{ lb-ft}^{3/2}$$

Hence the areal density 7,000 feet away from the center of the hypothetical (impossible) 390 ton detonation is

$$\delta = \frac{2.79 \times 10^9 \text{ lb-ft}^{3/2}}{4.58 \times 10^{13} \text{ ft}^{7/2}} = 6.09 \times 10^{-5} \text{ lb/ft}^2$$ 59

Since the area of critical structures is of the order of 10^5 ft^2, the total weight of missiles hitting these areas for the hypothetical maximum detonation would only be 6 lb.

Assuming that all of this mass is concentrated in one missile, and that missile travels at the maximum air particle velocity given by

$$u = \frac{5 p \quad c_o}{7P_o \, (1 + 6p/7P_o)} = 53 \text{ fps}$$

where p is the peak overpressure at the critical structure (1 psi), P_o the ambient pressure (14.7 psi) and C_o the ambient sonic speed, taken as 1,130 ft/sec, then the impact energy of this missile is 26 ft-lb, which is considerably below the energy required for penetration of the structures.

It is concluded therefore that missiles from the propane cloud detonation would present no hazards to the Allens Creek Nuclear Island.

1.11 COMPARISON OF MODELS WITH ACTUAL OBSERVATIONS

a) Ruff Creek (See Reference 26)

The actual wind condition during the Ruff Creek event is not reported. It is deduced to have been less than 23 fps; since this agrees with the description given that the wind was calm.

From Figure 1.12, further it can be deduced that the cloud spread should have been 300 by something less than 2500 feet if it occurred by gravity alone, which would agree well with the burn marks observed to be roughly 100 yards wide by more than one mile long, since the deflagration would have resulted in an expansion to roughly six to eight times the volume. Hence burn marks would appear at twice the distance computed for the cloud. The steady cloud speed exceeds the wind speed so long as the wind speeds are below 2.3 fps, except right at the beginning of the event.

If atmospheric dispersion is assumed to start at that instance of time and gravity spreading is arrested, similar results are obtained. A calculation of the extent of the cloud with Figure 1.14 yielded a length of the cloud equal to approximately 2000 feet.

b) Devers, Texas (See Reference 27)

Atmospheric dispersion started at the point at which gravity spreading velocities fall below the wind speed of 8 fps, yield downwind distances of 2800 feet, which when coupled with further lateral spreading of the cloud to the 2.4% concentration, give a cloud a dimension of 2800 feet by 1200 feet wide. The model overestimates the longer dimension which was observed to be roughly 1100 feet. 59

Pure gravity flow, coupled with wind translation (no dispersion), with suppression of spreading upwind and enhancement downwind, would result in a cloud of 1200 feet width and 2600 feet length. The upwind front part of the cloud, however, must be more diluted in reality than calculated as a result of increased entrainment.

c) Austin, Texas (See Reference 28)

Because the wind speed is always above the cloud gravity induced velocity, atmospheric dispersion takes place right away. The atmospheric dispersion model yields a cloud dimension of approximately 2000 feet by 180 to 200 feet which compares well with the observed 2400 x 200 foot cloud. Gravity flow in a channel 200 feet wide with no atmospheric dispersion would yield a cloud length of 1300 feet.

2.0 (THIS SECTION IS INTENTIONALLY DELETED)

3.0 BREAK OF 24" NATURAL GAS LINE

The consequences of a complete severance of the 24 inch natural gas line have been evaluated on the basis of the assumptions listed in the following sections.

3.1 CALCULATION OF FLOW RATE OUT OF THE BREAK

a) Double ended rupture of the 24 inch pipe.

b) Gas escapes from both ends of the ruptured line.

c) Stagnation pressure upstream of break in both ends is the maximum operating pressure (not design) of the line, ie, 750 psi. Pressure can vary up to 900 psi, but the analysis is not very sensitive to the pressure assumed.

d) The discharge from the break is divided into: initial transient when the flow from both broken ends may be choked, and steady flow (due to continued pump operation).

Gas line operates at a maximum flow rate of 2.5×10^5 MCF/day, ie 2.5×10^8 ft^3 (@ standard temperature and pressure) /day = 9.25×10^6 lb/day. 41 Q 312.5

$$\text{since } \rho_{CH_4} = .037 \text{ lb/ft}^3 \text{ (@ standard temp. and pressure)}$$

(See Figure 3.1)

Thus the flow in the unruptured line to deliver such quantity will be

$$\frac{9.25 \times 10^6 \text{ lb/day}}{2.1 \text{ lb/ft}^3} = 4.4 \times 10^6 \text{ ft}^3\text{/day} = 51 \text{ ft}^3\text{/sec}$$

where 2.1 lb/ft^3 is the density in the line at 750 psi & 105° F (see Figure 3.1).

Thus to deliver 2.5×10^5 MCF/day, the gas is pumped through the pipeline at 750 psi (105° F) with a velocity of 16.25 ft/sec.

To determine whether choked flow exits, the natural gas is assumed to behave as an ideal gas, even though this assumption can lead to errors of nearly 50 percent in the estimation of densities in the conservative direction.

For an ideal gas, the critical pressure, at which the mass flow rate in the break area cannot be increased regardless how low the back pressure is made is given by: 2/

$$P^* = \left(\frac{2}{k+1}\right)^{k/(k-1)} P_o$$ where P_o is the upstream stagnation pressure

The critical density is given by

$$\rho^* = \left(\frac{2}{k+1}\right)^{\frac{1}{k-1}} \rho_o$$

Thus for the stagnation pressure of 750 psi, choked flow will exist as soon as the exit pressure is dropped below 410 psi. Thus choked flow will exist at the break. The shock wave which is created at the exit plane at the instant of the break will travel back through the line (both ends) at sonic speed until the whole line to the pump stations has been decompressed. At this time the flow can be assumed to unchoke. 41 Q 312.5

To calculate the maximum flow rate out of the break, the equation of continuity at the critical (choking) plane can be used, thus

$$w \text{ (lb/sec)} = \rho^* V^* A$$

where ρ^*, V^*, and A^* are the critical density, velocity and area respectively.

For an ideal gas

$$V^* = \sqrt{2kg_c RT_o / (k+1)}$$

and $A^* = A$ for Mach No = 1.0 (choked flow).

A 105° F, using $k \simeq 1.3$ for natural gas, and $R \simeq 60 \frac{\text{ft-lbf}}{\text{lbm - R}}$,

$$V^* = 1{,}110 \text{ ft/sec}$$

Using the ideal gas law, the stagnation density corresponding to 750 psi and 105° F, is

$$\rho_o = \frac{P_o}{RT_o} = \frac{750\ (144)}{60\ (565)} = 3.18$$

$$\rho^* = 0.629\ (3.18) = 2.0$$

Note that the actual density for methane $\rho_o = 2.1$ thus ρ^* is really in the neighborhood of 1.35; hence the estimated maximum flow rate using ideal gas law will be conservative by more than 30 percent

$$W = (2.0)\ (3.14)\ (1.11 \times 10^3) = 6{,}910\ \text{lb/sec}$$

The maximum mass flow rate can also be evaluated from 1/

$$\frac{W}{A}* = \sqrt{g_c \ \frac{k}{R} \left(\frac{2}{k+1}\right)^{\frac{k+1}{k-1}}} \ \frac{P_o}{\sqrt{T_o}}$$

Substitution of the proper values would also yield 6,910 lb/sec.

If true densities had been used, the maximum flow rate would have been roughly 4,800 lb/sec. This last figure is obtained by using a critical density of 1.37 instead of 2.0 in the continuity equation. (Initial flow would be higher for 900 psis pressure by roughly 20 percent.) 41 Q 312.5

Assuming that the break occurs midway between pumping stations, and that pumping stations are separated by 65 miles, the time required for complete decompression of both ends of the ruptured line is calculated as:

$$\frac{32.5 \text{ mi } (5280 \text{ ft/mi})}{1{,}100 \text{ ft/sec}} \simeq 156 \text{ sec} \simeq 2.5 \text{ minutes}$$

Thus for the first 2.5 minutes following the break it may be assumed that both ends discharge natural gas at a rate of 4800 lb/sec for a total of 9,600 lb/sec. Thereafter only the pumped end will discharge (barring pump trips) at a rate of roughly 108 lb/sec.

In reality the upstream pressure will decrease with time with a corresponding decrease in the flow rate out of the break. Further because of friction effects, the maximum flow rate out of the break will be lower, hence the model assumed is known to be conservative. The following figure compares the model used with what is expected in reality.

3.2 CALCULATION OF DETONABLE CLOUD SIZE

The dimension of the detonable plume downwind of the break have been evaluated for a Category F stability, and a constant, invariant wind speed of 2.6 ft/sec.

For purposes of conservation buoyancy effects have been ignored in this section. Buoyancy and the jet momentum are considered in Subsection 3.5.

The centerline (directly downwind) concentration of the methane (excluding buoyancy) is determined by:

$$\chi_{cl} = \frac{Q}{\pi \sigma_y \ \sigma_z u}$$

Off-centerline concentrations are determined by 41 Q 312.5

$$X = X_{cl} \text{ Exp } \left\{ -\frac{1}{2} \left[(y/\sigma_y)^2 + (z/\sigma_z)^2 \right] \right\}$$

Since the flow out of the break varies with time it is necessary to discuss which flow out of the break is chosen for subsequent analysis.

During the initial 2.5 minutes of the event, flow out of the break is essentially sonic. Although a large portion of the total mass of gas is released during this time period, (i.e. 70 percent of the total quantity of gas emitted from the break in the 2.5 hrs. estimated to be required to terminate flow is released during the initial decompression period), the velocity at the break, coupled with the buoyancy of the gas will propel it away from the surface and past low atmospheric inversions so that less than 500 lbs. of this initial mass is calculated to fall within the flammable limits in the vicinity of the surface, but most of it will be dispersed in the upper levels. 56

To establish therefore the configuration of a credible, low lying detonable cloud in the vicinity of the plant (ie, cloud travelling toward the plant under worst meteorological conditions), it is assumed that the flow out of the break is a constant 108 lb/sec, corresponding to the condition existing after the initial 2.5 minutes transient. Buoyancy was neglected in this calculation. 41 Q

The potential cloud configuration (neglecting buoyancy) is plotted in Figure 3.3a. The dashed portion represent the fraction of the cloud which falls within the flammable limits (4.8 and 14.0 volume percent). 312.5

The volume of this cloud is calculated to be:

Volume of flammable cloud = [Volume of ellipsoid at 4.8% = $2\pi/3$ (2200)(106)(53)] -

[Volume of ellipsoid at 14.0% = $2\pi/3$ (1150)(62)(30)] =

$$= 2.11 \times 10^7 \text{ ft}^3$$

Density of 10% mixture = .069 lbm/ft^3

Weight of cloud (assumed to be all at 10%) = 1.48×10^6 lbm

3.3 CALCULATION OF EFFECTS FROM DETONATION TO CATEGORY I STRUCTURES

From an enthalpy of detonation release of 1225 $\frac{\text{BTU}}{\text{lbm}}$ = 310 $\frac{\text{kcal}}{\text{lbm}}$ of 10 methane and mixture, and a total volume of 2.11×10^7 ft^3, it is possible to compute the total energy released in a hypothetical detonation of the entire detonable cloud, assuming that the whole cloud is at a mixture averaging 10 percent.

$$\frac{2.11 \times 10^7 \ ft^3}{[(.9)(12.4) + (.1)(27.0)] \ \frac{ft^3}{lb}} = 1.48 \times 10^6 \ lbm$$

$$\text{Total energy released} = (1.48 \times 10^6 \ lbm)\left(310 \ \frac{Kcal}{lbm}\right) = 4.60 \times 10^8 \ Kcal$$

$$\text{Equivalent TNT} = \frac{4.60 \times 10^8 \ Kcal}{500 \ \frac{kcal}{lb \ TNT}} = .90 \times 10^6 \ lb \ TNT = 450 \ tons \ TNT = .45 \ KT \ of \ TNT$$

41 Q 312.5

The preceding assumes that the detonation yield will be 100 percent. In actuality this will not be the case. Reference 2 cites yields of 7.5 percent. Work by Iotti, et al, shows that indeed the yield of a gaseous detonation is lower than that of TNT. Reference 6 compares overpressures calculated by assuming gaseous point sources (Reference 9) to overpressures obtained by Kingery (Reference 8) for the same yield, and those measured by Kogarko (Reference 7), et al, for the given gaseous detonation. This comparison shows that the Kingery's results would have been comparable to those of References 7 and 9 if a TNT yield of less than 50 percent had been employed. Thus a conservative estimate of the TNT equivalent of the detonation of the entire cloud can be obtained by using 50 percent yield and Kingery charts.

A yield of 50 percent is conservatively used in computing all the blast wave, and seismic parameters at the plant side. These are tabulated in the following table for distances of 3,600 feet, 5,400 feet, and 10,600 feet from the plant critical structures to the closest point of the detonable cloud (which is conservatively assumed to be the center of the detonation). The breaks are assumed to occur at the point in the line closest to Category I plant structures (9,300 feet for the relocated line).

Assuming conservatively that plant critical structures can withstand 1.0 psi overpressure the closest distance between the 24 inch line and any of those structures should be approximately 7,200 feet.

3.4 POTENTIAL MISSILES FROM DETONATION

The potential hazards of plant critical structures resulting from missiles generated by the natural gas detonation have been evaluated. From Ahlers work (Figure 1.15) one can argue that missiles could reach the plant structures, since the farthest range for the 225 ton of TNT detonation is shown to be of the order of 11,000 feet.

Moreover, again assuming that no mass falls within the expected crater (97 ft. diameter and 18.2 feet depth), the total mass of the ejecta, 8.5×10^6 lbm, is distributed about the crater in an areal density given by:

$$\delta = 6.9 \times 10^8 \ r^{-3.5} \ lb/ft^2$$

Hence at the 10,600 feet distance, the areal density is expected to be

$$\delta = \frac{6.9 \times 10^8}{(1.06 \times 10^{43.5})} = \frac{6.9 \times 10^6}{1.22 \times 10^{14}} \quad 51.62 \times 10^{-6} \text{ lb/ft}^2$$ 41 Q 312.5

For a distance of 5,400 feet, the corresponding areal density is expected to be 6×10^{-5} lb/ft^2.

Hence for a critical plat area of 10^5 ft^2, the total mass hitting the plant is less than 6 lbs, which even if concentrated all in one missile travelling at the air particle speed, calculated to be 24.6 fps, cannot damage the structures.

3.5 EFFECTS OF BUOYANCY AND MOMENTUM

Since methane is lighter than air, the methane-air mixture formed from the escape of methane from the break will rise with respect to the surrounding air or denser mixture in accordance to Archimedes law. In the previous sections buoyancy has been neglected for the sake of conservatism. Also neglected has been the large momentum of the methane gas out of the break, which would result in a high plume.

In spite of neglecting these effects it has been shown that the plant will withstand the hypothetical detonation of the ensuing cloud without deleterious effects to Category I structures.

4.0 BREAK IN THE 8-INCH CRUDE LINE

The consequences of a break in the crude oil lines may be conservatively assumed by comparison with the consequences of a break in the 6 inch LPG line which is identically routed. 41 Q 312.4

The major difference between the crude oil lines and the LPG line stem from the lower design and operating pressure of the crude line (720 psi vs 1,000 psi design and 560 psi vs 750 psi operating). The size of the line (8" vs the 66" LPG), and operating flow (12,000 barrels/day vs 8,500 barrels/day).

An identical time is required to shut off flow to either pipe line after a break is identified.

Even though proportionately more fluid can escape from the break of the crude oil line than the LPG line, a smaller fraction of the crude vaporizes to give rise to potentially detonable clouds. Therefore the consequences of a break in the crude oil line will not have an adverse effect on the plant's Category I structures.

TABLE 1-1

DIMENSION OF PROPANE PLUME DOWNWIND OF 1000 FT^3/SEC SOURCE, (6 INCH LPG LINE)

STABILITY CATEGORY F, WIND SPEED 2.6 FT/SEC

Downwind Distance (ft)	Plume Dispersion (Std Deviation) ft		Centerline Concentration	Horizontal* Distance from Plume Axis, ft.				Vertical* Distance from Plume Axis, ft			
				To rich limit		To lean limit		To rich limit		To lean limit	
	y	z	%	yrd	yrf	yld	ylf	zrd	zrf	zld	zlf
150	27	5.4	83.9	60.2	37	70.4	77.86	12.0	11.4	14.0	14.6
1000	36	7.2	47.2	70.4	65	85.6	89.14	14.0	13.0	17.[illegible]	17.8
2000	72	14.4	11.8	73.6	53	172.2	131.2	14.7	10.5	24.4	26.4
2170	82.5	16.5	9.0	58.5	0	126	138.5	11.7	0	25.2	27.7
2500	90	18.0	7.56	35.3	--	126.8	141.8	7.06	--	25.4	28.3
2575	93.5	18.7	7.0	0	--	126.5	142.2	0	--	25.3	[illegible]3.4
3000	100	20	6.[illegible]2	--	--	125	143.6	--	--	25.0	28.6
3500	122	24.4	4.11	--	--	106.9	136.4	--	--	21.4	27.2
4000	138	27.7	3.5	--	--	92.2	133.0	--	--	18.4	26.6
4290	147.8	29.6	2.8	--	--	0	102.6	--	--	0	20.5
5000	165	33	2.25	--	--	--	35	--	--	--	7.0
5016	1[illegible].8	33.4	2.2	--	--	--	0	--	--	--	0
5500	181.5	36.3	1.86	--	--	--	--	--	--	--	--

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* rd - rich detonable limit   rf - rich deflagration limit
ld - lean detonable limit   lf - lean deflagration limit

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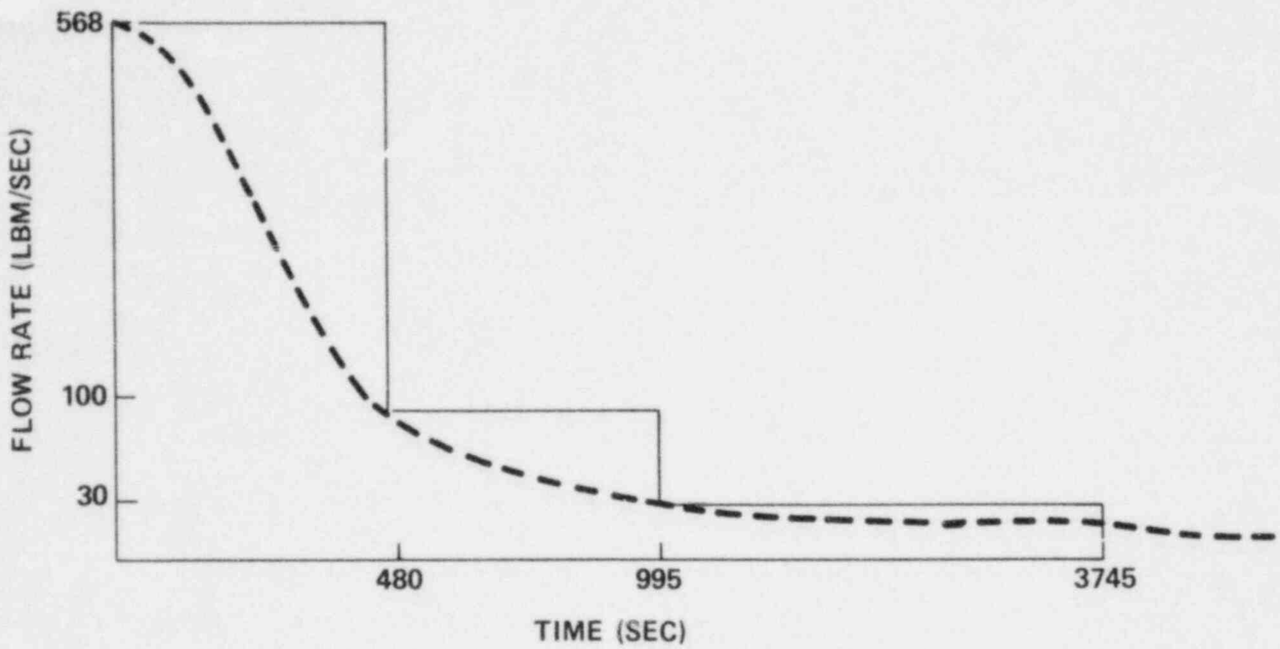

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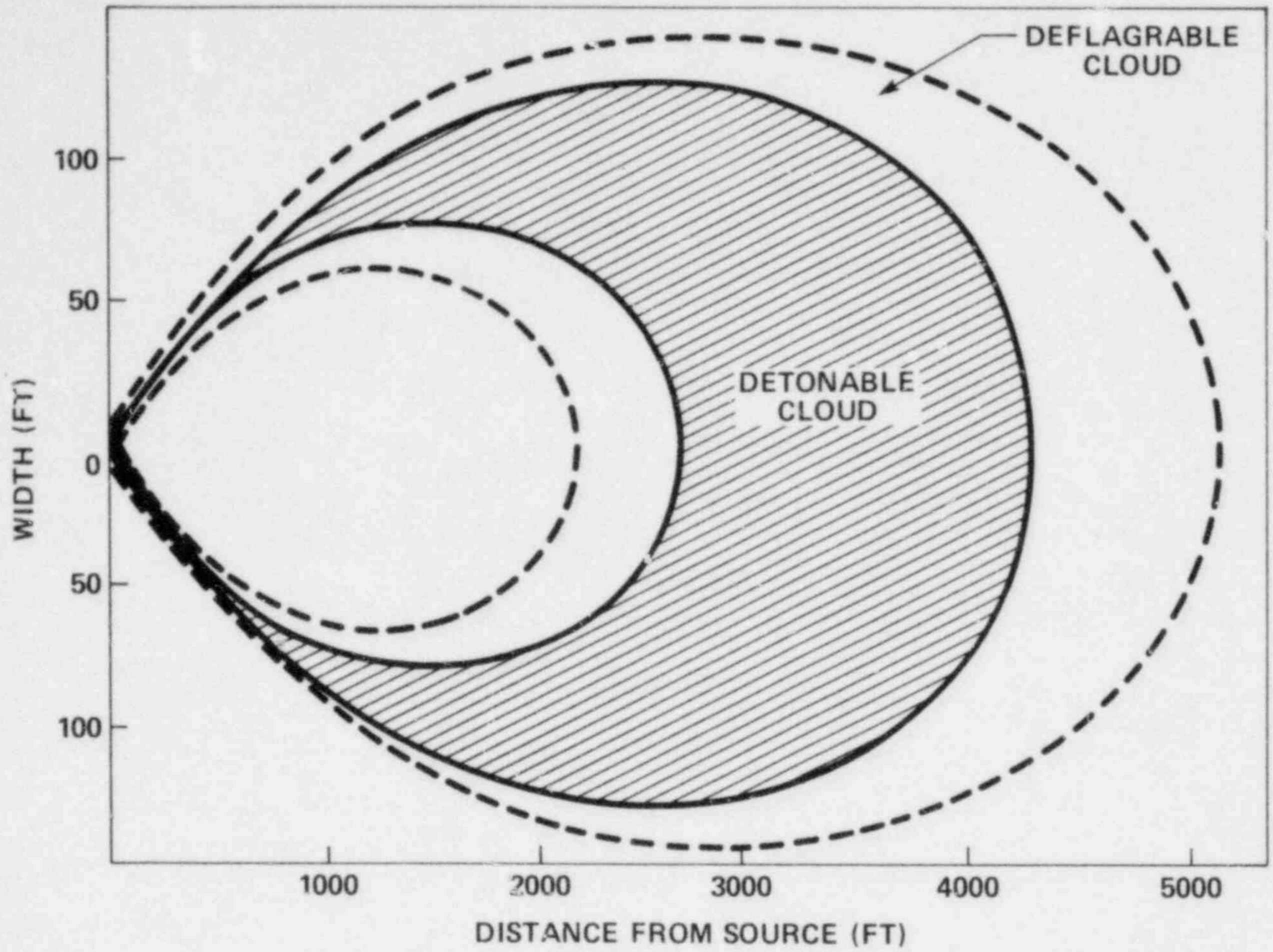
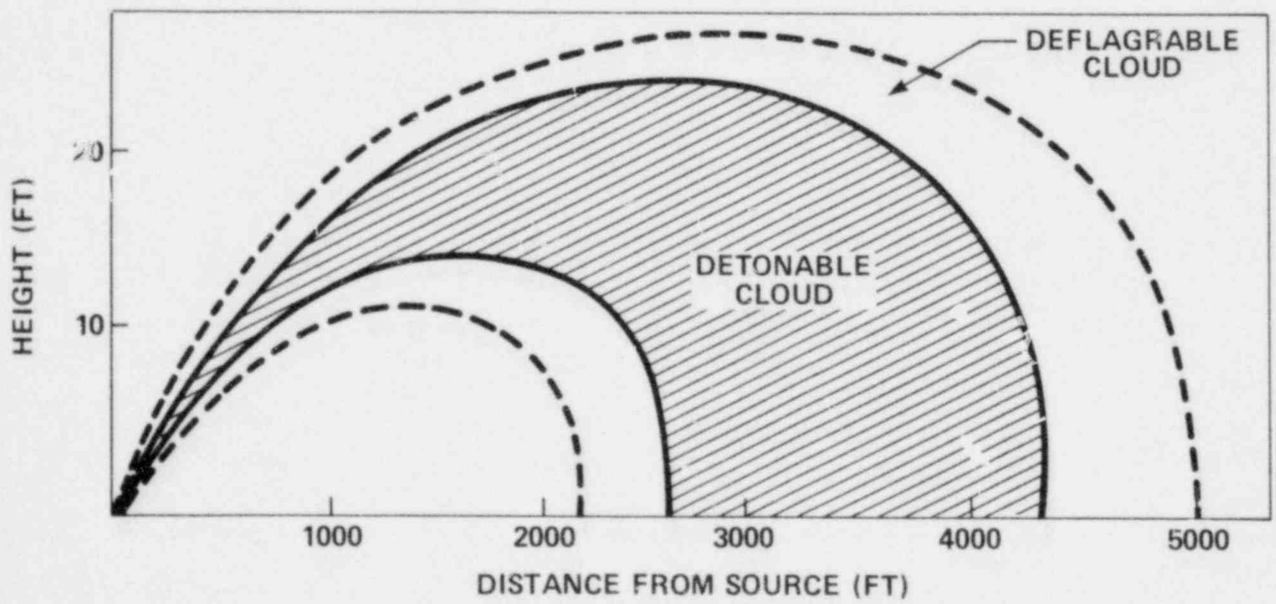
Unit 1

FLOW RATE ISSUEING FROM THE BREAK VS TIME
FIGURE 1.2





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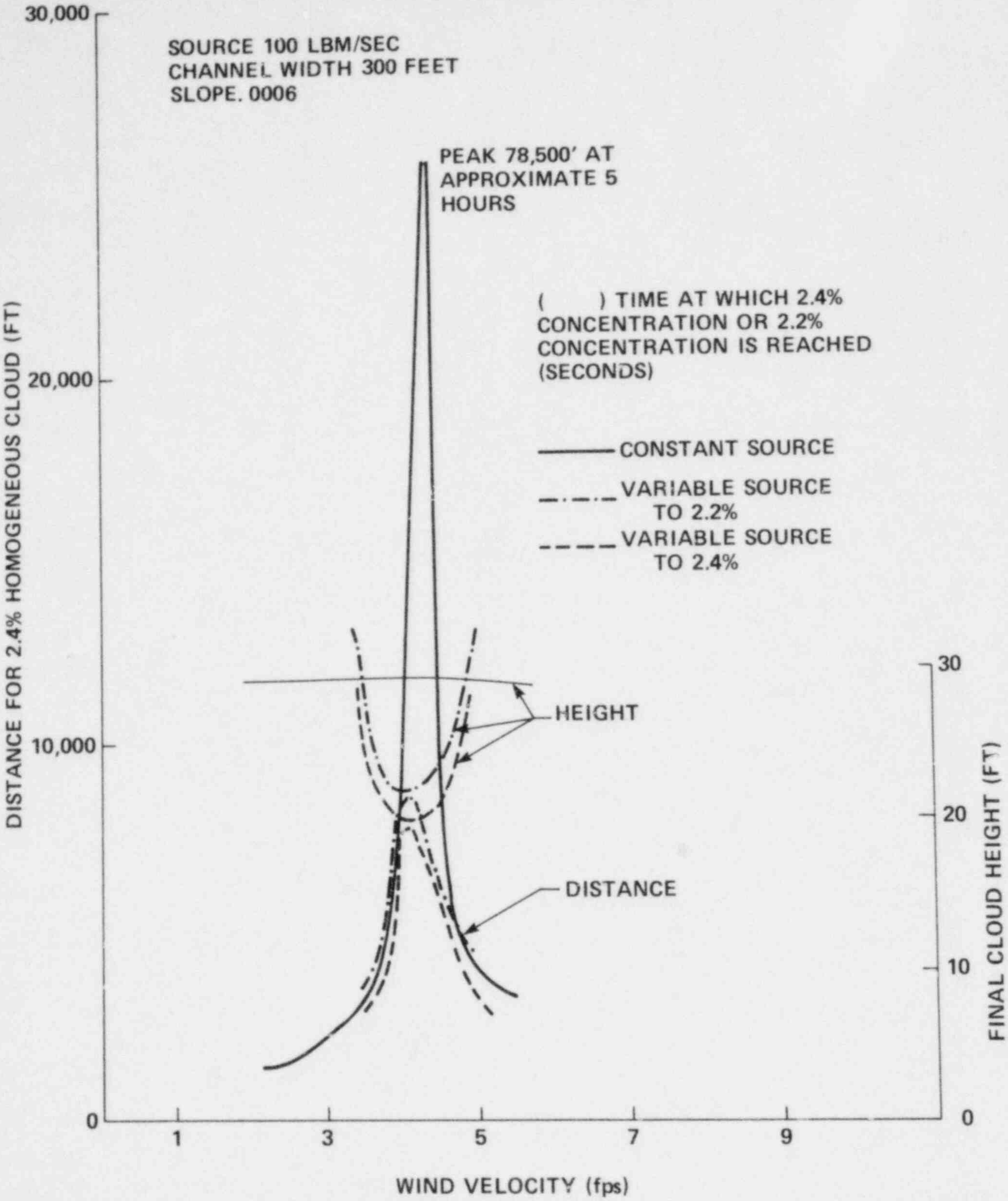
DETONABLE & DEFLAGRABLE CLOUD FORMATION FOR 6 INCH LPG LINE BREAK

FIGURE 1.3



1. CONSTANT 100 LBM/SEC SOURCE
2. INITIAL 100 LBM/SEC SOURCE DECAYING TO 30 LBM/SEC AT 995 SECONDS

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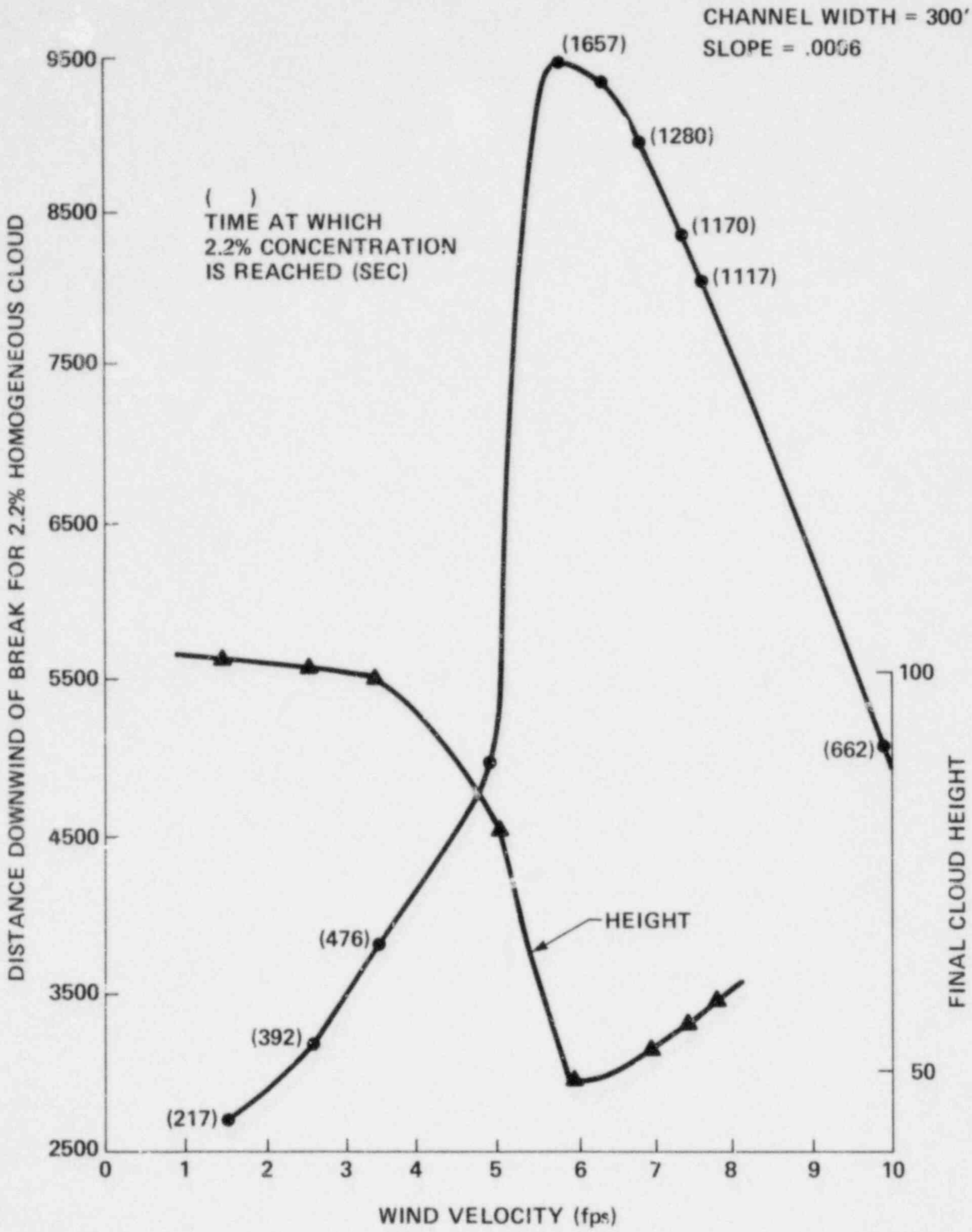
FARTHEST DOWN CHANNEL REACH & HGT. OF GRAVITY SLUMPING CLOUD AS A FUNCTION OF WIND VELOCITY

FIGURE 1.4



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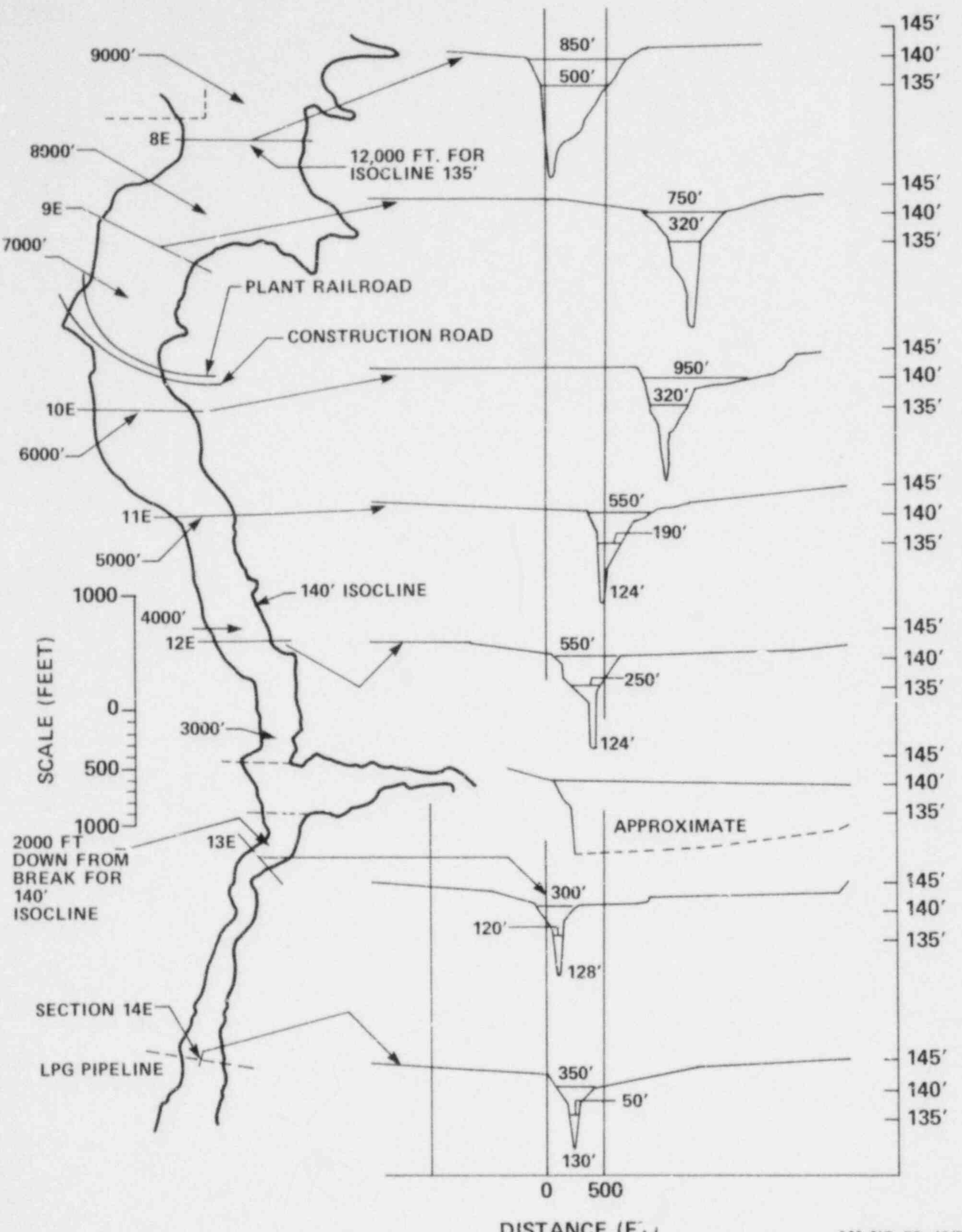
FARTHEST DOWN CHANNEL REACH AND HEIGHT OF GRAVITY SLUMPING CLOUD AS A FUNCTION OF WIND VELOCITY
FIGURE 1.5



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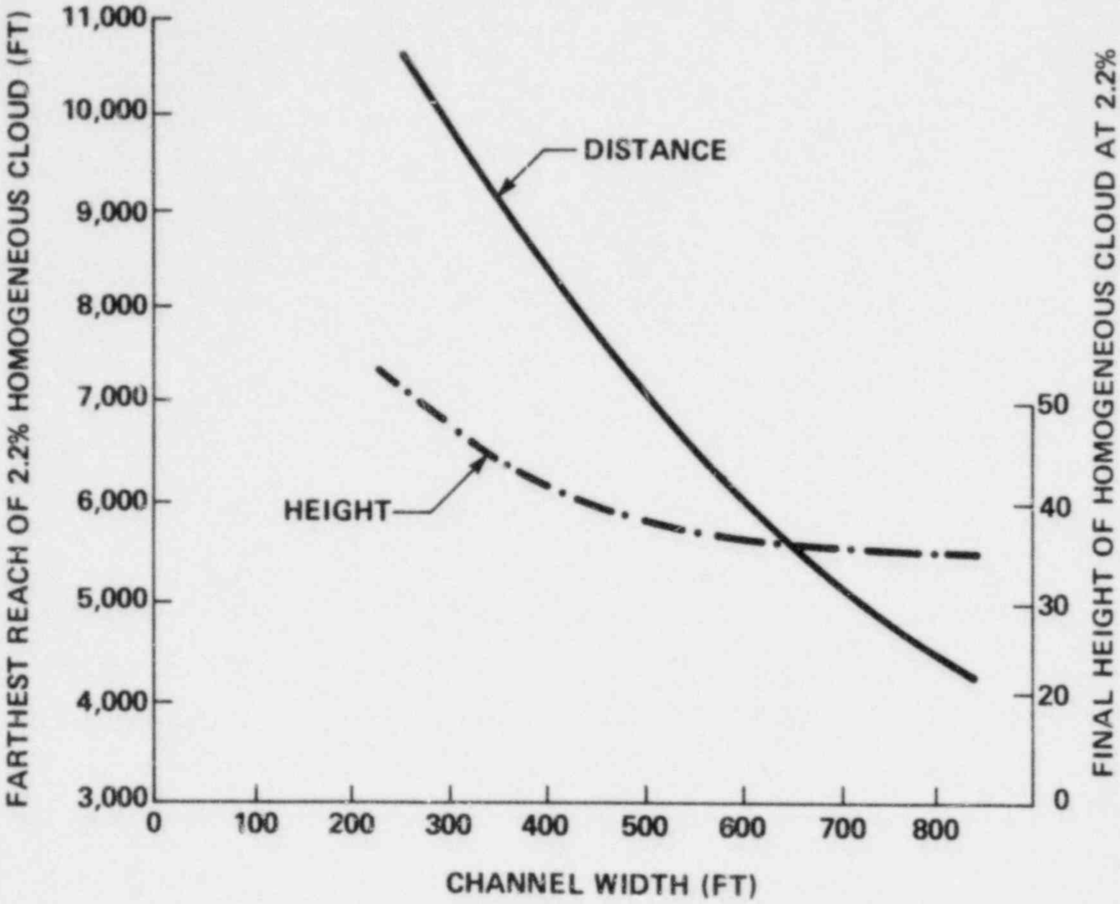
ALLENS CREEK CHANNEL
WITH A 140 FOOT
ISOCLINE

FIGURE 1.6



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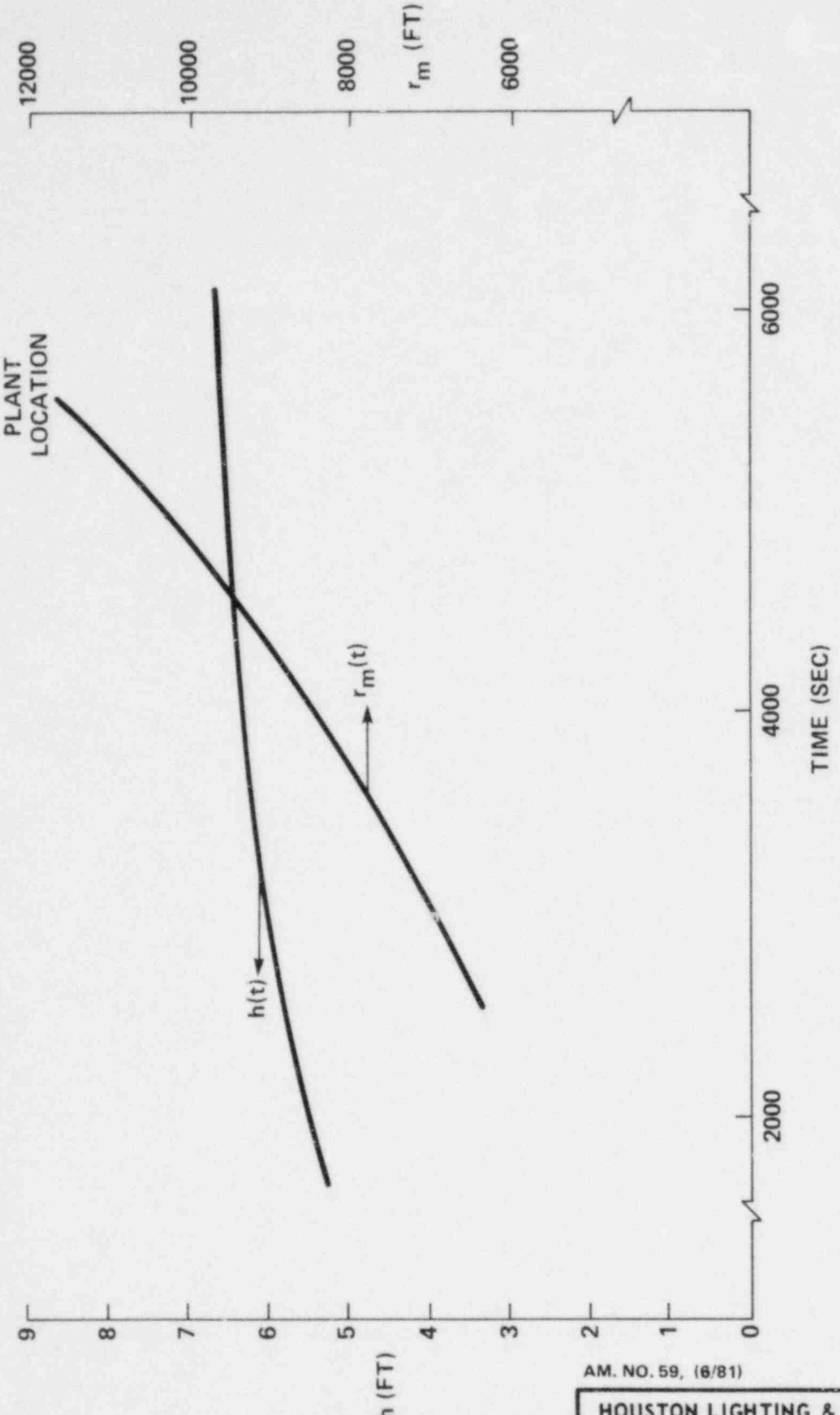
CLOUD DIMENSION VS CHANNEL WIDTH (WIND SPEED = 6 FPS) FIGURE 1.7



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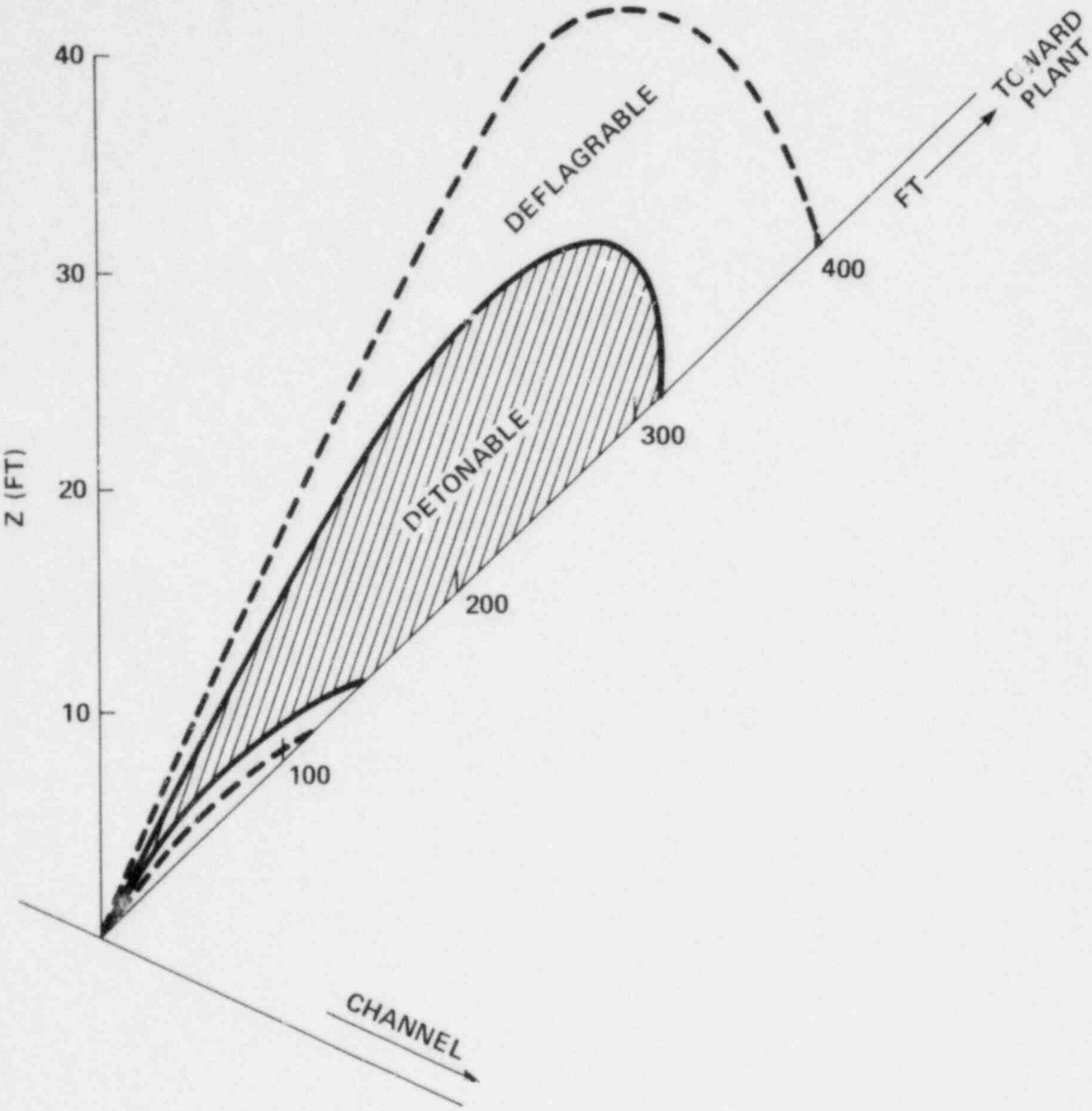
DEPTH & EXTENT OF PURE PROPANE CLOUD AS A FUNCTION OF TIME (SOURCE - 100 LB/SEC.)

FIGURE 1.8



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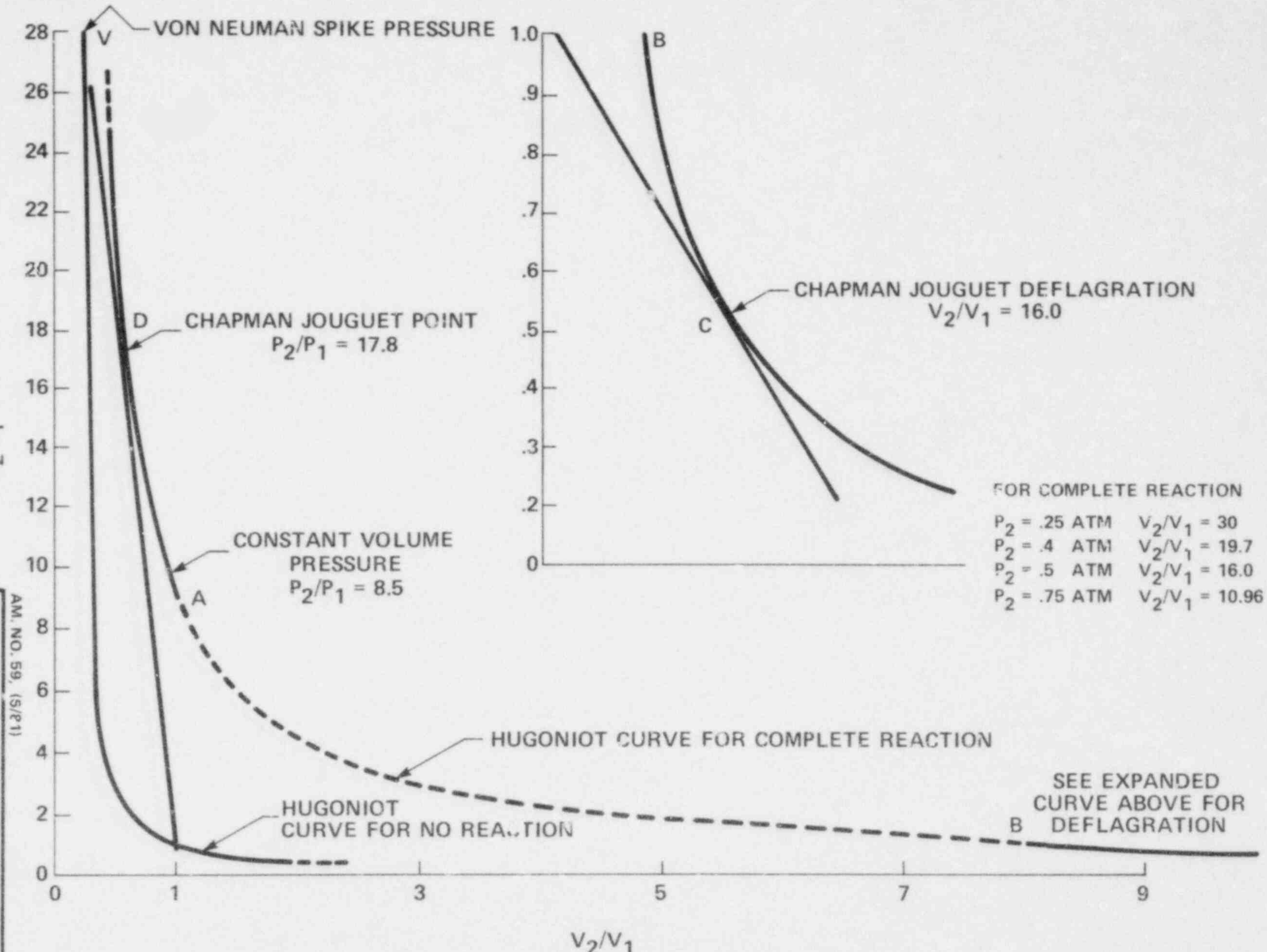
SECTION OF DETONABLE & DEFLAGRABLE CLOUD FROM INFINITELY LONG LINE SOURCE 0.179 FT^2/SEC STRENGTH
FIGURE 1.9



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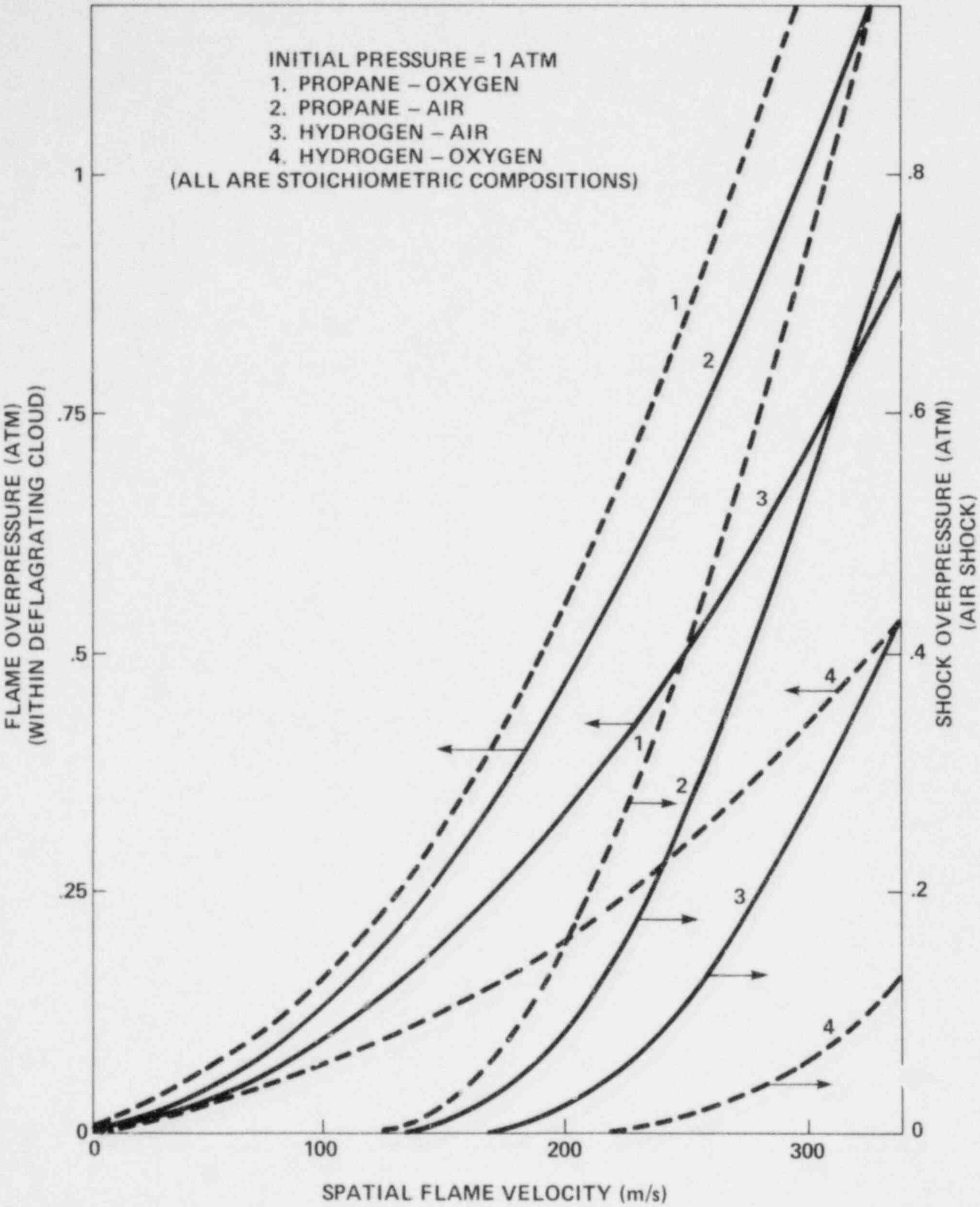
HUGONIOT CURVES

FIGURE 1.10



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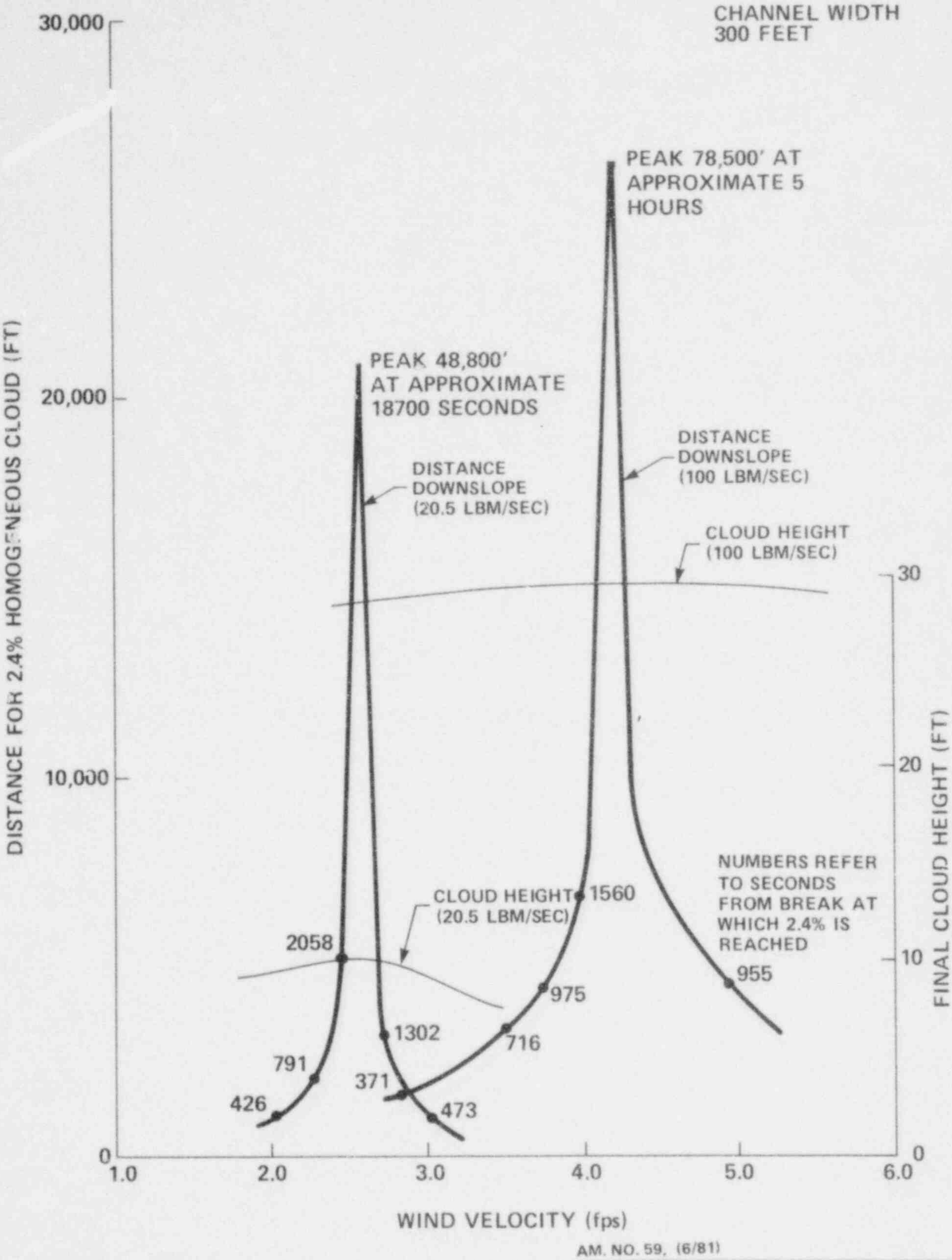
Unit 1

SPATIAL FLAME VELOCITY
VS. OVERPRESSURE

FIGURE 1.11



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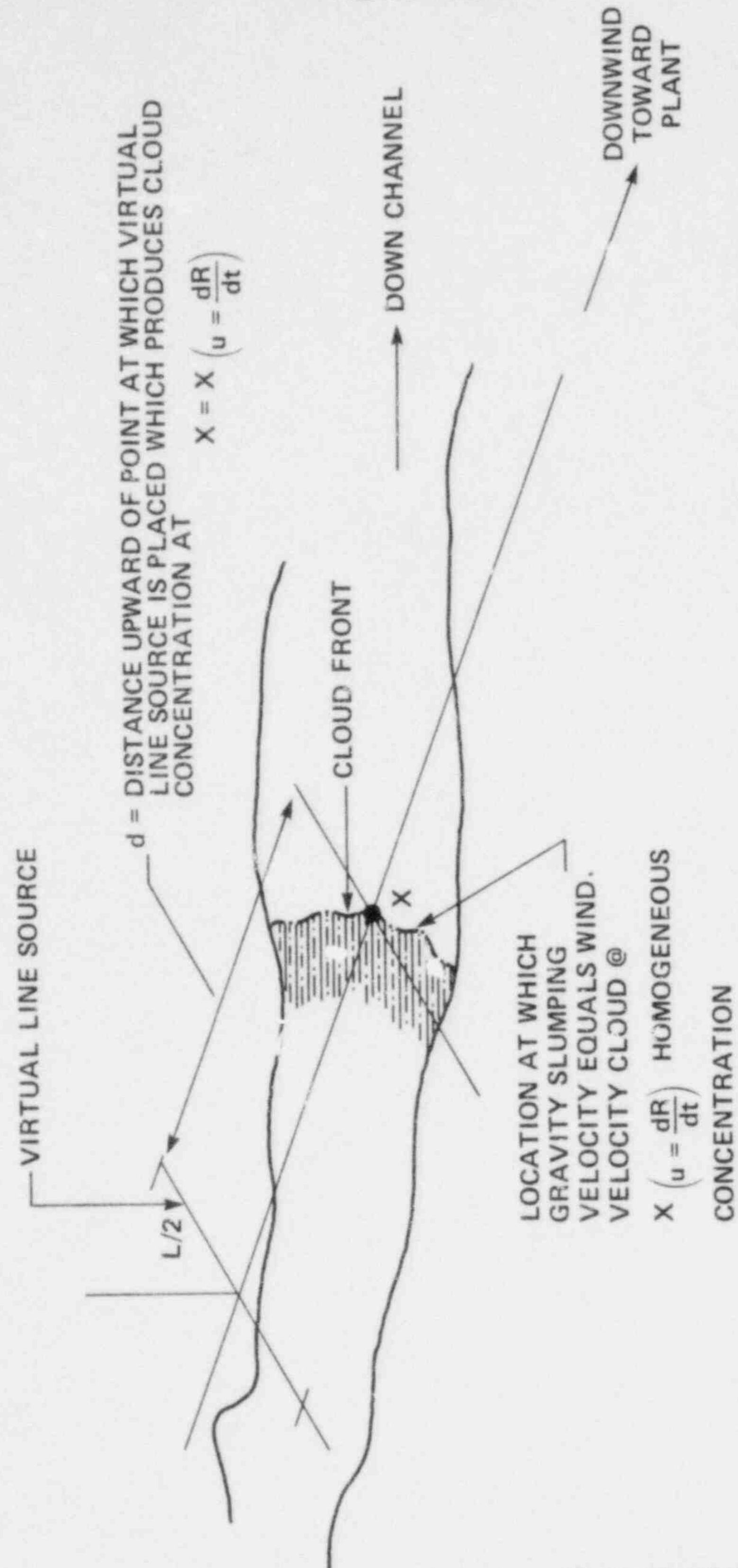
CLOUD FROM CONSTANT SOURCE FIGURE 1.12



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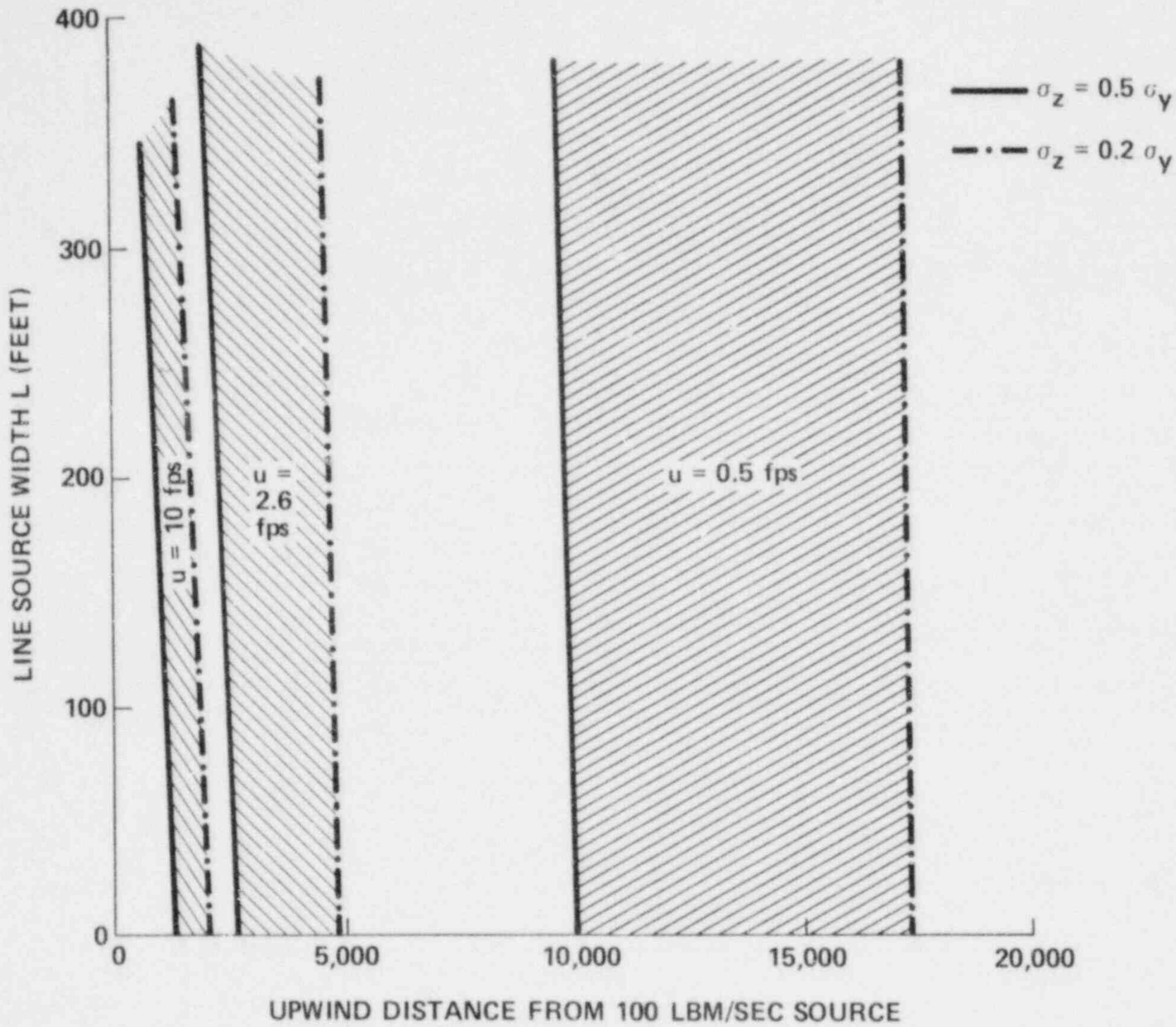
GRAVITY SLUMPING CLOUD MODEL
FIGURE 1.13



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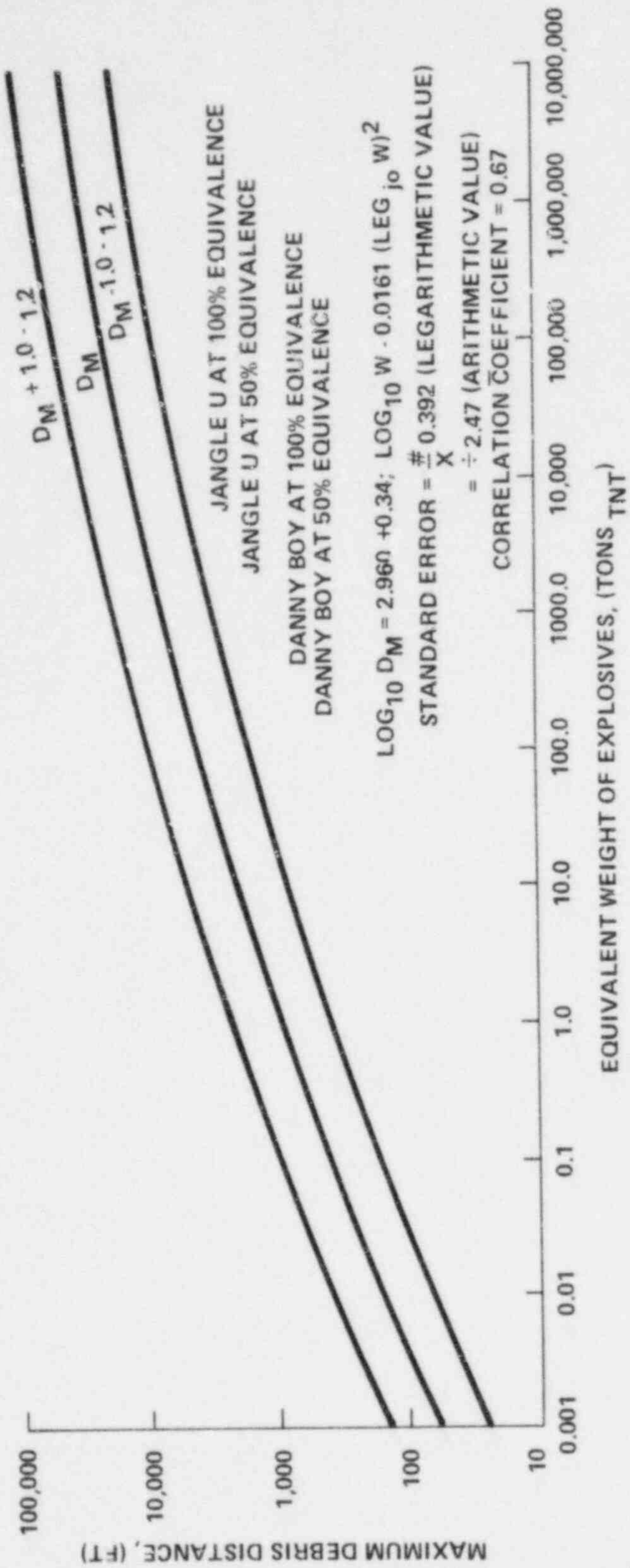
UPWIND DIST. OF 100 LB/SEC PROPANE SOURCE OF WIDTH L RESULTING IN A CNTRLINE. GRND. LVL. CONCEN. OF 2.4%

FIGURE 1.14



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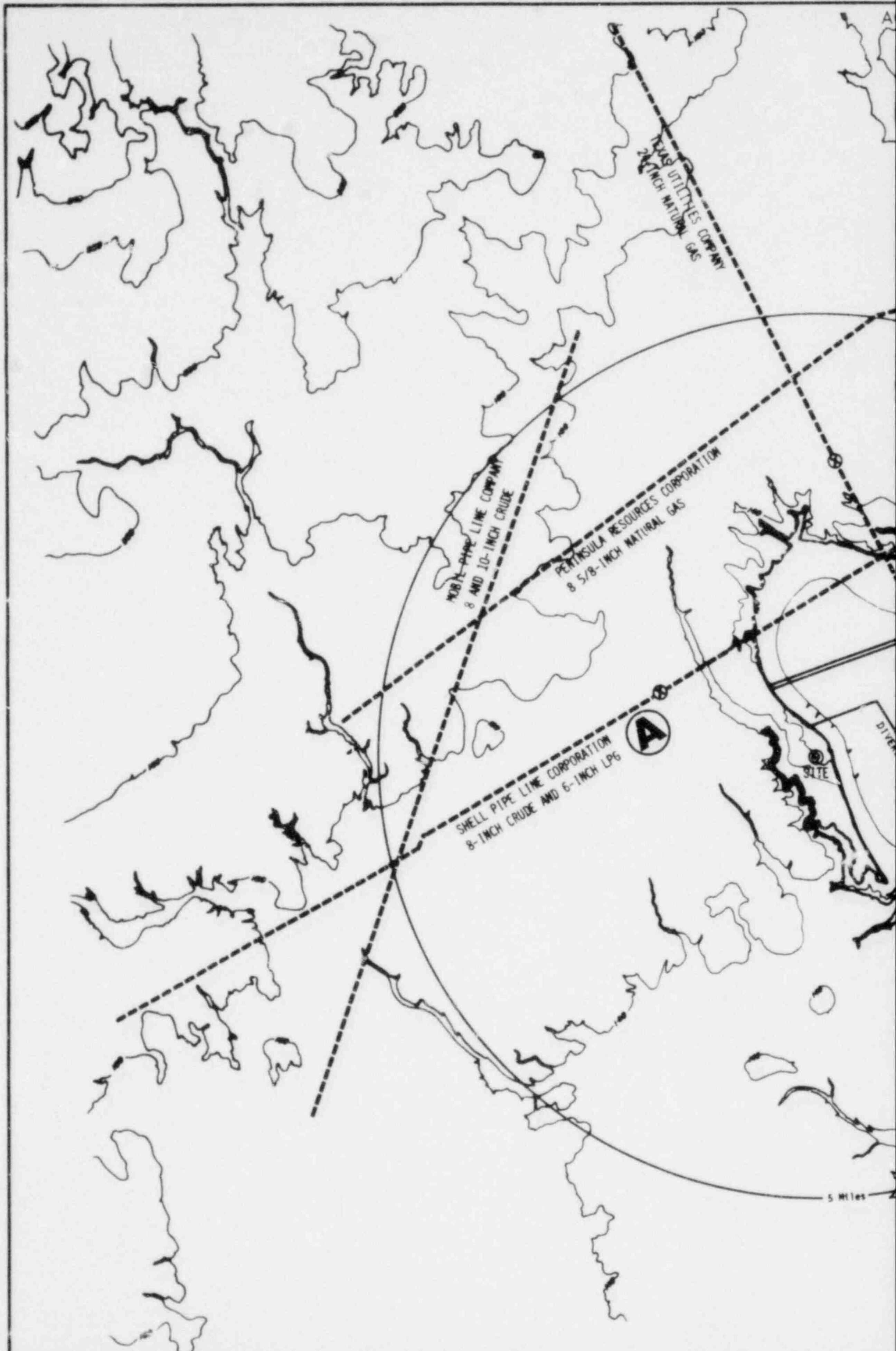
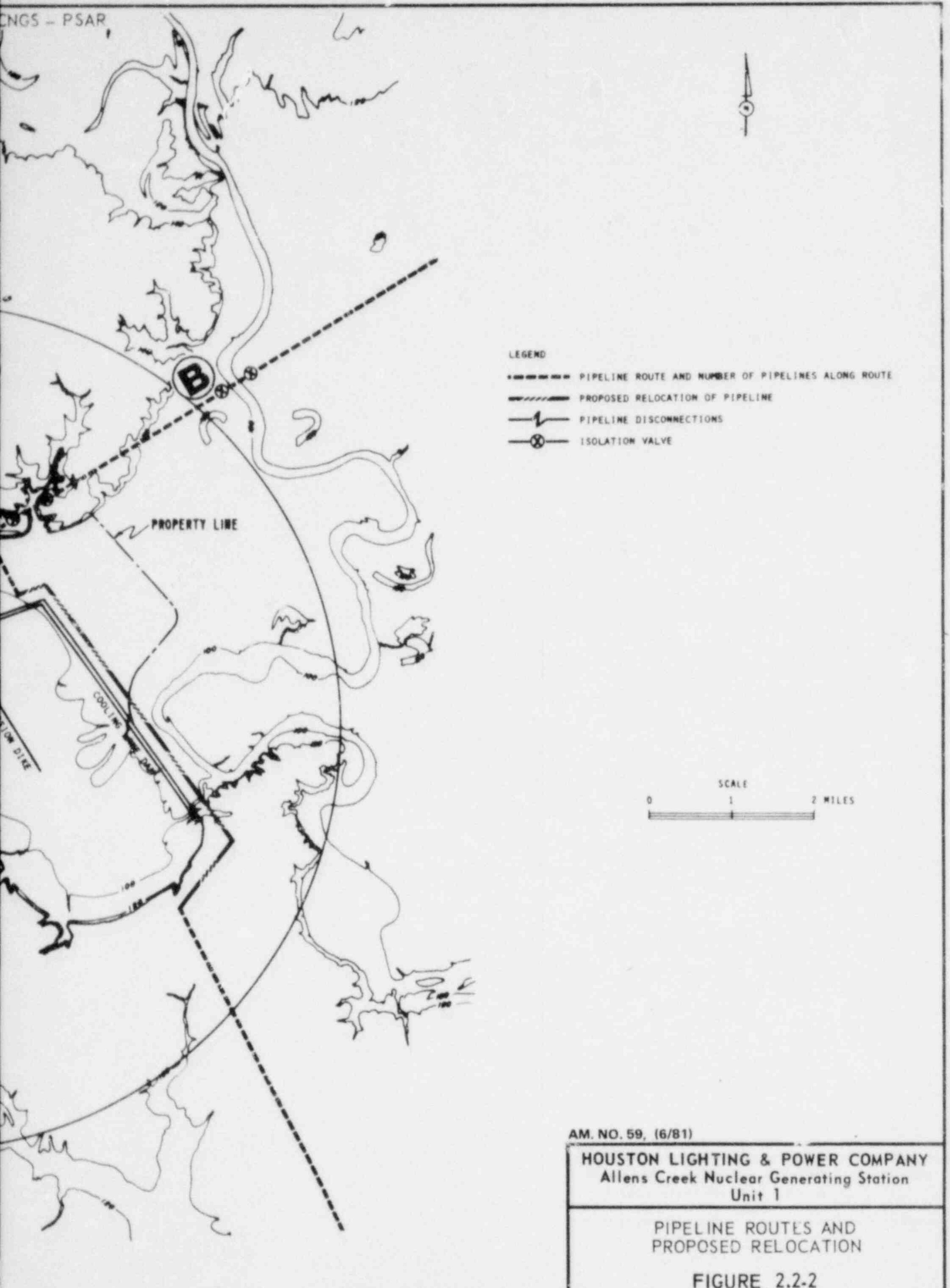
Unit 1

QUADRATIC REGRESSION LINE:
MAX. DEBRIS DISTANCE VS.
EQUIVALENT YIELD

FIGURE 1.15


TEXAS UTILITIES COMPANY
24-INCH NATURAL GAS
MOBIL PIPE LINE COMPANY
8 AND 10-INCH CRUDE
PENINSULA RESOURCES CORPORATION
8 5/8-INCH NATURAL GAS
SHELL PIPE LINE CORPORATION
8-INCH CRUDE AND 6-INCH LPG
A
SITE
5 Miles



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PIPELINE ROUTES AND
PROPOSED RELOCATION

FIGURE 2.2-2

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TABLE 3.2-1 (Cont'd)

	Principal Component[a]	Scope of Supply[b]	Safety Class[c]	AEC Quality Group[d]	Component Location[e]	Seismic Category[f]	ENVIRONMENTAL CAPABILITY: Extreme Wind[g]	Tornado/ Missile[h]	Flood Protection[i]	Quality Assurance Program[j]	Comments	
XXXVI	Control Room Air Conditioning System	P	3	NA	B	I	b	b	c	B		
XXXVII	ECCS Area Exhaust System											
	1. Filters	P	3	NA	A	I	b	b	c	B		35(C)
	2. Ductworks, valves	P	3	NA	A	I	b	b	c	B		35(C)
	3. Cables	P	3	NA	A	I	b	b	c	B		35(C)
	4. Exhaust Fans	P	3	NA	A	I	b	b	c	B		35(C)
	5. Lines penetrating ECCS Area envelope	P	3	NA	A	I	b	b	c	B		57
XXXVIII	Leak Detection System (Containment)											
	1. Temperature element	GE	2	NA	C	I	b	b	c	B	(z)	
	2. Differential temperature switch	GE	2	NA	C	I	b	b	c	B	(z)	
	3. Differential flow indicator	GE	2	NA	C	I	b	b	c	B	(z)	
	4. Pressure switch	GE	2	NA	C	I	b	b	c	B	(z)	
	5. Differential pressure indicator switch	GE	2	NA	C	I	b	b	c	B	(z)	
	6. Differential flow summer	GE	2	NA	C	I	b	b	c	B	(z)	
	Other Leak Detection											59
	1. Control Building Basemat Sumps Level Indication	P	2	NA	B	I	b	b	c	B		59
	2. Reactor Auxiliary Building Low Purity Sumps Level Indication	P	2	NA	A	I	b	b	c	B		59
	3. DG Building Cubicle and Day Temp Sumps Level Indication	P	2	NA	S	I	b	b	c	B		59
XXXIX	Civil Structures											
	1. Reactor Building including:											
	Base slab	P	2	NA	M	I	b	b	a	B		
	Shield Building Cylindrical Wall	P	2	NA	O	I	a	a	b	B		22
	Dome	P	2	NA	O	I	a	a	b	B		22
	Steel Containment	P	2	NA	c	I	b	b	c	B		

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it would be necessary to penetrate the operating floor at an impact angle of five to fourteen degrees from horizontal. The operating floor is composed of reinforced concrete three feet thick. (See Figure 1.2-22). At these angles of impact, the missile would not penetrate the operating floor. 35(U)

To impact the walls of the Radwaste Building adjacent to the Turbine Building, it would be necessary for the missile to penetrate the reinforced concrete turbine pedestal and several reinforced concrete internal walls at the mezzanine level. Therefore, the missile would not have sufficient energy to reach these walls. 35(U)

35(U)

3.5.2.2.4 Summary and Conclusion 59

The high trajectory turbine missiles are characterized by their nearly vertical trajectories. The total damage probability of a high trajectory turbine missile striking the safety related structures is less than 10^{-7} per unit year as listed in Table 3.5-4. In addition, the vulnerable safety related equipment area which is exposed to the potential turbine missile is redundant and physically well separated. Consequently the risk from high trajectory turbine missiles is insignificant. 58

The ACNGS turbine generator has been arranged in a peninsula orientation. With the exception of the Radwaste Building, this configuration excludes all major systems important to safety from the low trajectory turbine missile strike zones. The Radwaste Building does not contain any essential systems required for safe shutdown and is located below the turbine operating deck level. The location of the Radwaste System components relative to the turbine is such that they are adequately protected by the presence of the reinforced concrete pedestal, internal building walls and the turbine operating deck floor. Thus, the plant configuration complies with the guidelines of Regulatory Guide 1.115, "Protection Against Low Trajectory Turbine Missiles."

In addition to the above, due to the redundancy and testing features of the turbine overspeed protection, quality control manufacturing processes, materials, and inspection program, the hypothetical turbine missiles are considered very remote. Consequently the risk of potential turbine missile damage to safety related plant structures, systems, and components for the facility is acceptably low.

3.5.2.3 Site Related Missiles

3.5.2.3.1 Aircraft Missiles

No commercial airport facilities are located in the vicinity of the site. The nearest major commercial facilities are located at Hobby and Houston Intercontinental, both of which are located approximately 50 miles from the site. The Eagle Lake facility is located approximately eleven miles west of the site. No military facilities exist within 20 miles of the site. Consequently neither aircraft nor military projectiles are considered credible missiles for the site. Section 2.2.1.6 described airport locations 35(U)
in the site region.

3.5.2.3.2 Transportation Accidents

Section 2.2.3.4 demonstrates that the consequences of missiles arising from 35(U)
explosion of hazardous material being transported along Highway 36 and Gulf Colorado and Santa Fe line of the Santa Fe Railway is not significant.

3.5.2.3.3 Industrial Accidents

Pipelines in the vicinity of the plant include a 24 inch natural gas pipeline of the Texas Utilities Company and 8 and 6 inch oil pipelines of the 35(U)
Shell Pipeline Corporation. The 24 inch pipeline will be rerouted on the opposite side of the cooling pond from the plant, approaching to within approximately 2 miles of the plant. Section 2.2.3 demonstrates that the consequences of missiles arising from these sources is not significant.

No other major industrial facilities exist within 5 miles of the site.

3.5.2.4 Tornado Missiles

Structures or components whose failure could prevent safe shutdown of the reactor, or result in significant uncontrolled release of radioactivity from the unit, shall be protected from such failure due to design tornado wind loading or missiles by the following methods:

a) Structure or component is designed to withstand tornado loading (see Section 3.3.2) and/or tornado missile.

b) Component is housed within a structure which is designed to withstand the tornado and/or missile loading.

P_{mw} = A 45 psig internal static pressure to envelope the combined pressure induced by an accident that releases hydrogen generated from 100% active fuel clad metal-water reaction and the pressure from post-accident inerting assuming carbon dioxide. See Section 6.2.1.3.4 for the derivation of the accident pressure. 57 59

P_{in} = Pressure produced by an inadvertent actuation of the post-accident inerting system causing full inerting (with carbon dioxide). This internal static pressure is 25 psig, as derived in Section 6.2.1.3.5.

P_{st} = Containment Vessel structural acceptance test 57

(pressure = 110 percent of P_{in} as required by NUREG 0718 Item II.B.8.4.

b) Temperatures

T_a = Design (accident) temperature inside Containment. When coincident with P_a this temperature is 185°F (Table 6.2-1A). It is adjusted accordingly when negative (accident) pressure occurs. For T_a under IBA and SBA, see Sections 6.2.1.3.1.3 and 6.2.1.3.1.4 respectively. 35(D) 54

T_o = Operating Temperature (the range is 60 to 80 F inside Containment and 51 to 95 F in the annulus). During SRV blowdown the increased temperature in the suppression pool is included in T_o. During construction T_o is specified as the construction temperature.

T_{mw} = Temperature inside the containment associated with the pressure P_{mw}. When coincident with P_{mw} this temperature is 195°F, as described in Section 6.2.1.3.4.

T_{in} = Temperature inside the containment associated with the inadvertent actuation of the post-accident inerting system causing full inerting (with carbon dioxide). This temperature is 95°F, as described in Section 6.2.1.3.5.

T_{st} = Ambient temperature in the containment during the Containment Vessel structural acceptance test. This ranges from 30°F to 96°F. 57

c) Dead Loads

D = Dead loads; they shall include the following:

1) Weight of vessel shell, penetrations, hatches and locks.

2) The dead weight of the polar crane and its runway.

3) Weight of platforms, walkways, equipment, piping, ventilation duct, cable and trays, conduit, etc. These loads are generated either by direct attachment to the vessel, or through supporting structures. 35(C)

3.8.2.3.2 Load Combinations

3.8.2.3.2.1 Containment Vessel Shell 9

The design of the Containment will include consideration of the load combinations listed below. Stress limits for these loading conditions are discussed in Section 3.8.2.6. 54

a) Construction and Test Conditions

1) $D + L^{(1)} + T^{(1)} + W^{(1)}$

2) $D + L^{(2)} + T_o^{(2)} + P_t^{(3)} + F_n$ — Vessel Test 35(C)

3) $D + L^{(2)(4)} + T_o^{(2)} + P_o^{(2)} + F_n$ — Crane Test

4) $D + P_{st} + T_{st}$ — Vessel Test (PAIS actuation) 59

b) Normal Operating or Shutdown Conditions 35(D)

5) $D + L + T_o + P_o + R_o + F_n$ 57

6) $D + L + T_o^{(5)} + P_o + P_{bd} + R_o + F_n$ — SRV blowdown 35 (D) 57

7) $D + P_{in} + T_{in}$ — PAIS Inadvertent Actuation 57

c) Severe Environment Loads

8) $D + L + T_o^{(5)} + P_o + P_{bd} + R_o + F_n + F_{eqo}$ — SRV blowdown 57

9) $D + L^{(6)} + T_o^{(6)} + P_o^{(6)} + R_o^{(6)} + F_n + F_{eqo}$ — Refueling 9 Q1.33 57

d) Extreme Environmental Loads

10) $D + L + T_o^{(5)} + P_o + P_{bd} + R_o + F_n + F_{eqs}$ — SRV blowdown 35 (D) 44 57

e) Abnormal - Severe Environmental Loads

11) $D + L + T_a^{(7)} + P_a^{(7)} + P_{bd}^{(8)} + P_{ps}, P_{sc}$ or $P_{ch}^{(9)(13)} + F_n + F_{eqo} + R_a$ — Pool swell Steam Condensation or chugging 57 59

12) $D + L + T_a^{(10)} + P_a^{(10)} + R_a + F_n^{(10)} + F_{eqo}$ — Long Term LOCA 54 57

13) $D + L + T_a^{(11)(14)} + P_a^{(14)} + P_{bd}^{(12)} + P_{sc}$ or $P_{ch}^{(9)(13)} + R_a + F_n + F_{eqo}$ — Intermediate break, ADS 57

14) $D + L + T_a^{(15)} + P_a^{(15)} + P_{bd}^{(8)} + P_{ch}^{(9)} + R_a + F_n^{(10)} + F_{eqo}$ — Small break 57

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15) $D + L + T_a + P_e + R_a + F_n + F_{eqo}$	Negative Pressure	57
16) $D + L + T_o + P_o + R_o + F_n + Y_j + F_{eqo}$	Accident at penetration sleeve	57
17) $D + L + T_o^{(6)} + P_o^{(6)} + R_o^{(6)} + F_{pa} + F_{eqo}$	Post-accident flooding	57

f) Abnormal - Extreme Environmental Loads 54

18) $D + L + T_a^{(7)} + P_a^{(7)} + P_{pd}^{(8)} + P_{ps}$, P_{sc} or $P_{ch}^{(9)(13)} + R_a + F_n + F_{eqs}$	Pool swell steam condensation or chugging	57
19) $D + L + T_a^{(10)} + P_a^{(10)} + R_a + F_n^{(10)} + F_{eqs}$	Long term LOCA	57
20) $D + L + T_a^{(11)(14)} + P_a^{(14)} + P_{bd}^{(12)} + P_{sc}$ or $P_{ch}^{(9)(13)} + R_a + F_n + F_{eqs}$	Intermediate break, ADS	57 59
21) $D + L + T_a^{(15)} + P_a^{(15)} + P_{bd}^{(8)} + P_{ch}^{(9)} + R_a + F_n^{(10)} \times F_{eqs}$	Small break	59
22) $D + L + T_a + P_e + R_a + F_n + F_{eqs}$	Negative pressure	57
23) $D + L + T_o + P_o + R_o + F_n + Y_j + F_{eqs}$	Accident at penetration sleeve	57
24) $D + P_{mw} + T_{mw}$	Degraded core	57

values of R_a (thermal load) on the penetrations. The design accident loads
discussed above, result from a postulated pipe break inside the drywell. They 35(C)
would not occur simultaneously with the loads "Y_j" which are for a pipe break-
ing at a penetration.

The intent of load combinations (e) 16 and (f) 23 is to cover the design of 57
local areas around penetrations for any pipe break postulated to occur at a
penetration. In such a case, design accident loads (pressure, temperature,
and fluid) would not be acting simultaneously on the overall vessel. The
loads "Y" which would be acting at the penetration already include the
thermal effects on the penetration of the postulated pipe rupture (see
definition item (g) in Section 3.8.2.3.1). 35(C)

Local areas will be designed by investigating the applicable loads combined as in the above listed loading cases. Local areas to be investigated include penetration nozzles and the surrounding shell, anchorage details, crane runway and floor framing brackets, and the dome knuckle. Investigations of these are discussed further in Section 3.8.2.4.

3.8.2.3.2.2 Bottom Liner 59

The containment bottom liner plate will be designed in accordance with the
applicable rules listed in the ACI-ASME (ACI-359), Division 2 Code issued
in January 1975. It will also be designed in accordance with selected sec- 54
tions of ASME Code, Section III, Division 1, Subsection NE applicable to
strength, buckling and low cycle fatigue for cases where SRV negative
pressure occurs. The load combinations shown in Table 3.8-3 and the
additional combinations shown in Section 3.8.5.3.2(a) related to NUREG 0718 59
Item II.B.8.4 are applicable to the liner plate design except that load
factors for all load cases shall be taken to equal to 1.0. 54

3.8.2.4 Design and Analysis Proccedures

The design and analysis of the Containment will be the responsibility of the selected containment vendor. The scope of the vendor's responsibility includes the design and analysis of the vessel shell and bottom liner, the vessel anchorage, the crane runway, dome platforms, intermediate floor support seats, personnel locks, equipment hatch, and penetration nozzles. The penetration internals discussed in Section 3.8.2.1.2 are not included.

The vessel vendor will be required to report fully on the actual completed design and analysis, and a summary of this will be available for the FSAR.

Containment Vessel design and analysis procedures will vary somewhat
according to the selected vendor. However, the following discussion rep-
resents, in general a typical example of the approaches utilized, and, in
several areas, it represents specific requirements. 1
Q3.20

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The Standby Gas Treatment System is represented by input tables describing fan size, system resistance, and pressure dependent flow characteristics.

The analyses were performed for the steam line break described earlier in Section 6.2.1.3.1 with the same minimum Engineered Safety Features as described there. In addition, only one of the two 100 percent SGTS subsystems is assumed to be operating.

Table 6.2-5 lists the important assumptions used in the Shield Building annulus transient analyses.

A heat transfer coefficient between the containment atmosphere and the containment steel is taken to be equal to 250 BTU/hr.-ft^2-^{o}F. This number represents a typical condensation heat transfer coefficient on the containment steel. This heat transfer coefficient is higher than a convection heat transfer coefficient and will result in a faster heat transfer rate to the annulus. This is the reason that a conservative condensing heat transfer coefficient is used instead of a more realistic convection heat transfer coefficient.

Inleakage to the annulus is based on a design basis leak rate of 45 cfm at
-2 inches wg which will be verified by test. During phase 3 operation the
SGTS exhaust rate is equal or less than 2000 cfm at all times. This is the 17
value used throughout the phase 3 period for LOCA dose calculations given
in Chapter 15. Q2-58

Figure 6.2-22 shows the annulus pressure versus time for the above described conditions. Figure 6.2-23 is a linear time scale replot of the annulus pressure transient given in Figure 6.2-22. Figure 6.2-24 shows the corresponding annulus temperature versus time. All of the above curves are based on an assumption of zero percent initial humidity in the annulus, as this assumption results in the highest pressure and temperature transients in the annulus. These plots show that the temperature in the annulus will reach 130 F after 2 hours. The annulus reaches a peak of -1/4 in. H_2O at 50 minutes and would remain negative during the transient. The results show that the design criteria set out for the SGTS in Section 6.2.3 are met.

6.2.1.3.4 Post-Accident Containment Pressure Calculation

Based on a transient event which results in 100 percent metal water reac- 57
tion of the active fuel clad, the following conditions are calculated for
this event: containment pressure = 42.5 psig, containment temperature
= 133^{o}F and suppression pool temperature = 183^{o}F. See the response to 59
NUREG 0718 Item II B.8.4(a) in Appendix O for details of the analyses.

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6.2.1.3.5 Containment Pressure Calculation Inadvertent Actuation

For the purposes of containment structural evaluation, the following conditions are assumed to result from inadvertent actuation of the Post-Accident Inerting System during normal operation:

Containment pressure = 26.5 psig 59

Containment/Suppression pool temperature = 95^{o}F 57

Penetration Number	Penetration Type	Applicable GDC	System Service	(E)/(I)/(N)	Justi (F
I-Later	XI	56	RWCU - Sample Line	I	Not req and imm lowing but use acciden
M-137A & B M-141	IVd VI	56	Fire Protection	N	Availat essenti is not helpful followi
M-138A & B M-142	IVd VI	56	Fire Protection	N	Availat essenti is not helpful followi
M-139A & B	IVd	56	Seismic Fire Protection	N	Availat essenti is not helpful followi
M-140A & B	IVd	56	Seismic Fire Protection	N	Availat essenta is not helpful followi
M-143 & M-149	IV & VII	56	Post Accident Inerting	I	Not re diate use; m tiated

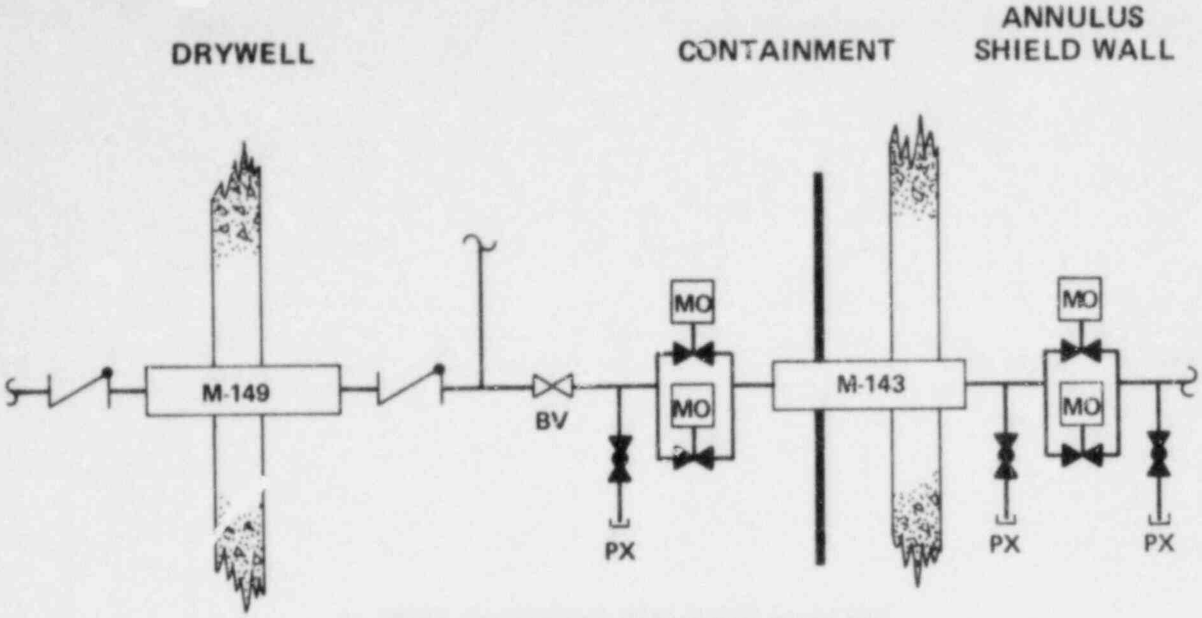
ification for)/(I)/(N)	Isolation Valve Tag Number	Valve Type	Line Size (In.)	Valve Operator	Maxim Closu Openi Time (
uired during	EV-71201	Gate	1/2	Solenoid	
mediately fol-	EV-71203	Gate	1/2	Solenoid	
an accident	EV-71205	Gate	1/2	Solenoid	
eful for post-	Later	Gate	1/2	Solenoid	
t sampling	EV-71200	Gate	1/2	Solenoid	
	EV-71202	Gate	1/2	Solenoid	
	EV-71204	Gate	1/2	Solenoid	
	Later	Gate	1/2	Solenoid	
bility is non-	2FP-V188-S1	Gate	4	Motor	Std (N
al as system	2FP-V190-S2	Check	4	-	-
required,	2FP-V546-S2	Gate	4	Motor	Std (N
or desirable	2FP-V548-S1	Check	4	-	-
ng accident.					
bility is non-	2FP-V229-S1	Gate	4	Motor	Std (N
al as system	2FP-V231-S2	Check	4	-	-
required,	2FP-V545-S2	Gate	4	Motor	Std (N
or desirable	2FP-V547-S1	Check	4	-	-
ng accident.					
bility is non-	2SF-V107-S1	Gate	6	Motor	Std (N
al as system	2SF-V110-S2	Check	6	-	-
required,					
or desirable					
ng accident.					
bility is non-	2SF-V108-S1	Gate	6	Motor	Std (N
il as system	2SF-V111-S2	Check	6	-	-
required,					
or desirable					
ng accident.					
quired for imme-	(Later)	Gate	8	Motor	Std (N
post-accident	(Later)	Gate	8	Motor	Std (N
anually ini-	(Later)	Gate	8	Motor	Std (N
system	(Later)	Gate	8	Motor	Std (N
	(Later)	Check	4	-	-
	(Later)	Check	4	-	-

TABLE 6.2-12 (Cont'd)

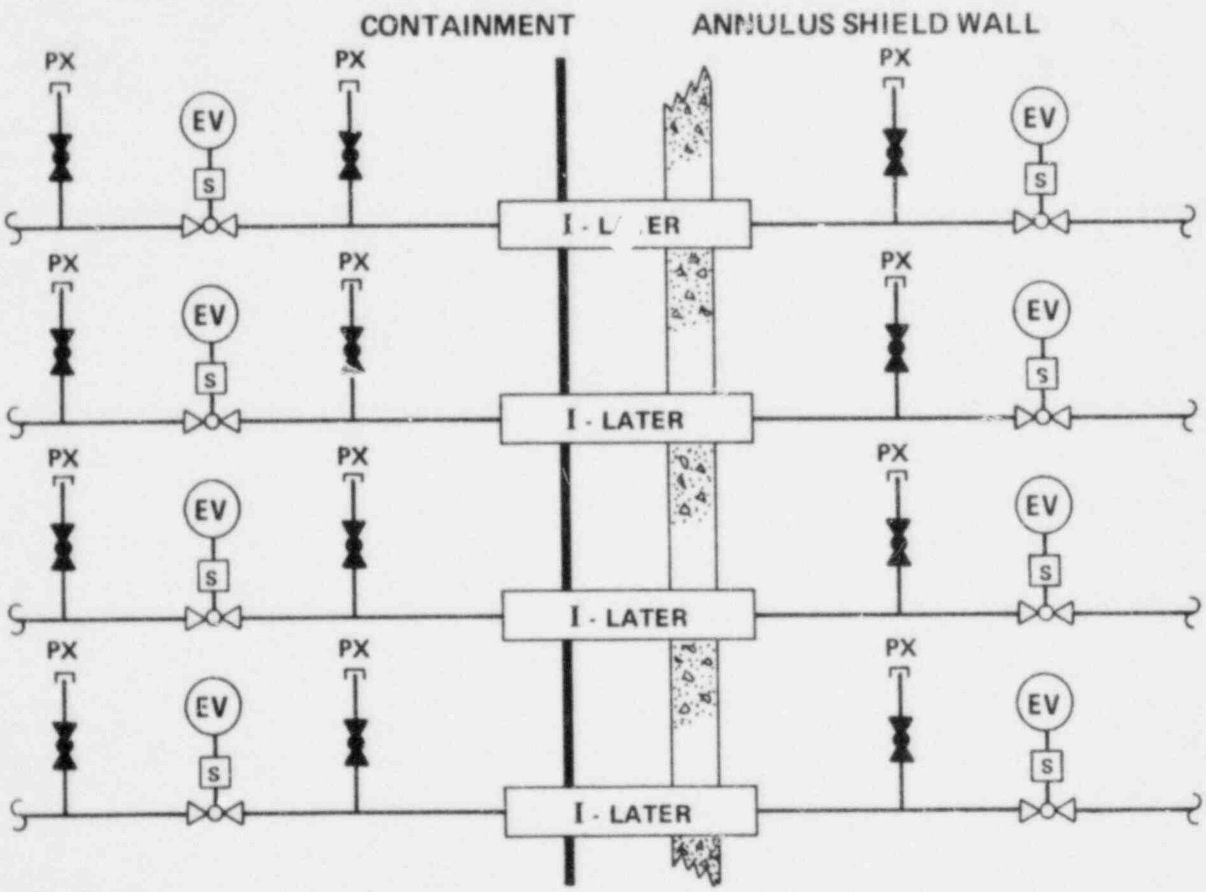
um re/ ng (Sec)	Valve Location	Flow Direction	Valve Position Normal	Accident	Power Failure	Isol Sig (Div Param (See Not
	Inside	Out	Open	Open or Closed	Closed	B,C,RM
	Inside		Open	"	"	"
	Inside		Open	"	"	"
	Inside		Open	"	"	"
	Outside		Open	"	"	"
	Outside		Open	"	"	"
	Outside		Open	"	"	"
	Outside		Open	"	"	"
te 2)	Outside Shield Bldg	In	Closed	Closed	As is	B,C,RM
	Inside Containment		-	-	-	Reverse
te 2)	Inside Containment		Closed	Closed	As is	B,C,RM
	Inside Drywell		-	-	-	Reverse
te 2)	Outside Shield Bldg	In	Closed	Closed	As is	B,C,RM
	Inside Containment		-	-	-	Reverse
te 2)	Inside Containment		Closed	Closed	As is	B,C,RM
	Inside Drywell		-	-	-	Reverse
te 2)	Outside Shield Bldg	In	Closed	Closed	As is	B,C,RM
	Inside Containment		-	-	-	Reverse
te 2)	Outside Shield Bldg	In	Closed	Closed	As is	B,C,RM
	Inside Containment		-	-	-	Reverse
te 2)	Outside Shield Bldg	In	Closed	Open or Closed	As is	B,C,RM
te 2)	Outside Shield Bldg		Closed	Open or Closed	As is	B,C,RM
te 2)	Inside Containment		Closed	Open or Closed	As is	B,C,RM
te 2)	Inside Containment		Closed	Open or Closed	As is	B,C,RM
	Inside Containment		-	-	-	Reverse
	Inside Drywell		-	-	-	Reverse

ation nal erse eter) (es 5,6,7)	Reopen By Manual Only	Appendix J Type C Test	Applicable Figure (See 6.2-26) (Sheet/Item)	Penetration Location	Bypass Leakage Barriers (See Note 11)
	Yes	Yes	27A / 31	Aux Bldg	A,B
	"	"			
	"	"			
	"	"			
	"	"			
	"	"			
	"	"			
	"	"			
	Yes	Yes	27 / 22a	Aux Bldg	A,B
flow	-	Yes			
	Yes	No			
flow	-	No			
	Yes	Yes	27 / 22a	Aux Bldg	A,B
flow	-	Yes			
	Yes	No			
flow	-	No			
	Yes	Yes	27 / 22b	Aux Bldg	A,B
flow	-	Yes			
	Yes	Yes	27 / 22b	Aux Bldg	A,B
flow	-	Yes			
	Yes	Yes	27A / 30	Aux Bldg	A,B,
	Yes	Yes			
	Yes	Yes			
	Yes	Yes			
flow	-	No			
flow	-	No			

Potential Bypass Path	Remarks	
Yes		
		59
Yes	Penetrations 137A & B and 141 are located on the same pipe run.	
		57
Yes	Penetrations 138A & B and 142 are located on the same pipe run.	
Yes		
Yes		
Yes		59



30. POST ACCIDENT INERTING SYSTEM



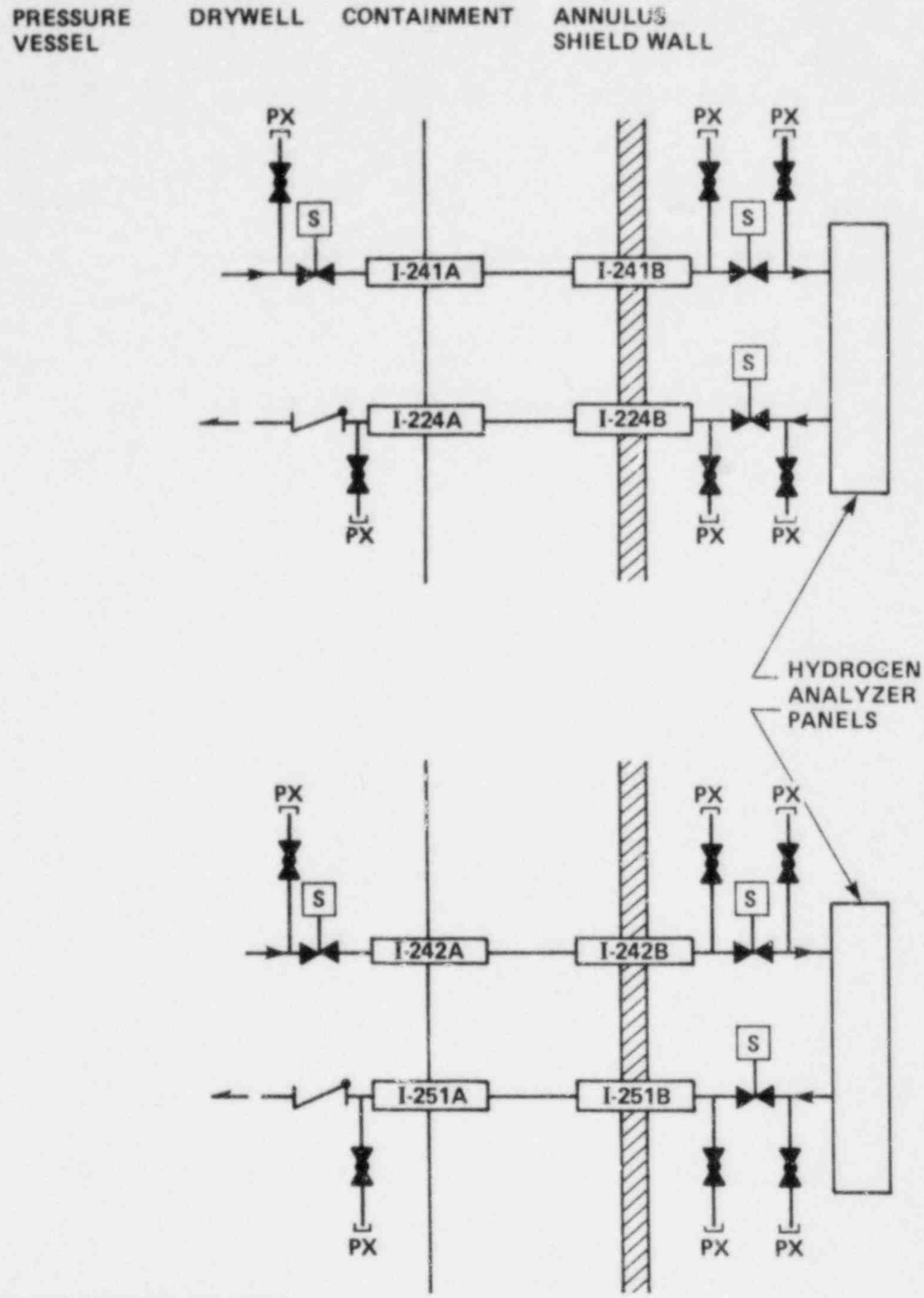
31. RWCU SAMPLING SYSTEM

AM. NO. 59, (6/81)

HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

CONTAINMENT ISOLATION
VALVE ARRANGEMENTS
SHEET 27A OF 31

FIGURE 6.2–26



CONTAINMENT HYDROGEN ANALYZER SAMPLING SYSTEM

AM. NO. 59, (6/81)

HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

CONTAINMENT ISOLATION
VALVE ARRANGEMENTS
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FIGURE 6.2–26

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ing the valves and monitoring the valve closure via the annunciators in the Control Room (10% and 90% closed indicators are provided). 37(G)

j) Manual Reset Capability

Isolation Valves classified as non-essential, will require manual, 57
operator initiated reset upon clearance of all automatic isolation signals. Isolation valves classified as intermediate will require manual operator initiated reset without clearance of the accident closure signal, but will not be possible in the presence of a system failure signal. Isolation valves classified as essential will receive no automatic containment isolation signals but will be manually controlled by the operator and/or required system performance interlocks. 59

TABLE 7.5-0 (Cont'd)

	VARIABLE	VARIABLE TYPE	RANGE	CATEGORY	PANEL NUMBER	COMMENTS	
26	Effluent Ra... activity	C				See Table 12.2-5a	57
27	Main Feedwater Flow	D	0-110% design flow	3	P680-03D		
28	Condensate Storage Tank Level	D	Bottom to Top	3	(Later)		
29	Containment Spray Flow	D	0-110% design flow	2	P601 P601		59
30	Drywell Pressure	D				Previously listed as Variable 8	
31	Suppression Pool Water Level	D				Previously listed as Variable 18	
32	Suppression Pool Water Temperature	D				Previously listed as Variable 1a	
33	Drywell Atmosphere Temperature	D	40-440^{o}F	2	P871 P872		57
35	Main Steamline isolation valves leakage control system pressure	D	0-15" w.g. 0-5 psia	2	P601-19B		

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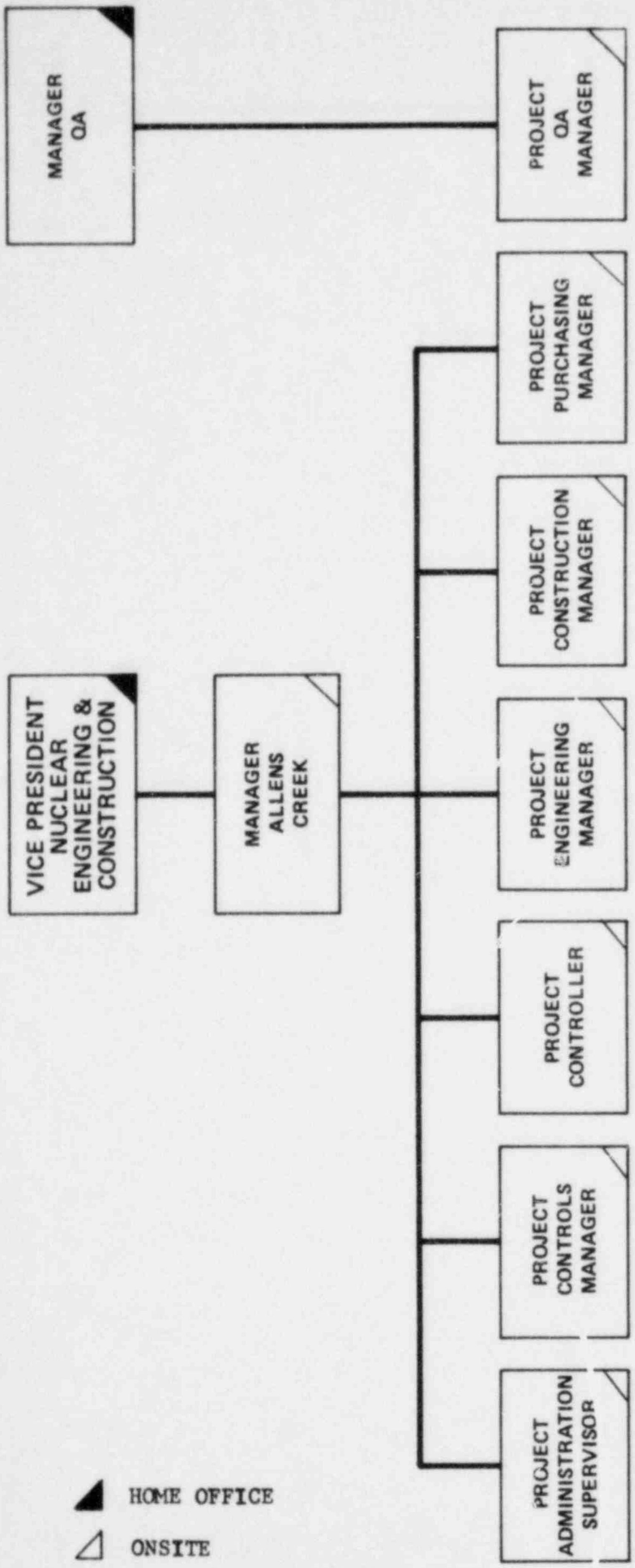


AM. NO. 57, (5/81)

HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

ALLENS CREEK
PROJECT MANAGEMENT ORGANIZATION
FIGURE 13.1–2

13.3.2.5 Planning Responsibility

Houston Lighting & Power Company, the State of Texas and local counties presently have qualified personnel assigned responsibility for the development of fixed nuclear facility emergency plans. The final ACNGS Emergency Plan, and the State and local plans approved by the NRC and FEMA, will describe these persons responsible for maintaining, reviewing, distributing and updating applicable emergency plans. HL&P will ensure that these responsibilities are executed. 57

13.3.2.6 Location of Key ACNGS Personnel During Alert or Greater Emergencies

For Alert Emergencies, the Technical Support Center (TSC) will be activated to support the Control Room. Other centers such as the Emergency Operations Facility (EOF) and Operations Support Center (OSC) may be brought to standby status. All of these facilities are activated for the Site Area and General Emergencies. Thus, many emergency functions will be performed in the TSC during Alert Emergencies and transferred to the EOF if the event escalates to a Site Area or General class event. 59

Table 13.3-2 provides, in matrix format, the location of key ACNGS personnel during Alert, Site Area and General Emergencies.

13.3.3 EMERGENCY CLASSIFICATION SYSTEM

Emergency conditions at ACNGS will be classified into four categories which will cover the entire spectrum of probable and postulated accidents. The four classes will be:

Unusual Event
Alert
Site Area Emergency
General Emergency

The Unusual Event and Alert categories are intended to provide early and prompt notification to the onsite and offsite emergency response organizations (see Section 13.3.2) that minor events have occurred or are in progress which could lead to more serious consequences if the plant status is in some way further complicated or which might be indicative of more serious conditions which are not yet fully realized. The Site and General Emergency categories are intended for more severe situations, which indicate that significant offsite effects are likely and require immediate action from both onsite and offsite emergency response organizations. The identification of a Site or General emergency should be assessed as quickly as possible and steps taken to mitigate the event and its effects and returning the plant to a safe status. 55

The decision to declare a particular emergency class is the responsibility of the Operating Supervisor at ACNGS, as Emergency Director. His decision will be based, to the extent feasible, on readiy available information about plant and offsite conditions which would indicate potential or actual hazards. The final ACNGS Emergency Plan will specify the criteria for declaring each emergency classification, as well as the provisions for upgrading the classification level and the corresponding response in the event of change of severity of the emergency condition.

13.3.3.1 Unusual Event

Events within the Unusual Event class will represent off-normal conditions in and around the plant. These events will not, by themselves, constitute significant emergency conditions and will have no offsite radiological consequences. Some of these events could, however, indicate a potential degradation in the level of plant safety and/or could escalate to a more severe condition if appropriate onsite action is not taken.

The primary purpose of notifying offsite agencies of an Unusual Event is to put the offsite response organization on standby, provide them with current information and provide unscheduled testing of the offsite communication link. An Unusual Event classification would initiate the augmentation of the onsite response resources to assist in the assessment and mitigation of the event. Recommendations will specify that no offsite actions are necessary.

13.3.3.2 Alert

Events within the Alert class will indicate an actual or potential degradation in the level of plant safety. The purpose of declaring this emergency class will be to assure that offsite emergency personnel and monitoring teams are ready to respond if needed. This class may also serve as an unscheduled test of the activation of the onsite and offsite emergency response facilities and the related communication systems. The response of offsite agencies will be to bring key elements of the emergency response organization into standby status, including offsite monitoring teams. 55

This class of emergency will also initiate the activation of the Technical Support Center and the Operations Support Center. The near-site Emergency Operations Facility will be brought to standby status.

In addition to the manning of the onsite response facilities, this emergency class might require that radiological and meteorological assessments be made and reported to the offsite agencies. No public action would be recommended in this emergency class. As a precautionary measure, visitors to the Allens Creek Lake and State Park will be evacuated.

13.3.3.3 Site Area Emergency

Events within the Site Area Emergency class involve actual or probable major failures of plant functions needed for protection of the public. The purpose of declaring the Site Area Emergency class is to assure the manning of all emergency response facilities, the dispatching of monitoring teams and the assembling of personnel required for evacuation if such action becomes necessary. Declaration of this emergency class will also be used as a means of informing the offsite agencies and the public that significant events are taking place. The response of offsite agencies following such a declaration will be to consider implementing protective actions and to assess information the public. HL&P will work closely with appropriate State and local governments to ensure that, before the ACNGS begins operation, the capability will exist to notify the public in the 10-mile EPZ within 15 minutes of notification to offsite organizations. An evaluation will be made to determine the specific system which will provide this capability. This system may include sirens, in-residence tone alert devices, alarms connected to electric meters and/or multiple telephone call-up techniques.

Arrangements will be made for broadcasting emergency instructions to the public via radio and/or television following the initial notification. The public will have received prior information as to how they will be notified in the event of an accident and what protective actions might be taken. 55

13.3.5 EMERGENCY RESPONSE FACILITIES

Emergency facilities, as well as special systems, will be established at and near the ACNGS for assessing an event, directing response and recovery efforts, mitigating accident consequences and informing the public. The facilities and systems will be described in detail in the final ACNGS Emergency Plan.

13.3.5.1 Control Room

During an emergency, the Control Room will be the location in which actions are taken primarily to bring plant systems under control. The Control Room is the location in which an accident event is initially recognized, classified and assessed, and notification procedures initiated. The Control Room will have communications with all other emergency facilities and will be equipped with terminals of data systems. The Control Room, being inside the plant, is designed as a totally safety grade facility.

13.3.5.2 Technical Support Center

An Onsite Technical Support Center (TSC) will be provided. Details of the TSC are described below. 59

13.3.5.2.1 TSC Function 57

The functions of the TSC are to:

a) Provide a location for plant management and technical support personnel to work to support operations personnel during emergency conditions.

b) Perform EOF functions during emergencies requiring EOF activation until the EOF is fully manned and functional.

It is expected that the TSC complex will be used for routine plant functions. For example, the NRC office may be the Resident IE Inspectors office and the work area with its displays may be used for training. Such uses will not interfere with rapid activation of the TSC for its emergency functions. 57

13.3.5.2.2 TSC Location

The TSC is located on the northeast side of the Control Building at el.187', as shown on Figure 1.2-3. Walking time between the TSC and Control Room is well under two minutes. The two areas and the hall between them are in the same ventilation envelope as described in Section 13.3.5.2.6, so there would be no need to don protective gear to pass from one area to the other. There are no major security barriers between the two areas. 59

57

13.3.5.2.3 TSC Staffing and Training

This will be provided in the ACNGS Final Emergency Plan.

13.3.5.2.4 TSC Size

The TSC is a complex consisting of the following directly adjacent areas. A working space of approximately 75 sq. ft. per person was used as a basis for the layout of the TSC. The design meets the minimum requirements of 25 persons as required by NUREG-0696, February 1981. The layout meets the minimum requirements for square footage per person, and the location of the TSC (adjacent to the Control Complex) adds additional benefit.

a) Open working area for 22 people of approximately 1580 sq. ft. divisible into up to 5 separate rooms by the use of moveable room dividers. This area also contains the SPDS displays and other plant data displays. These displays, as a minimum, will be Type A, B, C, D, and E variables specified in Regulatory Guide 1.97 Rev. 2 and meteorological variables as specified in Regulatory Guide 1.23.

   59

   Copying equipment and plant data displays are located in this work area. A portion of this work area is available to be separated out as a conference room, if required. The remaining work areas will each have a data display to allow addressing the parameters in the data acquisition system. This area will be used by technical personnel to enable them to carry out their function of supporting the operators in the Main Control Room.

b) Office for NRC representatives of approximately 150 sq. ft. This area provides working space for 2 NRC people as well as being used for private NRC consultations.

   Data is available to NRC personnel, as well as other TSC personnel, in any of the sections of the work area described in (a) above. Hard copies of any display can be made by the video copiers or the ERIS data acquisition system can be requested to print information on the printer located in the work area.

c) Communications equipment room of approximately 158 sq. ft. The area is normally manned by two people. These persons will perform tasks such as teletype operations, telephone operator functions, and secretarial functions.

d) Document Storage Room of 180 square feet.

e) Living Area, including kitchen, sleeping area, supply storage, and repair shop for storing tools and spare parts for the TSC instruments displays. This area is 900 sq. ft. and is not considered part of the TSC working area. 59

f) Secondary Alarm Station. This station will be continuously operational.

13.3.5.2.5 TSC Structure

The Control Building in which the TSC is located is designed to withstand the full range of natural phenomena specified for safety-related structures for ACNGS. This exceeds the structural requirements for the TSC.

13.3.5.2.6 TSC Habitability 57

The TSC is served by the Control Room Ventilation System, so it is habitable to the same extent as the Control Room. See Section 6.4.

13.3.5.2.7 TSC Communications

The TSC will be provided with reliable voice communications to the Control Room, OSC, EOF and NRC. Details of TSC communications will be provided in the ACNGS Final Emergency Plan.

13.3.5.2.8 TSC Instrumentation, Data System Equipment and Power Supplies

Plant data will be available for display in the TSC. The set of parameters to be displayed in the TSC has not been finalized, as HL&P intends to abide by the results of the BWROG efforts in this regard when approved by the NRC. As a minimum, the SPDS (see Section 7.5.1.6) and post-accident monitoring instrumentation (see Appendix C) will be available in the TSC. This information will be displayed on a minimum of four CRT's, three located in the Main Work Area and one in the TSC conference room, (see Figure 1.2-33 and 1.2-35. In addition, a hard-copy printer will be available in the Main Working Area, as well as two video copiers. 59

The TSC displays are not Class 1E or seismically qualified, but are provided with reliable backup power from the BOP diesel generator.

13.3.5.2.9 Records Availability and Management 57

The TSC will have a records storage area in which up-to-date plant records deemed useful for accident or off-normal condition diagnostics will be stored. A list of the types of these records will be provided in the ACNGS Final Emergency Plan.

13.3.5.3 Emergency Operations Facility

An Emergency Operations Facility will be provided. Details of the EOF are described below.

13.3.5.3.1 EOF Function

The ACNGS Emergency Operations Facility (EOF) will be controlled and operated by HL&P and will serve as the location for performing the functions of:

a) management of overall emergency response,

b) coordination of radiological and environmental assessment,

c) determination of recommended public protective action, and

d) coordination of emergency response activities with Federal, State and local agencies.

The EOF will be activated for Site Area and General Emergencies, and will be brought to standby status for the Alert Emergency. 59

It is anticipated that the EOF will be used as the post-accident recovery center.

The EOF space may be used for other purposes during normal operations, however, provisions will be set forth to assure that the emergency functions of the EOF are in no way degraded by those activities and that all necessary systems meet required availability. These provisions will include adequate security protection of the facility during normal and emergency conditions.

13.3.5.3.2 EOF Location

The location of the EOF will be described in more detail in the ACNGS Final Emergency Plan. In selecting the actual location, consideration will be given to the following factors:

a) Whether the location provides optimum functional and availability characteristics for carrying out the licensee functions specified for the EOF (i.e., overall strategic direction of licensee onsite and support operations, determination of public protective actions to be recommended by the licensee to offsite officials, and coordination of the licensee with Federal, State and local organizations)

b) Whether the EOF functions would be interrupted during radiation releases for which it was necessary to recommend protective actions for the public to offsite officials.

The siting of the EOF will be coordinated with State and local authorities.

It is anticipated that the EOF will be located on the site of HL&P's Service Center at Katy, Texas (Figure 13.3-1). The location is on property owned and controlled by HL&P and is approximately 19 miles from the ACNGS. This location is easily accessible from the Interstate 10 Highway.

Presently under construction at the Katy Service Center is an additional 15,000 square feet of office and conference room space. A plot plan of the service center is shown on Figure 13.3-2. The present service center office will be modified to include a coffee bar/meeting room of about 875 square feet.

The additions and modifications are scheduled for completion by early 1982.

A conceptual floor layout for the EOF using the modified Katy Service Center is shown on Figures 13.3-3 and 13.3-4. 59

Should the location of the EOF be changed at a later date, the new location and its description will be submitted to the NRC for approval prior to making the change.

13.3.5.3.3 EOF Staffing and Training

When the EOF is activated, it will be staffed by HL&P, Federal, State, Local and other emergency personnel designated by the ACNGS Emergency Plan. A designated senior HL&P official will manage licensee activities in the EOF to support the HL&P official managing activities in the Technical Support Center and the senior reactor operator serving as shift supervisor in the Control Room.

The EOF will be staffed to provide the overall management of licensee resources and the continuous evaluation and coordination of licensee activities during and after an accident. Upon EOF activation, designated personnel will report directly to the EOF to achieve full functional operation within 1 hour. The EOF staff will include personnel to manage the licensee onsite and offsite radiological monitoring, to perform radiological evaluations, and to interface with offsite officials. The specific number and type of personnel assigned to the EOF may vary according to the emergency class. The staffing for each emergency class shall be fully detailed in the ACNGS Final Emergency Plan. Operating procedures and staff training in the use of data systems and instrumentation will contain guidance on the limitations of instrumentation including whether the information can be relied upon following serious accidents. The EOF staff will participate in EOF activities drills, conducted periodically in accordance with the Emergency Plan. These drills will include operation of all facilities that will be used to perform the EOF functions.

13.3.5.3.4 EOF Size

The EOF building or building complex will be large enough to provide the following:

a) Working space for the personnel assigned to the EOF as specified in the ACNGS Final Emergency Plan, including State and local agency personnel. A working space of approximately 75 sq. ft. per person will be used as a basis for size and layout of the EOF. The conceptual EOF layout provided on Figure 13.3-3 assumes approximately 25 persons from the licensee, 10 persons from State and Local agencies, 9 persons from the NRC and 1 person from FEMA.

b) Space for EOF data system equipment needed to transmit data to other locations;

c) Sufficient space to perform repair, maintenance, and service of equipment, displays, and instrumentation; 59

d) Space for ready access to communications equipment by all EOF personnel who need communications capabilities to perform their functions;

e) Space for ready access to functional displays of EOF data;

f) Space for storage of plant records and historical data or space for means to readily acquire and display those records;

g) A separate room to accommodate at least 5 NRC personnel will be provided;

h) Conference rooms;

i) A space to brief select groups of about 25 persons, such as a press pool or government officials;

j) A secured entrance

k) Sufficient space outside for parking private vehicles, mobile laboratories and trailers. Electrical power and sanitary hook-up will be available for mobile laboratories and trailers.

13.3.5.3.5 EOF Structure

The EOF structure will be well engineered for the design life of the ACNGS, in accordance with the criteria in NUREG-0696, Table 2.

13.3.5.3.6 EOF Communications

The EOF will have reliable voice communications facilities to the TSC, the control room, NRC, and State and local emergency operations centers. The normal communication path between the EOF and the control room will be through the TSC. The primary functions of the EOF voice communications facilities will be:

a) Communications with the designated senior licensee manager in charge of the TSC,

b) manage licensee emergency response resources,

c) coordinate radiological monitoring,

d) coordinate offsite emergency response activities, and 59

e) disseminate information and recommended protective actions to responsible government agencies.

The EOF voice communications facilities will include reliable primary and backup means of communication. Voice communications may include private telephones, dedicated telephones, commercial telephones and radio. A means for EOF telephone access to commercial telephone services that bypasses any local telephone switching facilities that may be susceptible to loss of power during emergencies will be provided. Spare commercial telephone lines to the plant will be available.

The EOF communication system will include designated telephones for use by NRC personnel. The licensee will also furnish the access facilities and cables to the NRC for the Emergency Notification System (ENS) and the Health Physics Network (HPN) telephones.

Facsimile transmission capability between the EOF, the TSC, and the NRC Operations Center shall be provided.

13.3.5.3.7 EOF Instrumentation, Data System Equipment, and Power Supplies

The Emergency Response Information System (ERIS) is a data acquisition system that provides data acquisition and information display capabilities in various locations such as SPDS, TSC, EOF, OSC, and other offsite facilities.

The EOF will contain equipment for the acquisition, display, and evaluation of all radiological, meteorological and plant system data necessary to determine protective measures recommended to offsite authorities. This equipment will also be used to evaluate the magnitude and effect of actual or potential radioactive releases and to project offsite doses.

Data will be transmitted to the EOF from either the existing plant process computer system or a separate computer-based system, using CRTs located in the EOF. An evaluation is currently underway to determine which data acquisition system will be used. The evaluation is scheduled to be completed prior to the software development for the existing plant process computer system. Video graphics will be utilized to provide graphic displays of necessary plant status information. Video copier(s) will be provided in the EOF to ensure that hard copies of video displays can be obtained when needed.

The data acquisition system will gather, store, and display the data needed in the EOF to analyze and exchange information on plant conditions with the designated senior licensee manager in charge of the TSC. The system will perform these functions independently from actions in the control room without degrading or interfering with control room and plant functions.

Trend-information display and time-history display capability will be provided. The SPDS will be displayed in the EOF.

The EOF data set shall include radiological, meteorological, and other environmental data as needed to:

a) Assess environmental conditions,

b) Coordinate radiological monitoring activities, and 59

c) Recommend implementation of offsite emergency plans.

As a minimum EOF data set, sensor data of the Type A, B, C, D, and E variables specified in Regulatory Guide 1.97, Revision 2, and of those meteorological variables specified in proposed Revision 1 to Regulatory Guide 1.23, "Meteorological Measurements Programs in Support of Nuclear Power Plants," and in NUREG-0654, Revision 1, Appendix 2, will be available for display in the EOF. All data that are available for display in the TSC, including data transmitted from the plant to NRC, will be part of the EOF data set. The accuracy of data displays in the EOF will be equivalent to that for the data displayed in the TSC.

Data storage capability will be provided for the EOF data set. The sample frequency will be chosen to be consistent with the use of the data. Capacity to record at least two weeks of additional post-event data with reduced time resolution will be provided. A sufficient number of data display devices shall be provided in the EOF to allow all EOF personnel to perform their assigned tasks with unhindered access of:

- Plant systems variables,
- In-plant radiological variables,
- Meteorological information, and
- Offsite radiological information.

The total EOF data system shall be designed to achieve an operational unavailability goal of 0.01 during all plant operating conditions above cold shutdown.

The EOF electrical equipment load will not degrade the capability or reliability of any safety-related power source. Circuit transients or power supply failures and fluctuations will not cause a loss of any stored data vital to the EOF functions.

13.3.5.3.8 Records Availability and Management

The EOF will have ready access to up-to-date plant records, procedures, and emergency plans. The EOF records will include, but shall not be limited to:

- Plant technical specifications,
- Plant operating procedures,
- Emergency operating procedures,
- Final Safety Analysis Report,
- Up-to-date records related to licensee, State, and local emergency response plans, 59
- Offsite population distribution data,
- Evacuation plans,
- Environs radiological monitoring records, and
- Licensee employee radiation exposure histories
- Up-to-date drawings, schematics and diagrams showing conditions of plant structures and systems down to the component and in-plant locations of these systems.

These records will either be stored and maintained in the EOF (such as hardcopy on microfiche) or shall be readily available via transmittal to the EOF from another records storage location.

13.3.5.4 Operations Support Center 55

Appropriate space will be designated onsite for the assembly of operations personnel whose support is required in or near the plant, but not in the Control Room or TSC. The preliminary location of the OSC is in the personal 59
access building as shown on Figures 1.2-38a and 1.2-38b. Supplies such as 57
protective clothing, respiratory protection, portable lighting and communications equipment will be provided. The OSC will be described in 55
detail in the final ACNGS Emergency Plan.

13.3.5.5 News Media Center

HL&P will provide a location at or near the EOF to serve as a News Media Center (NMC) in which to conduct press conferences and briefings during an emergency. The NMC will be activated for the Site Area and General Emergency levels and will be brought to standby status for the Alert Emergency. The NMC will be large enough to accommodate 300-400 news media representatives. A backup location outside the plume exposure EPZ will be available, such as an auditorium or civic center. A small briefing room will be made available in the EOF in which to conduct briefings with small select groups. In the NMC, information packets or "press kits" will be available providing information 55
about the licensee, the plant and plant surroundings. Visual aids will be provided.

HL&P will designate an official company spokesperson to interface with the news media and the spokespersons of offsite response organizations. HL&P spokespersons will be trained in conducting press conferences and briefings and will be knowledgeable of plant operations and the Emergency Plan.

13.3.5.6 Safety Parameter Display System

The SPDS is described in Section 7.5.1.6. 57

13.3.5.7 Data Transmission 55

The ACNGS will be equipped with the capability to transmit plant data to the TSC and EOF. The design of this system will be described in the final ACNGS Emergency Plan. 57

13.3.5.8 First Aid Facility

A first aid room equipped with the first aid equipment and supplies which are appropriate for a major industrial facility will be provided at the ACNGS. At least one individual onsite will be trained and qualified in advanced first aid methods.

13.3.5.9 Decontamination Facility 55

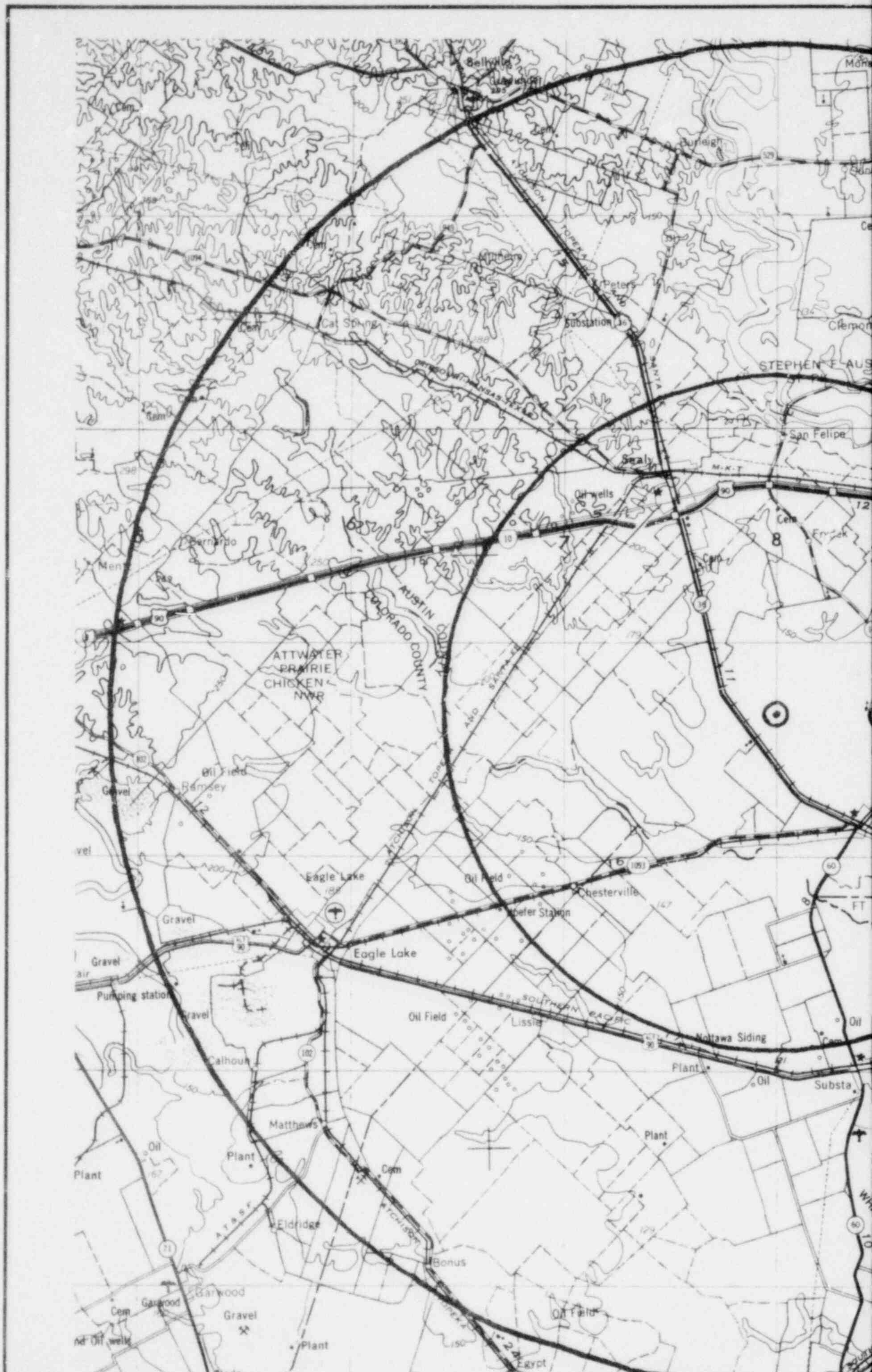
Personnel decontamination facilities, consisting of showers and sinks which drain to the radwaste system, will be provided. ACNGS personnel will be trained in decontamination methods. First aid to injured individuals will, in most cases, be performed in conjunction with any necessary decontamination. However, if immediate treatment of the injury is deemed necessary, that treatment will take precedence over decontamination. This philosophy will also extend to transportation and offsite treatment of contaminated, injured individuals.

TABLE 13.3-2

LOCATION OF KEY ACNGS PERSONNEL DURING ALERT OR GREATER EMERGENCIES

		Emergency Class		
	Personnel	Alert	Site Area	General
A.	Emergency Director	TSC	EOF	EOF
B1.	Radiation Protection Mgr.	TSC	TSC	TSC
B2.	Radiological Emergency Mgr.	(EOF)	EOF	EOF
C.	Public Affairs Spokesperson	TSC	EOF	EOF
D.	Recovery Mgr.	TSC	EOF	EOF
E.	Vendors, A-E's, Construction	TSC(EOF)	TSC or EOF	TSC or EOF 59
F.	Site Support Mgr.	TSC(EOF)	EOF	EOF
G.	Consultants, Mobile labs, etc.	TSC(EOF)	EOF	EOF
H.	Fire Brigade, Damage Control, Repair, etc.	OSC	OSC	OSC
I.	Federal, State & Local Agencies	TSC(EOF)	EOF	EOF
J.	NRC Site Team	TSC(EOF)	TSC & EOF	TSC & EOF

Note: Centers in parentheses indicate that this center is normally not activated for the emergency class, but may be activated for use by the personnel if necessary.

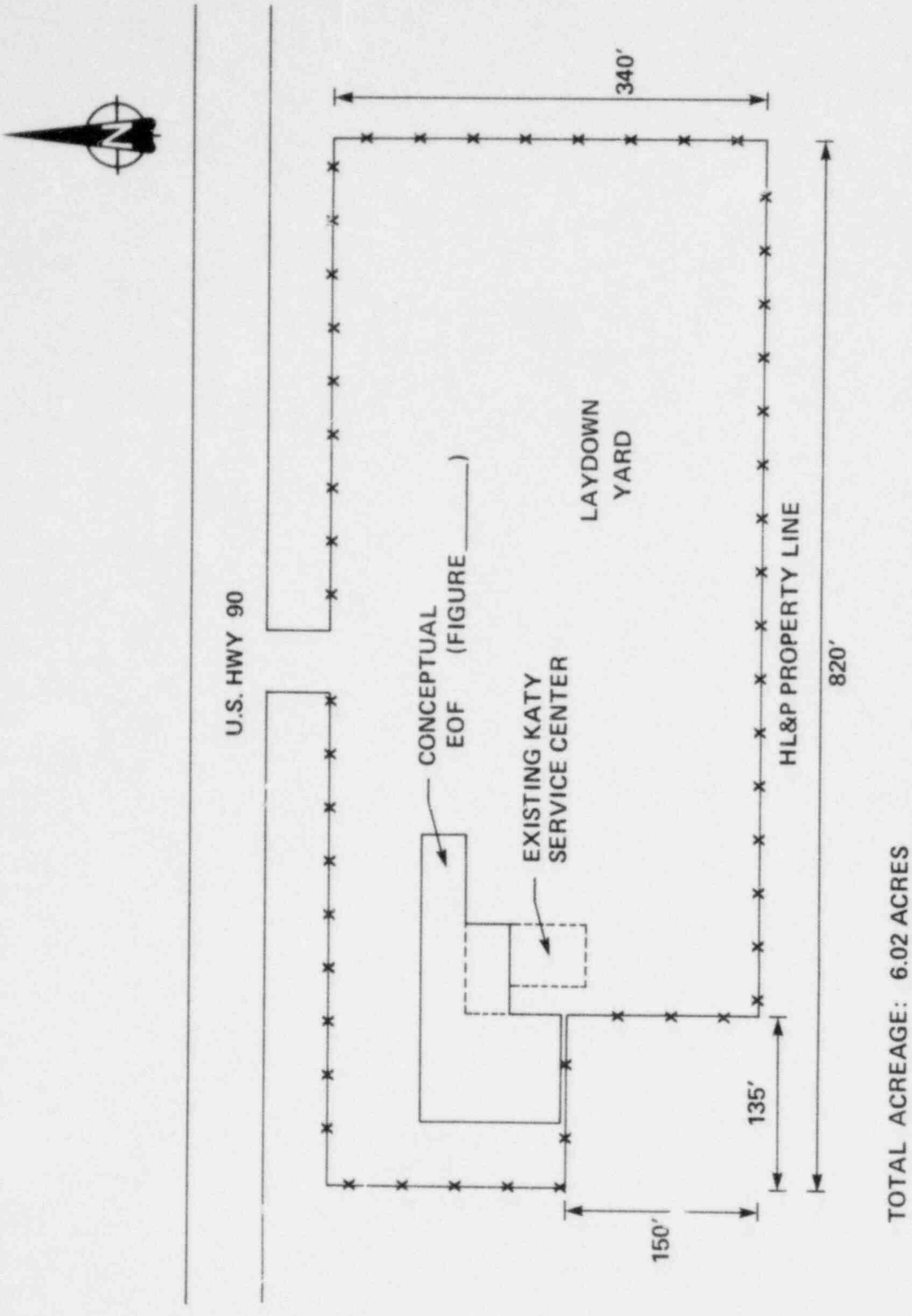

Bellville
Burleigh
Cat Spring
Peters
Substation
STEPHEN F. AUS
San Felipe
Sealy
Oil wells
M-K-T
Frydek
Clemons
Bernardo
AUSTIN
COLORADO COUNTY
ATTWATER
PRAIRIE
CHICKEN
NWR
Oil Field
Ramsey
Gravel
Eagle Lake
Oil Field
Chesterville
Eagle Lake
Pumping station
Oil Field
Lissie
SOUTHERN PACIFIC
Nottawa Siding
Plant
Oil
Substa
Calhoun
Matthews
Plant
Cem
Plant
Eldridge
Bonus
Garwood
Gravel
Oil Field
Plant
Egypt
Spanish Camp
Nada

AM. NO. 59, (6/81)

HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

LOCATION OF EOF

FIGURE 13.3-1



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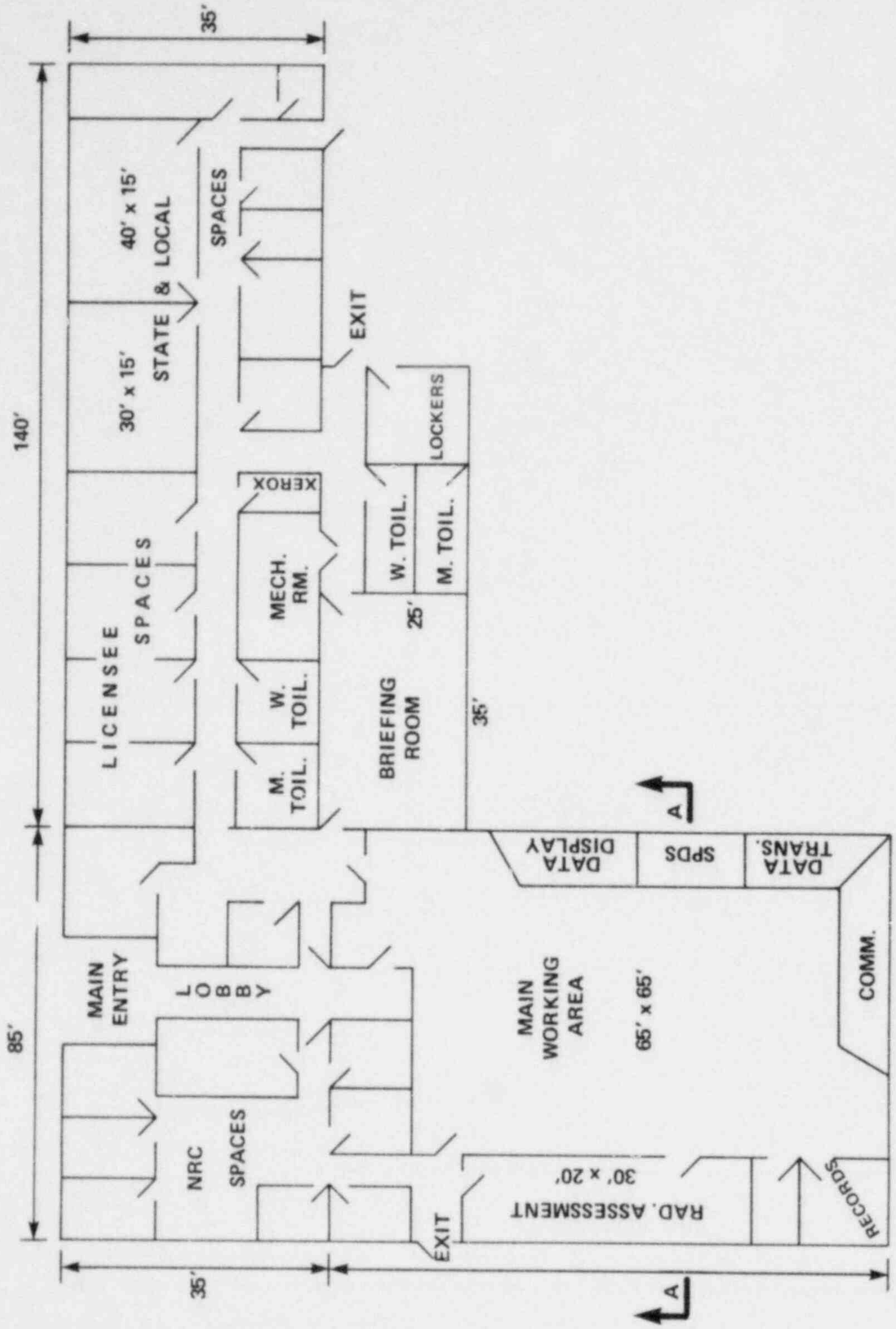
HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

KATY SERVICE CENTER
PLOT PLAN

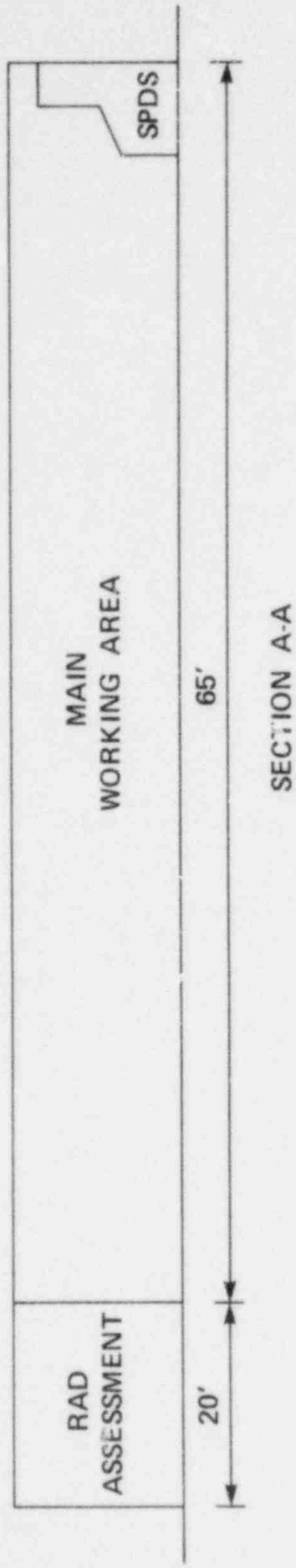
FIGURE 13.3–2



AM. NO. 59 6/1/81 (D) - DESIGN

HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

ACNGS EMERGENCY OPERATIONS FACILITY (CONCEPTUAL)
FIGURE 13.3-3



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Unit 1

ACNGS EOF CONCEPTUAL EOF
SECTIONAL VIEW

FIGURE 13.3–4

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RELIABILITY ANALYSIS PROGRAM

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APPENDIX 15B

ALLENS CREEK NUCLEAR GENERATING STATION RELIABILITY ANALYSIS PROGRAM 57

1.0 INTRODUCTION

Item II.B.8(1) of the "Proposed Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses" (NUREG 0718) requires that a site/plant specific probabilistic risk assessment be performed, and that the results be evaluated and considered in the design of the facility.
The following discussion describes the proposed program to be applied to the 59
Allens Creek Nuclear Generating Station (ACNGS) in response to NUREG 0718 including an outline of the program scope, methodology, schedule, quality
assurance procedures and the means by which the reliability analyses will be 57
integrated into the ongoing design process.

2.0 PROGRAM OBJECTIVE

The aim of the Reliability Analysis Program is to seek improvements in the 59
reliability of core and containment heat removal systems that are significant and practical and do not impact excessively on the plant. The reliability Analysis Program will be specifically structured to meet this objective, recognizing the advanced state of design (70% engineering complete) and fabrication (major plant components fabricated and in storage).

3.0 SCOPE

The ACNGS Reliability Analysis Program is similar in scope to the Interim
Reliability Evaluation Program (IREP) being performed by NRC for several oper- 57
ating plants, except that it will include the calculation of fission product
release quantities for the accident release categories. It will focus on core 59
and containment cooling systems together with all pertinent support systems. Individual accident sequences and their probabilities will be analyzed to
identify the initiating events and plant system and component failures which 57
are the dominant contributors to core damage risk. Event tree and fault tree
analyses will be performed for core and containment cooling systems and all 59
support systems to identify common-mode failure mechanisms.

The list of initiating events to be considered will be determined during the
initial phase of the study. However, as a minimum, the initiating events will 57
encompass loss of coolant accidents (small, intermediate and large) and transient events including loss of feedwater, loss of offsite power and turbine trip. The interdependence of support systems will be considered with any
initiating event that may lead to offsite radiological releases in excess of 59
10CFR100. The following key safety-related systems will be included in the

program:

- Reactor Core Isolation Cooling System
- Residual Heat Removal Cooling System
- High Pressure Core Spray Cooling System
- Low Pressure Core Spray System
- Automatic Depressurization System
- Containment Spray System 57
- Containment Isolation System
- Reactor Protection System
- Electric Power (AC and DC) System
- Essential Service Water System
- Essential Chilled Water System
- Essential HVAC Systems
- Standby Liquid Control System

Other systems are expected to be added to the above representative list as the core and containment cooling systems are evaluated with respect to interactions and dependencies.

HL&P has identified an alternative heat removal system based on a study recently performed to determine design features which would reduce the probability of occurrence of a degraded core. Results of the PRA study will be inspected to determine whether an obviously superior alternative is feasible.

A preliminary outline of the report is given in Table 15B-1.

4.0 PROGRAM ORGANIZATION AND RESPONSIBILITIES

59

HL&P is responsible for the PRA study and will ensure that the study will be performed by engineers who are highly qualified and experienced in risk assessment methodology. HL&P engineers will be actively involved in this study and will provide direction in the development of the program. Prior to decisions relating to identified design improvements, HL&P will appoint a third party to conduct a peer review on the study.

HL&P retains ultimate authority and responsibility of implementing design improvements as a result f this study.

5.0 METHODOLOGY

The approach to be used in the program will employ event tree/fault tree methodology similar to that used in WASH 1400 and other comprehensive plant 57 risk studies. The major tasks involved are discussed below.

5.1 INITIATING EVENT SELECTION

A list of initiating events will be established that, together with system 59
failures, have the potential for causing core damage and offsite radioactivity releases. This will be accomplished through a screening of the accidents and transients identified in PSAR Chapter 15 and in WASH 1400 to identify the basic
set of initiating events requiring operation of the key safety systems for core 57
protection and release mitigation. The frequency of these initiating events will be estimated based on available data including WASH-1400, EPRI NP-801, and, where it exists, pertinent plant specific or site-specific information (e.g., frequency of loss of offsite power on the HL&P system).

Initiating events will be classified according to the safety system response required for accident mitigation. Initiating events having the same or similar safety system response requirements will be grouped together and a unique event tree will be developed for each such group of events. Initiating events including LOCA, transients and recirculation line breaks will be evaluated. 59
Failures during cold shutdown, severe natural phenomena and fire will also be considered.

5.2 EVENT TREE DEVELOPMENT

For each unique group of initiating events, an event tree will be constructed, identifying the safety systems required to mitigate the event and 57
the expected effect on ability to maintain core and containment integrity given success or failure of each safety system involved. The full event tree will be reduced to reflect safety system interdependencies and required sequences of operation.

5.3 SYSTEM FAILURE MODES AND EFFECTS ANALYSIS (FMEA) 59

For each safety system involved, a FMEA will be conducted to identify and tabulate component and common cause failures and their effect on system operability for each initiating event. The FMEA will provide documentation of the basis for inclusion or exclusion of specific failure modes in the system fault 57
tree analysis. Failure modes will include mechanical and electrical faults, operator error, maintenance or testing outages, etc. Particular attention will be paid to potential common cause failures which could disable multiple components. Common cause failure mechanisms to be investigated include environmental factors, operator or maintenance errors, passive failures and system interactions.

5.4 SYSTEM FAULT TREE ANALYSIS (FTA) 59

Using the FMEA as input, fault trees will be constructed for each safety system identifying the failures (basic events) and their logical combinations which will result in system unavailability (top event). The fault tree will be 57
analyzed to determine the minimal cut sets and failure combinations which are the dominant contributors to system unavailability. Using the appropriate component failure data, a quantitative assessment of overall system unavailability and of the dominant cut sets will be performed.

The fault tree analysis will be performed using a computer program to perform
cut set determination and quantitative analysis and to provide computer-
generated graphical representation of the fault tree using standard logic 57
symbols.

5.5 DATA BASE DEVELOPMENT 59

A component failure data base for use in system fault tree analysis will be
developed from recognized reference sources including WASH-1400 and IEEE-500.
In addition, prototype-specific failure data will be requested from vendors of
selected components (e.g., diesel generators) being supplied to ACNGS. The
data base will identify the types of components and estimated median failure
rates on demand and, where appropriate, per hour of continuous operation. 57
Error ranges will be assigned to each median value to reflect the uncertainty
in the data base. The data base will include methodologies to adjust failure
data to account for varying testing and surveillance strategies. Test and
maintenance unavailability contributions will be included based on proposed
Technical Specifications operating and maintenance procedures.

Human error rates will be estimated for required or corrective actions by
control room operator and for maintenance or testing operations which are
included as failure modes in the system fault trees. Available human error
and performance data, including those provided NUREG/CR 1278 will be used.

5.6 ACCIDENT SEQUENCE PROBABILITIES 59

The unavailability of each system will be calculated by inputting the
appropriate failure rate data into the system FTA. The various accident
sequences, as represented by the branches on the event trees will then be
quantified by inputting the system failure probabilities determined from the 57
quantitative FTA. Each individual accident sequence will be classi-
fied according to release category and the total probability of a given
release category will be obtained by the summation of all accident sequence
probabilities assigned to that category.

5.7 ACCIDENT SEQUENCE FISSION PRODUCT RELEASES 59

The release categories of WASH-1400 will be re-examined and modified if neces-
sary to account for any changes due to the Mark III containment configura-
tion. A radiological release source term to the environment will be calcula-
ted for each release category. 57

5.8 UNCERTAINTY ANALYSIS 59

All quantitative results will be reported in terms of point values of a probability distribution function, including expected (mean) or median (50th percentile) value and upper (95th percentile) and lower (5th percentile) uncer- 57
tainty bounds. These point values will be determined based on a propagation 59
of component failure data, including error ranges, through the fault trees and event trees. The uncertainty propagation will be performed using standard statistical distribution functions (e.g. lognormal) or numerical (e.g. Monte 57
Carlo) techniques.

5.9 SENSITIVITY ANALYSIS 59

The results of the study will be reviewed to identify the accident sequences which are the dominant contributors to overall risk and, within those sequences, the significant system and component failure modes. Comparisons with existing risk studies, including WASH-1400, will be made to identify and explain any significant differences.

The sensitivity of the results to assumptions regarding component or common 57
cause failures will be evaluated by varying the assumed failure rates of key basic events which appear with a high frequency in the dominant event sequences and determining the resultant effect on system failure rates and overall results.

6.0 SCHEDULE 59

The Reliability Analysis Program will commence in mid-1981. The initial phase of the program is expected to take approximately 15 months, as shown on Figure
15.B-1, and will consist of a base line reliability analysis of the present 57
ACNGS design. The overall study, including radionuclide release quantification, will be completed within two years of CP issuance.

The overall design of ACNGS is approximately 70% complete as of May 1981. Fabrication of the NSSS is rapidly nearing completion. Similarly, supporting balance of plant major features are in fabrication and would be expected to be essentially completed prior to completion of the PRA study.

59

If the ACNGS construction permit is received in the March-June 1982 period, the PRA study would be completed about March-June 1984. Design and fabrication will by then be essentially complete for core and containment heat removal systems and support systems. However, the results will be used on a case-by-case basis to determine whether major redesign, repurchasing, and refabrication are warranted.

7.0 APPLICATION OF RESULTS TO FINAL DESIGN 59

There are currently no established regulatory requirements or acceptance criteria for judging the acceptability of quantitative system reliability 57 analyses. Thus the need for implementing changes in design or operating, testing or maintenance procedures to achieve improvements in system reliability will be based on judgmental acceptance criteria which are not 57 directly related to licensing requirements. These acceptance criteria will be established during the initial phase of the program and will include both quantitative and qualitative considerations of potential design changes on plant cost, schedule and availability.

Following completion of the base line reliability analysis, the results will 57 be reviewed and various options available for improvement in reliability will be evaluated with respect to the established acceptance criteria. Recommendations will be made regarding changes in design or operating procedures and the reliability analysis will be revised to reflect those selected for implementation.

Routine design changes will also be evaluated on an ongoing basis. A determination will be made regarding the effect of any proposed design change on the reliability analysis results. If the change is expected to affect reliability, the results will be reviewed for acceptability and need for 59 further modifications determined. In this manner, the Reliability Analysis Program will be kept current with respect to design modifications and a mechanism will be in place to evaluate reliability related changes for acceptability as the design is finalized. 57

Results of the study will be utilized to improve reliability of component selection, specification and testing and to improve systems interaction. Results will also be used to identify improvements to be considered in the future for the following areas: maintenance procedures, operator training and operating feedback.

8.0 QUALITY ASSURANCE 59

Results of the study will be used to identify those areas where additional quality assurance activity would improve reliability. The results of the program, including all calculations will be subject to review and verification in accordance with normal ACNGS quality assurance program practices. 57 Documentation will be maintained current so that all results can be reproduced and all assumptions checked from original references.

TABLE 15B-1

OUTLINE OF RELIABILITY ANALYSIS REPORT

I. INTRODUCTION

II. SUMMARY

III. METHODOLOGY OVERVIEW

A. Event Trees

B. Fault Trees

C. Quantification of Accident Sequences

D. Containment Failure Analyses

E. Fission Product Release Analyses

F. Treatment of Uncertainties

IV. SYSTEM DESCRIPTIONS

A. Performance Requirements 59

B. Actuation

C. Environment Considerations

D. Dependency Diagrams for Support Systems

V. CORE MELT PROBABILITIES

A. Dominant Sequences

B. Dominant Cut-Sets

VI. PLANT MODIFICATIONS THAT ADDRESS DOMINANT SEQUENCES

A. Improvement in Reliability Expected

B. How Factored into Design, Equipment Purchase, Fabrication, Procedures, Operation, etc.

C. Basis for Not Implementing More Reliable Alternatives

TABLE 15B-1 (Cont'd)

VII. FISSION PRODUCT RELEASE ANALYSIS

A. Release Groups

B. Containment Failure Probabilities

C. Fission Product Release Fractions

D. Total Radioactive Release from Containment to Environment for the Various Release Groups. 59

VIII. APPENDICES (DETAILS OF STUDY)

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17.0 QUALITY ASSURANCE

Each section in Chapter 17 has been assigned suffixes to the section numbers to identify that the section is applicable to the respective organizations as follows: A - Applicant, Houston Lighting and Power Company (HL&P), B - Architect Engineer/Constructor, Ebasco Services Incorporated (Ebasco), C - Nuclear Steam Supply System and Nuclear Fuel Supplier, General Electric Company (GE). 33(U)

Ebasco and HL&P corporate management responsible for quality related activities (ie: engineering, construction, quality assurance) will periodically visit the Allens Creek site to evaluate and assess the quality attitude and motivation of managers, supervisors, technicians, foremen, craft personnel and other cognizant personnel. Seminars, group meetings, discussions, etc. will be held as necessary to ensure that personnel are aware of executive management support for the quality assurance program. 59 Ebasco and HL&P corporate management support is further delineated in the corporate policy statements in the front of the respective Nuclear QA Program Manuals.

A brief introduction has been prepared to introduce HL&P, Ebasco and GE, respectively.

17.0.A HOUSTON LIGHTING & POWER COMPANY

Houston Lighting & Power Company (HL&P) as the Applicant, has the Quality Assurance responsibility for design, engineering, procurement, fabrication, construction, preoperational testing and operation of the Allens Creek Nuclear Generating Station (ACNGS). Although HL&P will delegate certain of its QA activities and authority to its contractors, it retains responsibility for the QA program controlling all aspects of the ACNGS.

The HL&P Quality Assurance Program requires that HL&P, its prime contractors, subcontractors and vendors comply with the criteria established by 10CFR50 57 Appendix B. It is the intent of HL&P to comply with ANSI N45.2 and the applicable daughter standards and implementing Regulatory Guides. Furthermore, HL&P shall assure through programmatic direction that Ebasco and all of its subcontractors and suppliers performing nuclear safety related work comply with 10CFR50 Appendix B, ANSI N45.2, and the Regulatory Guides as referenced herein consistent with their scope of work. In addition, HL&P will comply with 10CFR50 Appendix A and Regulatory Guide 1.29, Revision 1 for the identification of items "important to safety," and these items 59 will be under the control of the QA Program. Programmatic direction is defined as the role of HL&P in establishing the program requirements and ensuring the adequacy of the prime contractors QA Program. The programmatic direction consists of review and approval of the system features initially and continued monitoring of those systems during implementation and further refinement or revision of the systems if the systems need strengthening. 57

The HL&P QA Program is described in the corporate Nuclear Quality Assurance Program Manual (NQAPM). The NQAPM requires the establishment of a Project Quality Assurance Plan for each project to describe the QA program to be implemented during the design and construction phase of each project and an Operational Quality Assurance Plan to describe the QA Program to be imple-

mented during the preoperational testing and operational phases of the ACNGS. The Project QA Plan (PQAP) is described in Chapter 17.1 of the PSAR. The Operational QA Plan will be described in Chapter 17.2 of the FSAR. The PQAP specifies requirements applicable to prime contractors and HL&P. The HL&P quality assurance staff shall assure through implementation review that the HL&P staff and contractors are complying with the QA program and the PQAP. Implementation reviews are performed by qualified personnel based on experience, educational level, training, and proficiency 57
examinations. Certifications are issued for specific discipline oriented activities. The implementation reviews use prepared checklists and techniques such as interviews with personnel performing the activities, observations of actual work in progress, and reviews of final form.

The combination of the QA programs as described in the NQAPM, PQAP, and OQAP as augmented by definitive procedures provide HL&P with the assurance that its quality commitments are met.

17.0.B EBASCO

The Quality Assurance program for safety related activities and services performed by Ebasco in the design, engineering, procurement, and construction of the Allens Creek Nuclear Generating Station is now described in the 33(U)
Ebasco Nuclear Quality Assurance Program Manual for the Allens Creek Project. This manual is a modified version of Ebasco's Topical Report No. ETR-1001, Revision 9, which was accepted by the NRC on January 5, 1981. 59
The ACNGS specific modifications to ETR-1001 are described in Table 17.0.B-1.

Later NRC approved revisions to ETR-1001 may be incorporated when deemed necessary. If necessary to define any additional clarifications, or modi- 45(U)
fications to the project Nuclear Quality Assurance Program Manual because of HL&P contract requirements or to suit the unique Project conditions, 33 (U)
they will be submitted for NRC acceptance in accordance with established 46 (U)
provisions which require execution of an authorization form involving approval of specified authorities to assure, among other things, that safety and/or quality are not sacrificed or compromised. 59

The Ebasco Quality Program defined herein assures that structures, systems, and components important to safety as defined in Section 3.2 of this PSAR, are reliable and possess a high degree of quality. This objective is achieved by the implementation of the Ebasco Nuclear Quality Assurance Manual which defines the policy, procedures, and requirements by which Ebasco will design, procure and erect the Allens Creek Nuclear Generating 59
Station. Implementation of the Ebasco Nuclear Quality Assurance Manual provides a quality program which is in compliance with the requirements of the Code of Federal Regulations, 10CFR50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants," and ANSI N45-2-1971, "Quality Assurance Program Requirements for Nuclear Power Plants".

The Ebasco QA organization and division of responsibility for ACNGS is summarized on Figure 17.0.B-1. 59

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17.0.C GENERAL ELECTRIC

The Quality Assurance Program for safety related activities and services for the Allens Creek Nuclear Generating Station is described in the General Electric Nuclear Energy Divisions BWR Quality Assurance Program Description, NEDO-11209-04A. 33(U) 49

In addition, General Electric has been asked to review Regulatory Guides 1.58 Rev. 1 and 1.146 Rev. 0, specifically for the ACNGS Project. These positions are shown in Table 17.0.C-1. 59

17.1 QUALITY ASSURANCE DURING DESIGN AND CONSTRUCTION

17.1.1A ORGANIZATION

The major organizations involved in the ACNGS are: 33(U)

a) Houston Lighting & Power Company (HL&P)

As the Applicant, HL&P has and retains the overall responsibility for the engineering, design, procurement, fabrication, construction, pre-operational testing, operation and QA activities for the ACNGS.

HL&P will audit the activities of Ebasco, GE, consultants and other contractors to assure that their QA Programs are implemented and have sufficient authority and organizational freedom to be effectively implemented.

HL&P will perform surveillance of the activities of Ebasco, GE, consultants and other contractors during the manufacturing, fabrication and construction of the ACNGS.

b) Ebasco Services Incorporated (Ebasco)

As the Architect-Engineer, Ebasco is delegated the responsibility to provide HL&P with engineering, design, procurements and QA services. As the constructor, Ebasco, is delegated the responsibility to provide HL&P with construction and QA services at the site.

Ebasco has the responsibility to provide an acceptable QA program to HL&P for the activities that have been delegated to Ebasco. These delegated activities include the following: 57

1) design and engineering
2) procurement activities
3) home office QA activities
4) vendor surveillance activities
5) construction activities
6) site QA/QC activities

c) General Electric Company (GE)

As the Nuclear Steam Supply System (NSSS) and Nuclear Fuel Supplier, GE is delegated the responsibility to provide HL&P with the engineering, design, procurement, fabrication and QA services for the NSSS and Nuclear Fuel. GE has the responsibility to provide an acceptable QA program to HL&P for the activities that have been delegated to GE.

These delegated activities include the following:

1) design and engineering activities
2) procurement activities
3) fabrication activities
4) vendor surveillance activities
5) QA activities

d) Consultants

HL&P may utilize the services of qualified consultants or other contractors to assist in the performance of appropriate quality tasks, such as audits, inspections, interpretations of test results, reviews, etc. 57

Figure 17.1.1A-1 illustrates how the above companies interrelate for the ACNGS.

Figure 17.1.1A-2 is an organization chart showing the organizations within HL&P with responsibility for the engineering, design, procurement, construction, operation, and quality assurance activities for ACNGS.

Figure 17.1.1A-3 is an organization chart of the HL&P Quality Assurance group for the ACNGS.

The HL&P Project Quality Assurance Manager shall be located at the construction site and is responsible for directing and managing the site QA program. He is free from non-QA duties. The HL&P Project Quality Assurance Manager is responsible for providing the programmatic direction and administering the policies, goals, objectives, and methods for the Allens Creek Project which are described in the Project 59
Quality Assurance Plan. Programmatic direction is defined as the role of HL&P in establishing the program requirements and ensuring the adequacy of the quality assurance program for HL&P and the prime contractors. The Project Quality Assurance Manager reports to the Manager, Quality Assurance, who reports directly to the Executive Vice President and has the independent authority to identify quality-related problems, to initiate or recommend solutions, to control existing nonconformances, to verify implementation of approved dispositions, and when necessary to stop work.

The HL&P Executive Vice President reviews and approves the Project Quality Assurance Plan and has ultimate responsibility for Quality Assurance activities. The Project Quality Assurance Plan interfaces with the corporate Quality Assurance program objectives by describing specific Quality Assurance controls to be established by HL&P and the prime contractors on the Allens Creek Project.

Two levels of control have been implemented by HL&P to monitor the effectiveness of the Quality Assurance Programs for the Allens Creek Project: (1) Corporate level control relates to the overall activities 57
and performance of HL&P, Ebasco, subcontractors and suppliers. This is administered through the direct involvement of the HL&P Executive Vice President and through audits of project activities. (2) Project level control relates to monitoring the specific activities and performance of HL&P, Ebasco and its subcontractors. This is accomplished through review of documents, and implementation reviews that establish QA system features (e.g. procedures, specifications).

HL&P Executive Management is also involved through participation in the Quality Assurance Program Evaluation Committee (QAPEC). The QAPEC is comprised of HL&P management (and consultants as required) whose purpose is to provide a review and evaluation of the HL&P nuclear QA program. The primary responsibilities of the QAPEC is to assess the effectiveness of the HL&P nuclear QA Program from the management viewpoint including: 59

1) review of NRC reports
2) review of trend analysis reports
3) review of selected audit reports
4) review of management QA audits
5) review major changes to methods and systems being implemented as part of the nuclear QA Program

Designated QA individuals are involved in day-to-day plant activities important to safety, (i.e. the QA organization routinely attends and participates in daily plant work schedule and status meeting to assure they are kept abreast of day-to-day work assignments throughout the plant and that 57
there is adequate QA coverage relative to procedural and inspection controls, acceptance criteria, and QA staffing and qualification of personnel to carry out QA assignments).

Figures 17.1.1A-3 and 17.0.B-1 show the project QA organization and indicate 59
which personnel are "onsite" and "offsite". The PSAR Section 13.0 shows project personnel from other organizations. The criteria for determining staffing for the QA organization includes: 57

a) Establishing the number of QA/QC personnel based upon the project schedule to ensure that personnel are available, qualified, and certified to perform quality-related inspections and evaluations.

b) Establishing the need for specially qualified QA/QC personnel based upon the schedule for activities requiring special or unusual expertise as far in advance of the activity as possible.

c) Establishing the number of QA personnel based upon the number and criticality of problems identified during routine activities in order to perform additional or supplemental inspections, reviews, or evaluations as required to ensure implementation of project requirements. 57

Staffing projections are periodically reviewed based upon the project schedule and are re-reviewed and revised, as necessary, as the project schedule changes. QA management personnel participate in short and long range scheduling activities. Staffing levels for QA/QC are a prime consideration in determining the level of effort for quality-related activities. Prior to allowing quality-related activities to be conducted, adequate numbers of qualified QA/QC personnel must be available. Adequate QA/QC staffing must be available to prevent QA/QC personnel from being required to perform inspections without adequate preparation time or under pressure to complete inspections within a scheduled time period. Adequate QA/QC staff must be available to allow for prompt closeout of open nonconformances and proper follow-up to ensure corrective action has been taken.

17.1.1A.1 Manager, Quality Assurance

The Manager, Quality Assurance, has the authority and responsibility to identify, initiate, recommend, or provide solutions to quality related problems and verify the implementation and effectiveness of the solutions. This position has the authority to "stop work" for cause in engineering, design, procurement, fabrication, construction, and operation phases of the nuclear plant. The minimum requirements established for this position are:

a) a college degree in a field of engineering or science or equivalent experience.

b) familiarity with nuclear power generation facilities and the related operations.

c) knowledge of the industry's quality assurance standards and regulatory requirements.

d) management experience and familiarity with HL&P corporate organizations.

The Manager, Quality Assurance, provides technical guidance, project direction, and administrative direction to: 57

a) Project QA Manager, Allens Creek

b) Houston QA Manager

c) Operations QA Manager

The Manager, Quality Assurance, reports to the Executive Vice President.

17.1.1A.2 Project Quality Assurance Manager, Allens Creek

The Project Quality Assurance Manager, Allens Creek (Project QA Manager) must as a minimum have:

a) a college degree in a field of engineering or science, or equivalent experience.

b) familiarity with nuclear power generation facilities and related operations.

c) knowledge of the Quality Assurance standards and regulatory requirements.

d) management experience and familiarity with HL&P corporate organizations.

The major responsibilities of the Project QA Manager are:

a) administer QA policies established by management and ensure the proper planning, development, implementation, coordination and administration of the Project Quality Assurance Plan.

b) provide programmatic direction on QA related matters to HL&P and contractor management and interface with NRC. 57

c) coordinate activities relating to auditing and vendor surveillance in conjunction with the HL&P Houston Quality Assurance Manager.

The Project QA Manager has the authority to solve quality-related problems and to verify the implementation and effectiveness of the solutions. He has the authority to "Stop Work" for cause on any quality-related activity of the Allens Creek Project. 59

17.1.1A.3 Houston Quality Assurance Manager

The Houston Quality Assurance Manager reports on all technical and administrative matters directly to the Manager, Quality Assurance. This organizational arrangement provides independence from cost and scheduling influences.

The Houston Quality Assurance Manager is responsible for directing all HL&P Houston office auditing, vendor surveillance and technical support activities. He has the authority to "Stop Work" for cause on any quality-related activity of the Allens Creek Project.

The Houston Quality Assurance Manager as a minimum has:

a) a college degree in a field of engineering or science, or equivalent experience. 57

b) Familiarity with nuclear power generation facilities and the related operations.

c) knowledge of the industry's Quality Assurance standards and regulatory requirements.

d) management experience and familiarity with HL&P corporate organizations.

The major responsibilities of the Houston Quality Assurance Manager are:

a) provide administrative guidance and direction for the HL&P Quality Assurance audit program.

b) direct the HL&P vendor surveillance program.

c) provide technical support in the review of specifications, procedures, manuals, procurement documents, etc.

17.1.1A.4 Project Quality Assurance General Supervisor

The Project Quality Assurance General Supervisor reports directly to the Project QA Manager. He is responsible for technical direction and administrative guidance to the discipline Quality Assurance personnel, providing programmatic direction to Ebasco and interfacing with the NRC. He has the authority to "Stop Work" for cause on any activity related to fabrication and construction.

17.1.1A.5 Supervisor, Quality Systems

The Supervisor, Quality Systems reports directly to the Project QA Manager.
He is responsible for providing technical direction and administrative guid-
ance to the site Quality Systems personnel; developing and administering the
HL&P Project QA Plan; evaluating the Ebasco QA/QC program; administering the 57
HL&P site QA personnel training and certification program; administrative
control of HL&P quality assurance procedures and providing mechanisms to
correct the QA programs as necessary. He has the authority to "Stop Work" for
cause on any activity related to fabrication or construction.

17.1.1A.6 Discipline Project Quality Assurance Supervisors

The Discipline Project Quality Assurance Supervisors report to the Project Quality Assurance General Supervisor. They are responsible for technical direction and administrative guidance to the Discipline Quality Assurance personnel in their respective discipline group; coordinating implementation reviews; interface with NRC during audits; identifying deficiencies; reviewing and approving procedures applicable to their respective discipline; and providing programmatic direction to Ebasco. They have authority to "Stop Work" for cause on any activity related to fabrication or construction.

17.1.1A.7 Procurement Project Quality Assurance Supervisor

The Procurement Project Quality Assurance Supervisor reports directly to the Project QA Manager. He is responsible for providing technical direction and administrative guidance to procurement Quality Assurance personnel, coordinating the resolutions of vendor problems identified by HL&P, coordinating with site discipline Quality Assurance functions for input to vendor surveillance/audit activities and providing programmatic direction to Ebasco regarding vendor surveillance and auditing functions. He has the authority to "Stop Work" for cause on any activity related to engineering, design, or procurement.

17.1.1A.8 Manager, Allens Creek Project

The Manager, Allens Creek Project reports to the HL&P Vice President, Nuclear Engineering and Construction. He has overall responsibility for the engineering, construction, procurement, cost, schedule, and start-up of the Allens Creek Project.

He directs the personnel assigned to the Allens Creek Project in the performance of their activities to ensure that design and engineering, procurement construction, and start-up meets the requirements of the project specifications, procedures and policies. Ensures that interfaces and communication with and support by HL&P and Ebasco parent organizations are adequate to assure competent performance of project-related activities. He has authority to "Stop Work" for cause in all activities of the project.

17.1.1A.9 Project Engineering Manager

The Project Engineering Manager reports to the Manager, Allens Creek Project. He directs project engineering personnel in the performance of the HL&P review of the design and engineering work performed by the prime contractors. The Project Engineering Manager ensures that adequate engineering planning and coordination of solutions to problems and work priorities are established by the prime contractors. He can recommend "Stop Work" for cause in the engineering and design of all items. 57

17.1.1A.10 Supervising Project Engineer(s)

The Supervising Project Engineer(s) report to the Project Engineering Manager. They direct in their area of responsibility the daily activities and interface with prime contractors. These activities include adequate engineering planning, coordination of problems, work priorities and activities of the HL&P Project Engineering group assigned to each Supervising Project Engineer. The Supervising Project Engineer(s) monitor the Prime Contractors resolution of pertinent QA noncompliances, participate in the HL&P Incident Review Committee and identify and resolve critical problems in their area of responsibility. Direct the coordination and interface between design engineering and other project disciplines, ensure the HL&P review of ACP design documents and recommend "Stop Work" for cause in the engineering and design of all items within their area of responsibility.

17.1.1A.11 Project Construction Manager

The Project Construction Manager reports to the Manager, Allens Creek Project. He is responsible for monitoring the total construction effort and maintaining liaison between HL&P and the Prime Contractors Management. The Project Construction Manager provides technical direction and administrative guidelines for HL&P and Prime Contractors in the areas of construction, security, start-up, accounting, construction control, and ensures that the prime contractor's management properly implements the dispositions to various nonconformances as determined by the engineering resolution. Reviews and approves as applicable procurement documents, drawings, specifications and construction interface schedules with subcontractors and ensures that construction conforms to the plans, specifications and procedures that govern work activities. Has the authority to "Stop Work" for causing relating to construction.

17.1.1A.12 Project Purchasing Manager

The Project Purchasing Manager reports to the Manager, Allens Creek Project. He is responsible for the overall coordination and administration of purchasing and subcontracting activities for the Allens Creek Project including the development and implementation of procedures, vendor selection, contract negotiations and preparing purchase orders.

17.1.1A.13 Project Controls Manager

The Project Controls Manager reports to the Manager, Allens Creek Project. He is responsible for providing a detailed project budget and schedule integrating engineering, construction and start-up. The Project Controls Manager has no directly-related quality assurance responsibilities on the project.

17.1.1A.14 Project Administration Supervisor

The Project Administration Supervisor reports to the Manager, Allens Creek Project. He is responsible for coordination of support to the Allens Creek Project Team from Ebasco and HL&P, processing and distribution of project mail, development of project procedures and administrative support. 57

17.1.1A.15 Project Controller

The Project Controller reports to the Manager, Allens Creek Project. He is responsible for the coordination and execution of the accounting and financial administration. The Project Controller has no direct quality assurance responsibilities on the project.

17.1.1A.16 Project Environmental Engineer

The Project Environmental Engineer reports to the Manager, Allens Creek Project. He is responsible for the environmental protection of the environs of the plant and for the acquisition of all local, state, and federal permits and approvals exclusive of NRC licensing.

17.1.1A.17 Project Nuclear Fuel

The Project Nuclear Fuel group reports to the Director, Nuclear Fuels. They are responsible for fuel procurement, fuel management and providing technical support on nuclear fuel related matters.

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17.1.2A QUALITY ASSURANCE PROGRAM

The HL&P Project Quality Assurance program for the Project has been developed in accordance with the criteria of 10CFR50 Appendix B, ANSI N45.2 and Regulatory Guides as referenced herein, to provide programmatic direction on quality requirements for the prime contractors and subcontractors during design and construction. 57

The nuclear safety-related structures, systems and components covered by this program are listed in Section 3.2, Table 3.2-1, column designated "Quality Assurance Program". In Table 3.2-1, GE has the responsibility for Quality Assurance (QA) for items designated "GE" in the "Scope of Supply" column in Table 3.2-1 until delivery of the component to the site. Ebasco QA retains the responsibility for QA of all items designated "P" and the GE items upon receipt at the project site.

Items listed in Table 3.2-1, the Q-list, will be maintained in compliance with 10CFR50 Appendix A and R.G. 1.29, Revision 1. HL&P shall approve additions or deletions to Table 3.2-1. The criteria for and management of the the items on 59 the Q-List are described in Chapter 3. Measures shall be established for the initiation, control and maintenance of the Q-list by Ebasco. These measures shall include provisions for signature approval by Engineering and QA, controlled distribution of the lists to identify responsible personnel, and assurances to preclude use of obsolete lists.

The HL&P Quality Assurance program for the Allens Creek Project is described 57 by the HL&P Project Quality Assurance Plan (PQAP). A letter signed by the Executive Vice President in the front of the PQAP makes the requirements of the PQAP mandatory. Procedures are reviewed by project QA personnel during 59 preparation for inspections, surveillance, implementation reviews, and audits to ensure consistency with project requirements. Additionally, selected procedures are reviewed and concurred with by the project QA organization prior to issuance. The plan requires that written procedures, training and certification, issuance of specifications and drawings, and work and inspection planning be accomplished in advance of performing nuclear safety-related activities. HL&P Project Quality Assurance ensures through procedure reviews that this advance preparation is accomplished.

The Project Quality Assurance Plan for the Allens Creek Project is structured in accordance with the NRC regulatory position of the Regulatory Guides as 57 described in Appendix C of the PSAR and with ANSI N45.2.12. (Draft 3, Rev. 4 - February, 1974).

The HL&P QA Program and Procedures which are used to implement the quality related activities for each major organization and the reference to the applicable criteria of 10CFR50 Appendix B are listed in Table 17.1.2A-1. Verification that plans and procedures are properly implemented is accomplished by HL&P Quality Assurance through audits, inplementation reviews and regular management assessment of the Quality Assurance Program.

The project QA organization and the necessary technical organization participate clearly in the QA program definition stage to determine and identify the extent QA controls are to be applied to specific structures, systems, and components.

Implementation reviews shall be performed by HL&P Discipline Quality Assurance 59
personnel using prepared checklists to evaluate the effectiveness of compliance to the Quality Assurance Program at the Allens Creek Nuclear Generating Station site during construction. The implementation reviews use techniques such as interviews with personnel performing the activities, observations of actual work in progress, and reviews of final form.

Implementation reviews are performed at the construction site by personnel qualified based upon experience, education level, training, and proficiency 57
examinations. Certifications are issued for specific discipline oriented activities. This qualification and certification program is documented in written procedures. Personnel performing quality control functions at the site and at vendor facilities are qualified in accordance with ANSI-N45.2.6 (Regulatory Guide 1.58, Revision 1). Audit personnel will be qualified in 59
accordance with Regulatory Guide 1.146.

It is the policy of HL&P as applicant, to assure that the design, engineering, 57
procurement, fabrication, construction, preoperational testing, and opera- 59
tion of ACNGS are in conformance with project specifications, procedures, codes, and NRC regulations. It is the responsibility of each organization assigned to the Allens Creek Project to ensure that project procedural review methods include provisions to ensure that the requirements stated in this manual are incorporated into project procedures. The Project Quality Assurance Plan establishes activities and procedures which identify, initiate and verify the resolution of nuclear safety-related quality problems. The imple- 57
menting procedures call for the resolution of quality problems at the lowest possible authorized level. However, if a dispute is encountered in the resolution of a quality problem which cannot be resolved at lower levels, the HL&P Project QA Manager presents the problem ultimately to the HL&P Executive Vice President for resolution.

Allens Creek Project Quality Assurance is responsible for conducting a quality oriented indoctrination program for new personnel that have quality-related functions. The HL&P Project Quality Assurance Plan requires that prior to performing activities affecting quality the personnel are trained in the applicable procedures. The training, qualification and certification programs are established such that: 59

a) Personnel responsible for performing quality affecting activities are instructed as to the purpose, scope, and implementation of the quality related manuals, instructions, and procedures.

b) Personnel verifying activities affecting quality are trained and qualified in the principles, techniques, and requirements of the activity being performed.

c) For formal training and qualification programs, documentation includes the objective, content of the program, attendees, and date of attendance.

d) Proficiency tests are given to those personnel performing and verifying activities affecting quality, and acceptance criteria are developed to determine if individuals are properly trained and qualified. 59

e) Certificate of qualifications clearly delineates (1) the specific functions personnel are qualified to perform, and (2) the criteria used to qualify personnel in each function.

f) Proficiency of personnel performing and verifying activities affecting quality is maintained by retraining, reexamining, and/or recertifying as determined by management or program commitment.

g) The description of the training program provisions listed above satisfies the regulatory position in Regulatory Guide 1.58, Revision 1.

HL&P Quality Assurance audits are performed to ensure compliance with these criteria.

The Project QA Manager and the Houston Quality Assurance Manager are directly responsible for assuring effective implementation of the Quality Assurance program. The qualifications for these positions are defined in Sections 17.1.1A.2 and 17.1.1A.3.

The HL&P Project Quality Assurance Plan requires the prime contractor (Ebasco) to submit all procedures which control nuclear safety-related construction 57
activities to HL&P Project Quality Assurance for review. Procedures are reviewed by Project QA personnel during preparation for inspection, surveillance, implementation reviews and audits to ensure consistency with project requirements. Additional selected procedures are reviewed and concurred with prior to issuance. It is the responsibility of HL&P Project Quality Assurance to determine that the prime contractor's procedures require proper equipment, environment and other prerequisites to perform the associated activity. These requirements are verified through implementation reviews by HL&P Discipline QA and audits by HL&P Houston QA.

The results of the HL&P implementation reviews and audits are presented in a monthly report to the HL&P Executive Vice President. Regular executive management review of the monthly activities and the direct involvement of the HL&P Executive Vice President assures that an objective program assessment of the Allens Creek Project Quality Assurance program is being performed. 59

HL&P Project Quality Assurance reviews and documents concurrence with the Ebasco Quality Assurance manual and audits are performed by HL&P Houston Quality Assurance to ensure compliance.

When required to implement/originate quality activities (10CFR50, Appendix B), HL&P will apply the same controls, as is specified in this Chapter 17, for
Principal Contractors. Additional QA/QC requirements will be described for 59
the operational phase, which includes preoperational and startup in the FSAR.

HL&P is committed to maintaining the Project Quality Assurance Plan as an effective and meaningful document to provide directions to HL&P and the prime contractors on the Allens Creek Project. When proposed substantive changes to this Project Quality Assurance Plan affect the docketed Quality Assurance
Program description, HL&P will notify the NRC of the change(s) for their 57
review and acceptance prior to implementation. Organizational changes of a substantive nature will be reported to the NRC within 30 days of announcement.

Table 17.1.2A-1 is a matrix showing 10CFR50, Appendix B criteria compared to appropriate sections of the QA Program and Plan. This matrix illustrates how the HL&P QA Program and Allens Creek QA Plan are in compliance with the Regulatory criteria. HL&P positions on Regulatory Guides (RGs) are enumerated in Appendix C.

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17.1.3A DESIGN CONTROL

HL&P has the overall responsibility for design and engineering of the Allens Creek Project and imposes the requirements of 10CFR50, Appendix B, Criterion III, Regulatory Guide 1.64 (Rev. 2) and ANSI N45.2.11-74 on the prime contractors and applicable subcontractors.

HL&P has contracted with Ebasco and General Electric to perform the design, engineering, and design verification. HL&P Engineering performs reviews of selected elements of the completed design, design documents, and specifications to ensure that contractual requirements are met.

The HL&P Project Engineering Manager is responsible for ensuring that project engineering activities are conducted in accordance with approved engineering procedures. The project engineering organization provides programmatic direction and overview of the Ebasco engineering activities. The HL&P project engineering activities are conducted in accordance with approved project procedures.

When HL&P has direct responsibilities or assumes direct responsibility for conducting design activities, these activities will be conducted in accordance with the requirements of this section and/or the FSAR Section 17.2.3.

HL&P contractors are required to provide the following design control measures in their quality assurance programs:

a) A design control system is established to document the methods of accomplishing and controlling essential design activities. 57

b) Design documents such as calculations, diagrams, specifications, and drawings are prepared and records developed such that the final design is traceable to its sources.

c) Design activities, documents, and interfaces are controlled to assure that applicable input such as design bases, regulatory requirements, codes, and standards are incorporated into the final design.

d) Design input requirements, including design criteria, are documented and their selection reviewed and approved.

e) Design documents include an indication as to their importance to safety and shall specify the quality characteristics, including materials, parts, equipment and processes, that are essential to functions of structures, systems, and components. Design documents also include, as appropriate, acceptance criteria for inspections and tests.

f) Design control measures are applied to items such as seismic, stress, thermal, hydraulic, radiation, and accident analyses, as they apply to the development of design input or as they are used to analyze the design.

g) Safety-related and/or seismic Category I designs are verified for adequacy and accuracy through independent objective review of design documents by individuals competent in the subject activity. This verification may include the use of alternate or simplified solution methods or qualification testing, as appropriate.

h) Design changes, including engineering, vendor, and construction originated changes, are controlled in a manner commensurate with the control imposed on the original design.

i) Document distribution is controlled such that all individuals using a design document or its results and/or conclusions for further design work can be notified if the document is revised or cancelled.

j) Design documentation includes evidence that design control requirements have been satisfied.

k) Errors and deficiencies in approved design documents, including design methods (such as computer codes), that could adversely affect structures, systems, and components important to safety are documented; and action taken to assure that all errors and deficiencies are corrected.

l) Deviations from specified quality standards are identified and procedures are established to ensure their control.

m) A documented check to ensure dimensional accuracy (including tolerance for accept/reject criteria and inspectability) and the completeness of the drawings and specifications. 57

n) A system to ensure design requirements from engineering specifications and drawings for that system, component, or structure are included in inspection documents and that the cognizant engineering group perform an engineering evaluation and signoff on deviations identified on the inspection documents.

o) A system is established to require that design specifications and drawings are reviewed by individuals knowledgeable and qualified in QA/QC techniques to assure that the documents are prepared, reviewed, and approved in accordance with written procedures and that the documents contain the necessary QA requirements such as inspection and test requirements, acceptance requirements, and documenting of inspection and test results.

HL&P Houston Quality Assurance performs audits of HL&P, Ebasco, and General Electric to ensure that design controls, requirements, specifications, and documents are in accordance with the design control criteria.

In addition HL&P Project Quality Assurance reviews quality/construction procedures to ensure that the quality requirements of the design specifications are incorporated. HL&P Project Quality Assurance also performs implementation reviews to ensure that the work is accomplished in accordance with the design requirements and to ensure that field changes to the design are processed in accordance with the design control criteria.

17.1.4A PROCUREMENT DOCUMENT CONTROL

To assure that nuclear safety-related items are purchased in a planned and controlled manner, the HL&P Project Quality Assurance Plan establishes basic requirements which are to be used by HL&P in preparing procurement procedures for the Allens Creek Project. Ebasco performs procurement activities for nuclear safety-related equipment, materials, and services, 57 exclusive of the NSSS contract, which is performed by General Electric. Ebasco and General Electric ensure through contract, vendor surveillance, and audit that their suppliers comply with the established requirements.

The basic requirements are:

a) Written procedures are established clearly delineating the sequence of actions to be accomplished in the preparation, review, approval, and control of procurement documents. 1 Q17.9

b) A review of the adequacy of quality requirements stated in procurement documents is performed by qualified personnel knowledgeable in the QA requirements. This review is to determine that all quality requirements are correctly stated; they can be inspected and controlled; there are adequate acceptance and rejection criteria; and the procurement document has been prepared in accordance with QA Program requirements. 33(U)

c) Documented evidence of the review and approval of procurement documents is provided and available for verification. 33(U)

d) Procurement documents identify those QA requirements which must be complied with and described in the supplier's QA Program to meet 10CFR Part 50, Appendix B. This QA Program or portions thereof shall be reviewed for adequacy by qualified personnel knowledgeable in QA.

e) Procurement documents contain or reference applicable design bases technical requirements including regulatory requirements, component and material identification, drawings, specifications, codes and industrial standards, including their revision status, tests and inspection requirements and special process instructions for such activities as fabrication, cleaning, erecting, packaging, handling, shipping, storing, and inspecting. 33(U)

f) Procurement documents contain as applicable, requirements which identify the documentation to be prepared, maintained, submitted, and made available to the procuring agent for review and/or approval, such as drawings, specifications, procedures, inspection and test records, personnel and procedure qualifications, and material and test reports. 33(U)

g) Procurement documents contain the requirements for the retention, control, and maintenance of records.

h) Procurement documents contain the procuring agency's right of access to vendor's facilities and records for source inspection and audit.

i) Changes and/or revisions to procurement documents are subject to at least the same review and approval requirements as the original document.

j) Purchase documents for spare or replacement parts of safety-related structures, systems, and components are reviewed for adequacy of quality requirements by qualified personnel knowledgeable in QA. The review is to determine the adequacy relative to the quality assurance requirements and acceptance criteria of the original design. 1 017.9

k) The evaluation and selection of suppliers are determined by qualified personnel in accordance with written procedures.

l) A written procedure acceptable to HL&P shall be used for source evaluation.

m) Procurement documents, records, and changes thereto are collected, stored, and maintained in a systematic and controlled manner.

HL&P Engineering is responsible for review and approval of Ebasco procurement specifications. Engineering also coordinates with HL&P Procurement QA for performance of a quality assurance review. HL&P Procurement QA coordinates with Ebasco and HL&P Engineering in the review of the procurement package. 57

In addition, HL&P Discipline QA is responsible for reviewing field procurement packages to ensure that all quality assurance requirements have been included.

HL&P Houston Quality Assurance is responsible for performing audits and vendor surveillance to verify that the requirements have been implemented and that they are effective.

17.1.5A INSTRUCTIONS, PROCEDURES, AND DRAWINGS 59

The HL&P Project Quality Assurance Plan requires HL&P, the prime contractors, and their suppliers to establish and implement a Quality Assurance Program which is in compliance with 10CFR50, Appendix B. The program is effective in verifying that the defined activities are accomplished and documented in accordance with written procedures, instructions, and drawings and that they provide quantitative and qualitative acceptance criteria. 57

Procedures for the review, approval, and issuance of documents (including procedures, instructions, specifications, and construction drawings), and changes thereto are established and described to assure technical adequacy and inclusion of appropriate quality requirements prior to implementation. Selected documents are reviewed and concurred with by the project QA organization for Quality Assurance related aspects.

HL&P Project Quality Assurance reviews the Ebasco Allens Creek Project
Quality Assurance Program. To measure the effectiveness of the Quality
Assurance Program, HL&P has implemented a monitoring program consisting of
audits which are performed by HL&P Houston Quality Assurance and imple-
mentation review and trend analysis performed by the HL&P Project Quality
Assurance Department. HL&P Houston Quality Assurance also audits HL&P 57
organizations and General Electric for compliance with their respective
Quality Assurance programs.

Table 17.1.5A-1 is a matrix showing HL&P procedures that are used on the Allens Creek Project compared to the appropriate 10CFR50, Appendix B criteria. This matrix illustrates how the requirements of the applicable 10CFR50, Appendix B criteria are addressed in the written procedures used on the Allens Creek Project.

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17.1.6A DOCUMENT CONTROL

The HL&P Project Quality Assurance Plan and implementing procedures require that HL&P, the prime contractors, and subcontractors implement a document control system for nuclear safety-related items for the Allens Creek Project. The established system ensures that design, engineering, procurement, fabrication, construction, and QA/QC procedures, plans, and changes 57
thereto are reviewed and approved by procedurally authorized groups and that the documents are issued, maintained current, and controlled by the use of controlled lists of document holders to ensure that superseded documents are replaced in a timely manner.

The document control system applies to the control of quality-related documents, including as a minimum:

a) Design documents; such as calculations, drawings, specifications and analysis

b) Procurement documents

c) Instructions and procedures for such activities as fabrication, construction, installation, test inspection, modification, operation, maintenance and refueling 59

d) As-built drawings

e) Safety Analysis Reports

f) QA/QC manuals

g) Nonconformance reports

h) Audit reports

Project procedures will be developed to ensure that drawings are provided to indicate the as-built configuration. The as-built drawings will stand alone and delineate actual location - elevation, azimuth, etc.; actual component identification or numbering; and dimensions and other relevant information. When changes occur subsequent to issuance of as-built drawings, procedures will require a re-review and reissue of the drawings.

Measures are established and documented to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe activities affecting quality. These measures shall assure that documents, including changes, are reviewed for technical adequacy and the inclusion of appropriate quality requirements and approved for release by authorized personnel and are distributed to and used at the 57
location where the prescribed activity is performed. Changes to documents are reviewed and approved by the same organizations that performed the original review and approval unless other organizations are specifically designated. The reviewing organization has access to pertinent background information upon which to base its approval and shall have adequate understanding of the requirements and intent of the original document.

Those participating in an activity are made aware of and use proper and current instructions, procedures, drawings, and engineering requirements for performing the activity. Participating organizations have procedures for control of the documents and changes thereto to preclude the possible use of outdated or inappropriate documents.

Document control measures provide for:

a) identification of individuals or organizations responsible for preparing, reviewing, approving, and issuing documents and revisions thereto;

b) identifying the proper documents to be used in performing the activity; 57

c) coordination and control of interface documents;

d) ascertaining that proper documents are being used;

e) establishing current and updated distribution lists

The document control system includes a listing identifying the current revision of instructions, procedures, specifications, drawings, and procurement documents. This list is updated and distributed to cognizant responsible personnel.

HL&P Discipline Quality Assurance performs implementation reviews at the construction site to ensure that document control systems are in place and effectively implemented. HL&P Quality Assurance audits are performed to ensure compliance with these criteria.

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17.1.7A CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

The HL&P Quality Assurance Plan and implementing procedures require that HL&P, prime contractors, and subcontractors define and document the system and requirements for the control of nuclear safety-related purchased material, equipment, and services.

Control and verification of supplier's activities during fabrication, inspection, testing, and shipment of materials, equipment, and components are planned and performed as early as possible, as required, to assure conformance to the purchase order or contractual requirements. These procedures provide for:

a) Requiring the supplier to identify processes to be utilized in fulfilling procurement requirements. 57

b) Reviewing documents required to be submitted by the procurement requirements.

c) Specifying the characteristics or processes to be witnessed, inspected, or verified and accepted based upon the fabrication schedules; the method of surveillance, and the extent of documentation required; and those responsible for implementing these procedures.

d) Audits, surveillance, and/or inspections which assure that the supplier complies with quality requirements and his QA program. 57

For items determined to be important to safety where specific QA controls cannot be imposed in a practical manner, an evaluation will be made to determine special quality verification requirements to be applied during installation or testing to provide the necessary assurance that the item(s) meet project requirements. 59

Control and verification of organizations performing service is accomplished by technical verification of data provided, surveillance and/or audit of the activity, and review of objective evidence such as certifications, reports, etc.

The selection of suppliers is based on evaluation of their capability to provide items or services in accordance with the requirements of the procurement documents prior to award of contract.

Procurement source evaluation and selection measures are implemented by HL&P and Ebasco and provide for identification of the organizational responsibilities for determining supplier capability.

Measures for evaluation and selection of procurement sources, and the results thereof, are documented and include one or more of a) through d) below: 57

a) Evaluation of the supplier's history of providing an identical or similar product or service which performs satisfactorily in actual use. The suppler's history shall reflect current capability.

b) Supplier's current quality records supported by documented qualitative and quantitative information which can be objectively evaluated.

c) Supplier's technical and quality capability as determined by a direct evaluation of his facilities and personnel and the implementation of his approved quality assurance program.

d) Evaluation of bid documents including review for technical adequacy, quality assurance, and commercial considerations.

Procurement of spare or replacement parts for structures, systems, and components important to safety is subject to QA program controls, to codes and standards, and to technical requirements at least equal to the original technical requirements or any properly reviewed and approved revisions there to.

A receipt inspection is planned and implemented to assure:

a) Timely inspection of items upon receipt.

b) The material, component, or equipment is properly identified and corresponds to the identification on the purchase document and receiving documentation.

c) Material, components, equipment, and acceptance records satisfy the inspection instructions prior to installation or use.

d) Specified inspection, test, and other records are accepted and available at the Allens Creek Project prior to installation or use where required.

e) Items accepted and released are identified as to their inspection status prior to forwarding them to a controlled storage area or releasing them for installation or further work.

f) Coordination of receipt inspection with vendor surveillance activities to ensure the required vendor inspection has been 57
performed and deficiencies have been resolved prior to shipment.

Supplier's certificates of conformance are evaluated by audits, vendor inspection, or tests to ensure that they are valid. Supplier's records will include a description of those nonconformances from the procurement requirements dispositioned "accept as is" or "repair".

Ebasco receiving inspection ensures that, for nuclear safety-related items received at the Allens Creek Project, there is accompanying documentation that indicates review and concurrence by the prime contractor or designee, that the item complies with established requirements or has an authorized waiver prior to shipment. HL&P Quality Assurance audits are performed to ensure compliance with these criteria.

HL&P Houston Quality Assurance ensures by an overview of the Ebasco vendor surveillance function that source surveillance and inspection are performed in accordance with the quality assurance program. In addition, HL&P Discipline QA performs implementation reviews of activities commencing with receiving inspection at the site to ensuring proper controls of purchased material and equipment are exercised.

HL&P Houston Quality Assurance performs audits of these activities to ensure overall compliance.

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17.1.8A IDENTIFICATION AND CONTROL OF MATERIALS, PARTS AND COMPONENTS

The HL&P Project Quality Assurance Plan requires that prime contractors and suppliers establish written procedures which identify, control, and ensure traceability of materials, parts, and components including partially assembled components. Prime contractors and suppliers procedures shall include the documented verification of correct identification of materials, components, and subassemblies, and that the identification does not affect the function or quality of the item prior to release of the items for assembly or installation. Specific procedures have been developed to:

a) Establish controls to identify and control materials (including consumables), parts, and components (including partially fabricated subassemblies).

b) Provide a method for identification of quality related materials and parts and to provide traceability to the appropriate drawings, 57
specifications, purchase orders, manufacturing, and inspection documents, deviation reports, and physical and chemical mill test reports.

c) Provide a method for identification and control of incorrect or defective items. This system will include verification and documentation prior to release for fabrication, assembling, shipping, and installation.

HL&P Project Quality Assurance ensures that the above criteria are incorporated into the Ebasco quality/construction procedures during the procedure review and then follows up with implementation reviews to ensure compliance.

In addition HL&P Houston Quality Assurance performs audits for evaluation of the conformance to identification and control criteria.

17.1.9A CONTROL OF SPECIAL PROCESSES 59

The HL&P Project Quality Assurance Plan requires that written procedures be established by prime contractors and subcontractors for the activities associated with all special processes. For special processes the qualification of personnel, procedures, and equipment relating to specific
codes, standards, specifications, and contractual requirements shall be 57
documented and maintained current.

Special processes are defined as the processes where direct inspection if impossible or disadvantageous and which must be carefully controlled and monitored to ensure the required results. Special processes for the Allens Creek Project include:

a) welding

b) heat treating

c) cadwelding

d) concrete placement 57

e) nondestructive testing

f) chemical cleaning

Organizational responsibilities are defined in the Allens Creek Project procedures for qualification of special processes, equipment, and personnel. These responsibilities include the provision to assure that special processes are performed by qualified personnel using procedures qualified and approved in accordance with applicable codes, standards, or other requirements.

Procedures are established for recording evidence of acceptable accomplishment of special processes using qualified procedures, equipment, and personnel. 59
The QA organization verifies the recorded evidence and documents the result.

Special processes are performed under controlled conditions by qualified personnel using procedures qualified and approved in accordance with applicable codes, standards, or other requirements. For special processes not covered by existing codes or standards the specific equipment, personnel qualification, and procedure qualification requirements are defined prior to application of the special process.

Records are maintained for the qualification of procedures, equipment, and personnel associated with special processes. Records are in sufficient detail 57
to clearly define the procedures, equipment, or personnel being qualified; criteria or requirements used for qualification; and the individual approving the qualification.

HL&P Discipline Quality Assurance ensures that the special process control criteria are met by the review of all Ebasco special process procedures and performance of implementation reviews to ensure compliance.

HL&P Houston Quality Assurance performs audits of special process activities to ensure compliance with all aspects of the Quality Assurance program.

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17.1.10A INSPECTION

The HL&P Project Quality Assurance Plan requires the prime contractors to establish and implement an inspection operation whose activities are independent from the group performing the activities being inspected. The training, qualifications, and certifications of inspectors includes criteria from appropriate codes, standards, and the prime contractors procedures and shall be documented and kept current. Inspection activities relating to construction, fabrication, installation, and testing are documented, kept current and identify all mandatory inspection hold and test points and the criteria to be witnessed by authorized inspectors. Operations and inspections (including rework, replaced items) are performed in predetermined, documented sequences, and deviations or deletions must be accomplished in accordance with approved and documented systems. Inspection procedures include all required inspection operations defined by the specifications, drawings, codes, and standards. These procedures provide for the following:

a) Identification of characteristics and activities to be inspected

b) A description of the method of inspection 57

c) Identification of the individuals or groups responsible for performing the inspection operation

d) Acceptance and rejection criteria

e) Identification of required procedures, drawings, and specifications and revisions

f) Recording inspector or data recorder and the results of the inspection operation

g) Specifying necessary measuring and test equipment including accuracy requirements and verification of calibration

h) Evaluation of inspection results

The project QA organization participates in the definition of the scope of the inspection program. Procedures provide criteria for determining the accuracy requirements of inspection equipment and criteria for determining when inspections are required or define how and when inspections are 59
performed.

Procedures are established to identify in pertinent documents, mandatory inspection hold points beyond which work may not proceed until inspected by a designated inspector.

Where direct inspections are impossible or disadvantageous, in-process monitoring is specified in the inspection procedures and both direct and in-process monitoring are used when control is inadequate without both. All required procedures, specifications, and drawings are made available 57
to the inspectors prior to performing inspection. If mandatory inspection hold points are required beyond which work cannot proceed without specific

consent of the designate representative, the specific hold points will be indicated in appropriate documents. Inspection results are documented, 57
evaluated, and their acceptability determined by a responsible individual or group.

Personnel performing quality control functions at the site and at vendor facilities are and will be qualified in accordance with ANSI-N45.2.6 59
(Regulatory Guide 1.58, Rev.1).

HL&P Discipline Quality Assurance ensures that inspection control criteria are complied with by review and approval of the inspection procedures and 57
by implementation reviews of inspection in each discipline activity.

HL&P Houston Quality Assurance performs audits of HL&P and Ebasco inspection activities to ensure compliance with these criteria.

17.1.11A TEST CONTROL 59

The HL&P Project Quality Assurance Plan requires that a test control program be developed and documented by the prime contractors and subcontractors which demonstrates that the facility performs in accordance with the Allens Creek Project requirements and specifications. The training, certification of personnel, calibration and certification of test equipment, 57
system or component status, environmental conditions, inspection hold points, and configuration of the items to be tested are included in the procedures. Test results are documented, evaluated, and the acceptance status determined by the authorized departments.

A test control program will be established to include proof tests prior to installation and preoperational tests. 59
Procedures provide criteria for determining acurracy requirements of test equipment and criteria or determining when a test is required and how and when testing activities are performed.

Test procedures or instructions provide for the following as required:

a) The inclusion of requirements and acceptance limits contained in applicable design and procurement documents.

b) Instructions for performing the test

c) Test prerequisites such as calibrated instrumentation, adequate test equipment, and instrumentation including their accuracy require- 57
ments, completeness of item to be tested, suitable, and controlled environmental conditions, and provisions for data collection and storage

d) Mandatory inspection hold points for witness by Owner, contractor, or inspector (as required)

e) Acceptance and rejection criteria

f) Methods for documenting or recording test data and results

g) Provisions for assuring that test prerequisites have been met

h) Evaluation of test results

HL&P Discipline Quality Assurance ensures inclusion of adequate test control criteria by review of the Ebasco quality/construction testing procedures. They also perform follow-up implementation reviews to verify that the controls are implemented and effective. 57

HL&P Houston Quality Assurance audits both HL&P and Ebasco activities to verify QA program compliance.

The test control activities are an example of a case in which HL&P Discipline Quality Assurance monitoring activities and the Operational Quality Assurance monitoring activities will interface and in some instances overlap. 59 HL&P Project Quality Assurance procedures will specifically define the responsibilities for this transition period. 57

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17.1.12A CONTROL OF MEASURING AND TEST EQUIPMENT

The HL&P Project Quality Assurance Plan requires the establishment, documentation, and implementation of a Measuring and Test Equipment Control System. The system is to include calibration techniques, specifications and accuracy, frequency, and maintenance of all measuring instruments and test equipment used in the measuring, inspection, and monitoring of nuclear safety-related items. Calibration and maintenance data shall be filed and kept current. Calibration standards are to be traceable to nationally recognized standards. If standards do not exist, the basis for calibration of the equipment is to be documented. If measuring or test equipment is found to be out of calibration, an investigation is required to be performed to determine the validity of the use of the instrument and whether measurements or tests are required to be reperformed.

Equipment is identified and traceable to the calibration test data and suitably marked to indicate calibration status. Markings include the last day calibrated and next calibration due date. 57

Measuring and test equipment is calibrated at specified intervals based on the required accuracy, purpose, degree of usage, stability characteristics, and other conditions affecting the measurement. Calibration of this equipment is against standards that have an accuracy of at least four times the required accuracy of the equipment being calibrated, or when this is not possible, have an accuracy that assures the equipment being calibrated will be within required tolerance and that the basis of acceptance is documented and authorized by responsible management.

Calibrating standards will, when possible, have greater accuracy than standards being calibrated. Calibrating standards with the same accuracy may be used if it can be shown to be adequate for the requirements and the basis of acceptance is documented and authorized by responsible management.

HL&P Discipline Quality Assurance reviews and documents concurrence with Ebasco calibration procedures to ensure these criteria are incorporated. In addition implementation reviews are performed to ensure compliance.

HL&P Houston Quality Assurance audits the measuring and test equipment controls to ensure compliance to the QA program in this area.

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17.1.13A HANDLING, STORAGE, AND SHIPPING

The HL&P Project Quality Assurance Plan requires that for nuclear safety-related items, written procedures be developed in accordance with design requirements, specifications, and standards to control the cleaning, handling, storage, packaging, shipping, and preservation to preclude damage and deterioration by environmental conditions. The activities are to be accomplished by appropriate trained and experienced personnel.

HL&P Discipline Quality Assurance reviews and documents concurrence with 57
construction procedures for receiving, handling, storage, and cleaning to ensure that the appropriate criteria of Regulatory Guide 1.38 and ANSI N45.2.2 are included. Periodic implementation reviews are conducted to ensure compliance to the procedures.

HL&P Houston Quality Assurance performs audits to ensure overall program compliance.

17.1.14A INSPECTIONS, TEST, AND OPERATING STATUS 59

The HL&P Project Quality Assurance Plan requires that the prime contractor and subcontractors indicate the current inspection, test, and operating status of nuclear safety-related items through the use of stamps, markings, tags, or other suitable means. During the startup and testing activities, HL&P is responsible for complying with this section for inspection status, test status, and operating status. Procedures include the requirements for:

a) Controlling the application and removal of inspection status indicators such as tags, markings, labels, and stamps. 57

b) Documenting the status of nonconforming, inoperative, or malfunctioning structures, systems, and components to prevent inadvertent use.

c) Defining and documenting the use, application, removal, and status of inspection tags, labels, and markings which identify the status of inspections or tests performed or attest to the acceptability of the structure, system, or component.

d) Controlling the altering of the sequence of required tests, inspections, and other operations important to safety.

e) Providing a system for the indication of the inspection, test and operating status of structures, systems and components throughout 59
fabrication, installation and test.

HL&P Discipline Quality Assurance personnel review and document concurrence with these procedures and conduct periodic verification to assure compliance.
Houston Quality Assurance audits both HL&P Project Quality Assurance and 57
Ebasco to verify compliance.

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17.1.15A NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

The HL&P Project Quality Assurance Plan requires that HL&P and the prime
contractors' Quality Assurance Program include a system which is documented by
written procedures for the identification, segregation, and disposition of 57
nonconforming materials, parts, and components. The procedures shall specify
the preparation and handling of nonconformance documents, segregation require-
ments, and which groups are responsible for review and disposition of the
items. The procedures also provide identification of authorized individuals
for independent review of non-conformances, including disposition and closeout. 59

Documentation identifies the nonconforming item; describes the nonconformance, the disposition of the nonconformance, and the inspection requirements; and includes signature approval of the disposition. Nonconformances are corrected and resolved prior to initiation of the preoperational test program on the item. Rework, repairs, and subsequent reinspection and tests are conducted in accordance with the original inspection and test requirements or accepted alternatives and shall be performed in accordance with controlled procedures and contain mechanisms for providing information to the identifying group as to the disposition of the nonconformance.

For NSSS items, HL&P coordinates nonconformance resolution through GE. 57

HL&P Project Quality Assurance reviews for concurrence the proposed disposition of selected Ebasco nonconformance reports and performs an evaluation of Ebasco nonconformance trend analyses.

Procedures are established by HL&P to report significant deficiencies during the design, construction, and operations phase to HL&P executive management and to the Nuclear Regulatory Commission in accordance with 10CFR50.55(e), 10CFR21, and 10CFR71, where applicable.

Compliance of these activities with Project Quality Assurance Plan requirements is ensured through the performance of audits and implementation reviews.

17.1.16A CORRECTIVE ACTION 59

The HL&P Project Quality Assurance Plan requires that a system be established
and documented by HL&P and the prime contractors which defines the responsibi-
lities, authorities, and methods used by specific groups involved in the
evaluation of nonconformances and trending to determine the need for correc-
tive action. The system includes measures to identify the cause of signifi-
cant conditions adverse to quality, measures to ensure that the root causes 57
are corrected, and measures to ensure that timely action is taken. Follow-up
is performed to ensure the effectiveness of corrective action and that appro-
priate levels of management are informed of the results. HL&P Project Quality
Assurance performs a review for concurrence of selected Ebasco nonconformance
reports and corrective action reports. HL&P Project Quality Assurance also
performs trend analyses to determine the need for corrective action.

The results of the trend analyses are included in the QA monthly activity report which is sent to the Executive Vice President. 59

Compliance of these actions with Project Quality Assurance Plan requirements is verified by HL&P Quality Assurance through the performance of audits and implementation reviews. 57 59

17.1.17A QUALITY ASSURANCE RECORDS

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The HL&P Project Quality Assurance Program requires that a Quality Assurance record system be developed by HL&P and the prime contractors for the Allens Creek Project. The record system provides evidence that activities relating to quality are defined, implemented, and that inspection and test documents contain a description of the type of observation, reference to nonconformance reports, evidence relating to status of observation, date, and inspector identification.

The Project Quality Assurance Plan requires that HL&P and prime contractors establish requirements to ensure that records generated during the design, procurement, construction, properational and start-up testing are identifiable, retrievable and meet the requirements of 10CFR50, and ANSI N45.2.9 as endorsed by Regulatory Guide 1.88, Revision 2. 59

As an alternative, records may be maintained for the Allens Creek Project in a two hour rated fire resistance file room meeting NFPA No. 232 including the following provisions:

a) An automatic fire suppression system and an early warning fire detection system is utilized. 57

b) Records are stored in fully enclosed metal cabinets.

c) Smoking, eating, and drinking should be prohibited within the records storage facility.

d) Work not directly associated with record storage or retrieval is prohibited within the records storage facility.

e) Ventilation, temperature, and humidity control equipment is controlled where they penetrate fire barriers bounding the storage facility.

Compliance with Project Quality Assurance Plan requirements is verified by HL&P Quality Assurance through the performance of audits and implementation reviews.

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17.1.18A AUDITS

The HL&P Project Quality Assurance Plan establishes the requirement that HL&P, prime contractors, and subcontractors develop, document, and implement audit activities which are structured in accordance with the requirements of ANSI 57
N45.2.12 for the Allens Creek Project. As required by the ANSI standard, results of audits are presented for review to management of the audited organization and the HL&P Executive Vice President. Where indicated, HL&P performs follow-up action, including re-audit of the deficient areas. Audits are conducted by personnel qualified in accordance with ANSI N45.2.23 and the results analyzed by QA. Audit reports indicate any quality problems and the 59
effectiveness of the audited QA Program. Reaudits of deficient areas are conducted as necessary to assume implementation of corrective action and recurrence control. Audit results are reported to management for review and assessment.

HL&P has the ultimate responsibility for the auditing of the quality related activities on the project. This responsibility is fulfilled by Houston Quality Assurance, which audits the activities of HL&P, its prime contractors, and their suppliers and subcontractors.

The prime contractors and subcontractors perform quality related audits of internal activities and suppliers of material, components and systems.

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HL&P and Ebasco perform supplemental audits when required, based on such factors as significant changes in the Quality Assurance Program, results of trending programs, or investigations into the root causes of problems.

The HL&P Project Quality Assurance Plan requires that each year an independent outside firm shall conduct an overall audit of the Allens Creek Project Quality Assurance activities. The audit results are presented to the HL&P Executive Vice President and the Project QA Manager. The audit results will be used by HL&P management to evaluate the effectiveness of the Quality Assurance program and to determine the need for changes in the Quality Assurance programs of HL&P and its contractors.

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TABLE 17.0.B-1

ACNGS-SPECIFIC MODIFICATIONS TO EBASCO
TOPICAL REPORT ETR-1001, REVISION 9

1. General

Where the work "client" appears within the appropriate sections of EBASCO's Nuclear Quality Assurance Program Manual it shall be understood to mean "Houston Lighting & Power Company".

2. Section QA-I-2 Organization and Responsibilities

In Paragraph 2.1, a direct line of communication for quality related matters has been established between Ebasco's Chief Quality Assurance Engineer or his alternate and HL&P's Quality Assurance Manager.

3. Section QA-I-5 Quality Assurance Evaluation of Suppliers/Contractors

a. Paragraphs 3.1.2 and 5.1 are modified to allow for alternate methods of evaluation and qualification of supplier's capabilities by methods other than audits by Ebasco. Such methods are detailed as follows:

i) Audits of suppliers by HL&P or others qualified to do so. 59

ii) Historical data is available substantiating the capability of the supplier to provide products which have performed satisfactorily in actual use and were fabricated in accordance with an acceptable quality assurance program. Such historical data shall only qualify suppliers who have provided identical or similar products in the past.

b. Paragraphs 2.3, 4.1, and 5.1 are modified such that in the event Construction Contractors are awarded a contract before review and approval of their quality assurance manual or their facility, but prior to start of any safety-related work, the following shall be complied with:

i) The "Terms and Conditions" section of the Purchase Order will stipulate that the award of the contract is predicated on: 1) Submittal of construction contractors quality assurance manual for review and comment by Purchaser, 2) a satisfactory quality assurance audit by Purchaser of the construction contractors Quality Assurance Program. If the manual review and/or audit are unsatisfact , and if, in the opinion of the Purchaser there is no hope of successful corrective actions, the terms of the contract will permit Purchaser to absolve himself of the contract.

ii) A visit will be made to the home offices of the contractor to discuss the techniques they intend to use in implementing their program at the construction site.

TABLE 17.0.B-1 (Cont'd)

iii) Records of past or similar jobs shall be examined at the contractors office to verify implementation of constructors quality assurance program or at least make an evaluation of the contractors qualifications and capability.

iv) The results of the preceeding reviews will be forwarded to HL&P for their concurrence.

4. Section QA-I-6 Quality Assurance Records

All Quality Assurance documents maintained by Ebasco will be inspected for legibility, proper identification, and microfilmability. The requirement for vendor quality assurance records to be legible and microfilmable will be imposed on vendors through Ebasco's Purchase Order Specification.

5. Section QA-II-4 Purchasing 59

Section QA-II-4 is modified such that Ebasco proposes only the recommendations for purchasing and has no responsibility for issuance of the purchase orders. HL&P has assumed this responsibility and the system of control for issuance of these purchase orders is outlined in HL&P's Quality Assurance Program Manual. Otherwise the remainder of this Section is applicable in its entirety.

6. Section QA-II-5 Supplier Surveillance

Paragraph 3.4 is modified such that after issuance of a purchase order and prior to start of fabrication, the Project Quality Assurance Engineer prepares the Vendor Quality Assurance Plan for approval by HL&P QA. After HL&P approval, the PQAE forwards the Vendor Quality Assurance Plan to the Vendor Quality Assurance Supervisor for use by Vendor Quality Assurance Representatives.

7. Section QA-III-4 Construction Site Procurements

a. In Paragraph 2.7.3 the following is a clarification of the term "Direct Evaluation":

Under certain circumstances and to assist the vendor evaluation group and to expedite the vendor evaluation process, the QA Site Supervisor or qualified members of his staff who he may appoint, may perform facility audits, primarily in their local geographic areas.

b. In Paragraph 3.1.2 delete subparagraph (b).

8. Section QA-III-11 Inspection

The Ebasco QC organization reporting to the Quality Program Site Manager is responsible for the performance of inspection activities during construction.

TABLE 17.0.B-1 (Cont'd)

9. <u>Section QA-III-14 Control of Receiving, Handling and Storage</u>

The following is a clarification of Paragraph 2.0;

When vendor surveillance is not required for certain items purchased by the site organization, receiving inspection will include the review of Certified Material Test Reports, NDE Records, etc. In these cases the review of such documents will be the responsibility of the Quality Control Site Organization. In turn, this operation would be audited by the Quality Assurance Site Group. 59

TABLE 17.0.C-1

GE POSITION ON
REGULATORY GUIDES 1.58, REV. 1 AND 1.46, REV. 0

Regulatory Guide 1.58 - "Qualification of Nuclear Power Plant Inspection Examination and Testing Personnel".

Comply with the provisions of Regulatory Guide 1.58, September, 1980, including the requirements and recommendations in ANSI N45.2.6-1978, except it is recognized that in lieu of the required education and experience requirements expressed in Position 6 of Reg. Guide 1.58, that the alternate mechanism can be ercised as described in Position 10 of the Reg. Guide. These additional controls will be forward fit at the CP issue date and will not be back fit.

Appropriate certification/qualification records are kept of the inspection, examination, and testing personnel.

Regulatory Guide 1.146 - "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants". 59

Comply with the provisions of Regulatory Guide 1.146, August, 1980, including the requirements and recommendations in ANSI N45.2.23-1978, except that it is recognized that in lieu of the education and experience controls described in Section 2.3.1 of ANSI N45.2.23, it is agreed that GE will have documented objective evidence (i.e., procedures and records of written tests) demonstrating that lead auditors possess the required educational and experience skills to fulfill Reg. Guide 1.146 positions. These controls will be forward fit at the CP issue dat and will not be back fit.

TABLE 17.1.2A-1

PROGRAM COMPLIANCE MATRIX 57

SECTIONS ADDRESSING 10CFR50 APPENDIX B REQUIREMENTS

10CFR50 APPENDIX B CRITERIA	HL&P NUCLEAR QUALITY ASSURANCE PROGRAM	ACNGS QA PLAN*	
			57
1) Organization	1.0	2.0, 3.2	
2) QA Program	2.0	1.0	
3) Design Control	3.0	4.0	
4) Procurement Document Control	4.0	5.0, 7.2	
5) Instruction, Procedures, and Drawings	5.0	3.0, 6.3	
6) Document Control	6.0	3.4, 6.3 7.0, 8.0	
7) Control of Purchased Material, Equipment, and Services	7.0	5.0, 6.2, 6.3	59
8) Identification and Control of Materials, Parts, and Components	8.0	6.2, 6.3	
9) Control of Special Processes	9.0	6.2, 6.2	
10) Inspection	10.0	6.2, 6.3	
11) Test Control	11.0	6.2, 6.3	
12) Control of Measuring and Test Equipment	12.0	6.2, 6.3	
13) Handling, Storage, and Shipping	13.0	6.2, 6.3	
14) Inspection, Test, and Operating Status	14.0	6.2, 6.3	

TABLE 17.1.2A-1 (Cont'd)

SECTIONS ADDRESSING 10CFR50 APPENDIX B REQUIREMENTS

10CFR50 APPENDIX B CRITERIA	HL&P NUCLEAR QUALITY ASSURANCE PROGRAM	ACNGS QA PLAN*	
			57
15) Nonconforming Items	15.0	3.5, 6.2, 6.3	
16) Corrective Action	16.0	3.5, 6.2, 6.3	59
17) QA Records	17.0	6.2, 6.3, 7.0	
18) Audits	18.0	6.2, 8.0	

*NOTE: Approximately July 1, 1981 the Allens Creek Nuclear Generating Station Quality Assurance Plan will be rewritten and issued with eighteen sections consistent with the 18 Criteria of 10CFR50 Appendix B.

TABLE 17.1.5A-1

PROJECT PROCEDURE MATRIX

I. Project Procedures	Procedure Number	10CFR50 App. B Criterion	
1) INTRODUCTION			
Project Description and Policy	ACPP-1Q	II	
Project Organization	ACPP-2Q	I	
Purpose and Scope of the Manual	ACPP-3Q	II, V	
Issuance and Control of the Manual	ACPP-4Q	II, V, VI	
Issuance and Control of Procedures	ACPP-5Q	II, V, VI	
2) PROJECT ADMINISTRATION			59
Correspondence Processing and Control	ACPP-52Q	VI	
Telephone Minutes	ACPP-53Q	VI	
Meetings and Meeting Minutes	ACPP-54Q	VI	
Trip and Trip Reports	ACPP-56Q	VI	
Issuing and Controlling Project Directives	ACPP-58Q	II, V, VI	
Administrative Training	ACPP-59Q	II	
3) COST/SCHEDULE			
Processing the Cost Estimate Change Request	ACPP-101	N/A	
Annual Budget Development and Control	ACPP-102	N/A	
Project Estimate Review	ACPP-103	N/A	
4) ENGINEERING REVIEW			
Introduction	ACPP-EIQ	III	
Design Review	ACPP-152Q	III	

TABLE 17.1.5A-1 (Cont'd)

I. Project Procedures	Procedure Number	10CFR50 App. B Criterion
4) ENGINEERING REVIEW (Cont'd)		
Design Review Authority	ACPP-153Q	III, I
AE Design Change Notice (DCN) Processing	ACPP-154Q	III, VI
Design Change Control within HL&P	ACPP-155Q	III
Transmittal of Engineering Correspondence	ACPP-156Q	III
Engineering Review of Procurement Documents	ACPP-157Q	III, IV
Engineering Training	ACPP-158Q	II
Designation and Handling of Confidential Security Documents	ACPP-159Q	VI
Processing Supplier Deviation Requests	ACPP-160Q	III, IV
5) PROCUREMENT		
General Procurement	ACPP-201Q	IV, VII
Establishing Bidders List	ACPP-202Q	IV, VII
Inquiry Issuance	ACPP-204Q	IV, VII
Proposal Evaluation and Supplier Selection	ACPP-205Q	IV, VII, XVII
Purchase Order, Preparation, Changes Approval, and Issuance	ACPP-206Q	IV, VII
Training	ACPP-208Q	I
Procurement File System and File Control	ACPP-210Q	IV, VI
Document Review	ACPP-211Q	IV, VII

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TABLE 17.1.5A-1 (Cont'd)

I. Project Procedures	Procedure Number	10CFR50 App. B Criterion	
5) PROCUREMENT (Cont'd)			
Document Control	ACPP-212Q	IV, VI	
Material Control Organization	ACPP-213Q	VII, VIII, XIII, XIV	
Preventative Maintenance	ACPP-214Q	VII, VIII, XIII, XIV	
Spare Parts	ACPP-215Q	IV, VII, XIII, XIV	
6) ACCOUNTING			
Invoice Review and Approval	ACPP-251	N/A	
Vendor Back Charge	ACPP-252	N/A	59
7) LICENSING			
Reporting Design and Construction Deficiencies to NRC	ACPP-301Q	XV, XVI	
Handling of NRC Inspection Reports and Immediate Action Letters	ACPP-302Q	VI	
Review of NRC Inspections and Enforcement Bulletins and Circulars	ACPP-303Q	VI	
8) CONSTRUCTION	(Later)		
9) STARTUP/WARRANTY	(Later)		
10) PROJECT DIRECTIVES			
Use of Nuclear Division Procedure NDP-130 for Evaluation of Reportable Defects and Deficiencies	AC-PDIR-1	III, VI, VII	
Authorization to Approve for Release Certain Project Correspondence	AC-PDIR-2	I, VI	
Storage Procedure for Early Deliveries or Material for the Allens Creek Nuclear Generating Station	AC-PDIR-3	VIII, XIII, XIV	

TABLE 17.1.5A-1 (Cont'd)

II. Project Site Quality Procedures (PSQP)	Procedure Number	10CFR50 App. B Criterion
Organization & Responsibility of Project QA Personnel	PSQP-A1	I, II
Project Site Quality Procedures	PSQP-A2	V, VI
Handling of NRC Inspection Reports	PSQP-A3	XV, XVI
Control of Site Documentation	PSQP-A4	VI
Non-Nuclear Site Quality Assurance	PSQP-A5	N/A
Document Reviews	PSQP-A6	V, VI
Stop Work	PSQP-A7	XV, XVI
Trend Analysis Administration	PSQP-A8	XVI
Implementation Review	PSQP-A9	II, IV thru XVII
Audit Overview	PSQP-A10	XVIII
Vendor Surveillance Overview	PSQP-A11	IV, VII
Construction QA - Operations QA Interface	PSQP-A12	II, XI, XVII
III. HL&P Houston Quality Assurance Procedures		
Indoctrination & Training of HL&P Houston QA Home Office Personnel	QAP-2.1	II
Procedure for Qualification and Certification of Surveillance Personnel	QAP-2.2	II
Training and Qualification of Audit Personnel	QAP-2.3	II
Procedure for Document Review	QAP-3.1	III
Procedure for Procurement Document Review	QAP-4.2	IV
Standard Definitions and Abbreviations	QAP-5.1	V
Standard Format for Writing and Controlling	QAP-5.2	V

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TABLE 17.1.5A-1 (Cont'd)

		Procedure Number	10CFR50 App. B Criterion
III.	HL&P Houston Quality Assurance Procedures (Cont'd)		
	Issuance and Control of Documents	QAP-6.1	VI
	Procedure for Vendor Quality Surveillance	QAP-7.3	VII
	HL&P Vendor Surveillance	QAP-7.4	VII
	Second Party Vendor Surveillance Category I	QAP-7.5	VII
	Second Party Vendor Surveillance Category II	QAP-7.6	VII
	Review of Nuclear Steam Supply System Quality Assurance Records Packages	QAP-7.7	VII
	Control of Nonconformances	QAP-15.1	XV
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III. HL&P Houston Quality Assurance Procedures (Cont'd)	Procedure Number	10CFR50 App. B Criterion
Document Logging	T2-1	VI
Log Maintenance	T2-2	VI
Document Distribution	T2-3	VI
Storage & Maintenance of Nuclear Records	T2-4	XVII
Document Checkout	T2-5	XVII
Correspondence Serial Number Assignment	T2-6	VI
Correspondence Serial Number Corrections	T2-7	VI
Subject File Number Assignment	T2-8	XVII
NSSS Data Package Handling	T2-10	XVII

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CORPOR

QUALITY PROGRAM SITE MANAGER

QUALITY ASSURANCE SITE SUPERVISOR

- REVIEW OF FIELD CHANGE REQUESTS
- REVIEW OF DESIGN CHANGE NOTICES
- REVIEW AND APPROVE FINAL NDE
- ISSUE QA PLANS FOR SITE PURCHASE ORDERS
- DEVELOP GUIDELINES FOR QUALITY PROCEDURES AND INSTRUCTIONS
- DEVELOP SITE QA/QC PROCEDURES & CHECKLISTS
- REVIEW AND APPROVE NON-CONFORMANCE REPORTS
- SITE PURCHASE ORDER REVIEW

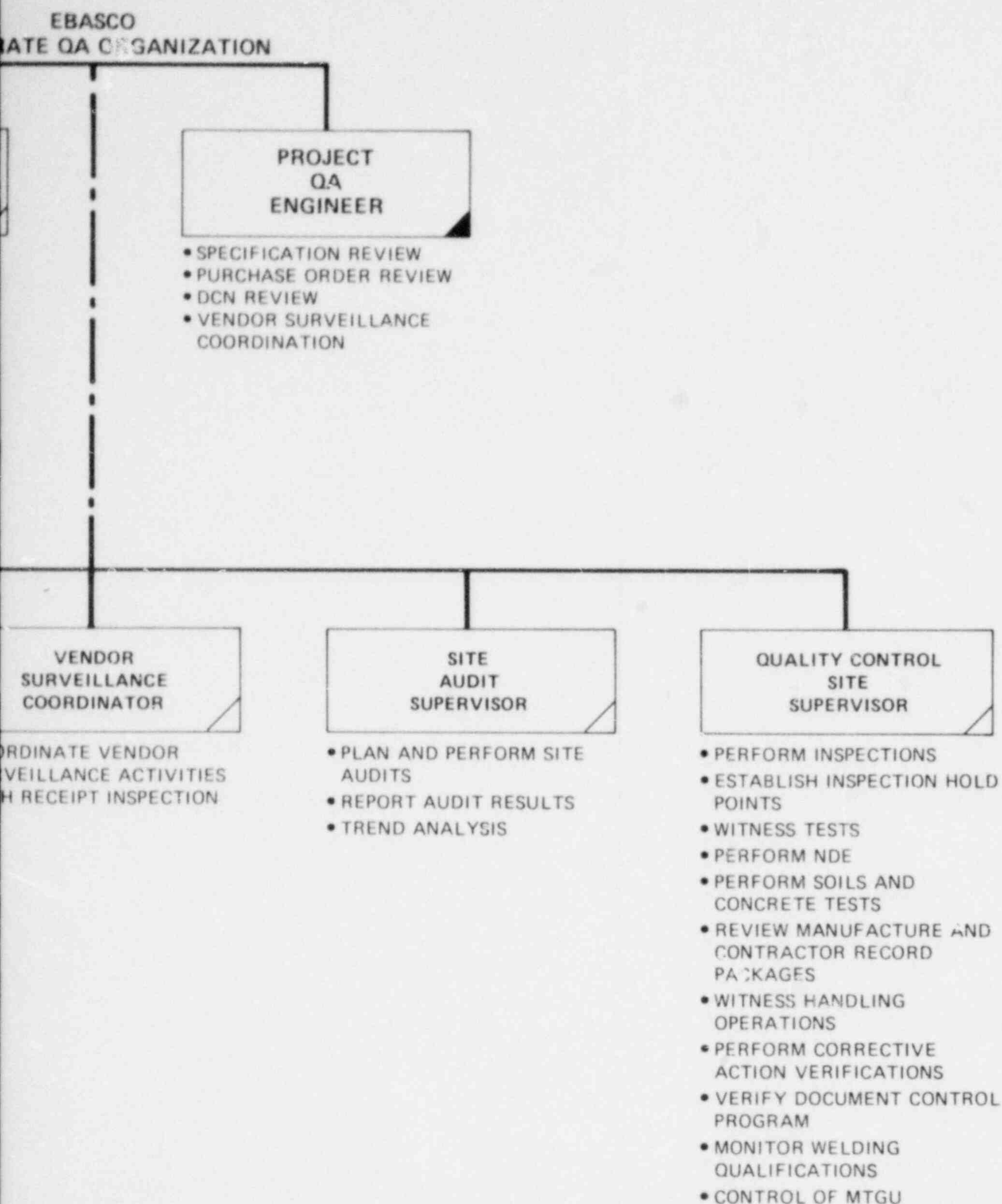
RECORDS CONTROL SUPERVISOR

- RECEIPT, INDEX, ETC. QA RECORDS
- ESTABLISH AND MAINTAIN PROJECT SITE FILE
- REVIEW RECORDS RECEIVED FROM FIELD QC/QA
- PROCESS DEFICIENCY REPORTS

- COO
SUF
WIT

HOME OFFICE

SITE

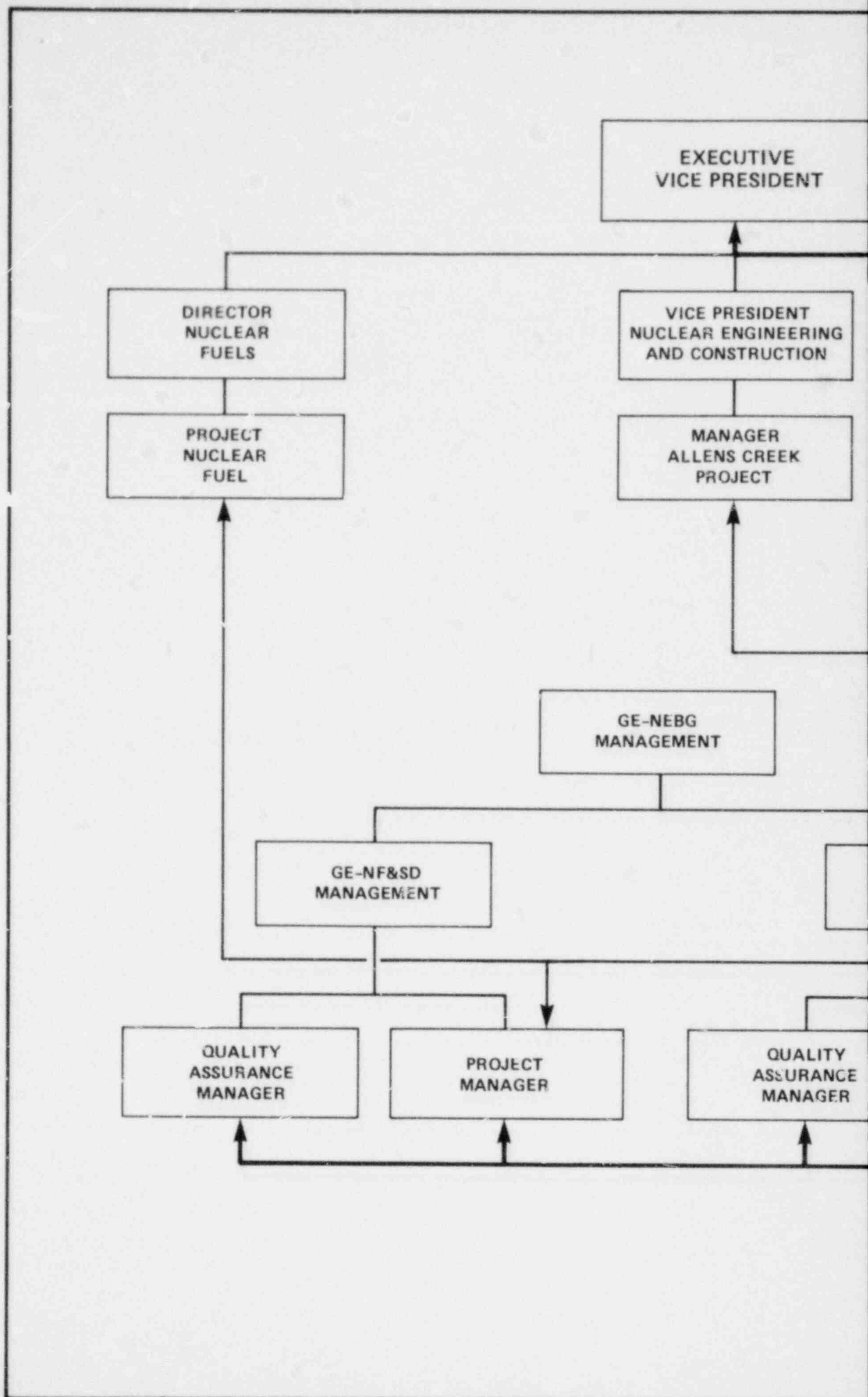


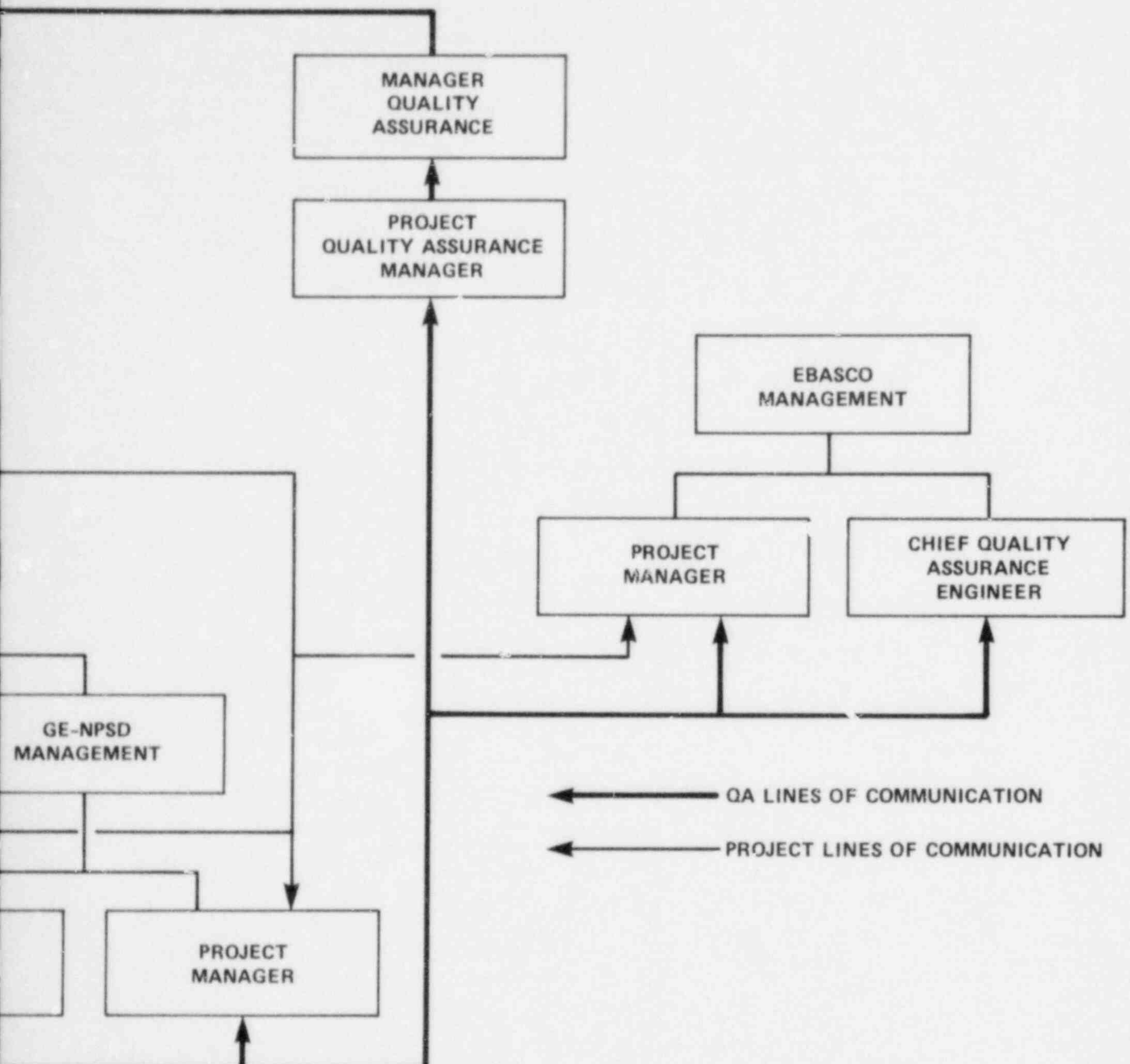
AM. NO. 59, (6/81)

HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

EBASCO PROJECT
QA ORGANIZATION

FIGURE 17.0.B-1


EXECUTIVE VICE PRESIDENT
DIRECTOR NUCLEAR FUELS
VICE PRESIDENT NUCLEAR ENGINEERING AND CONSTRUCTION
PROJECT NUCLEAR FUEL
MANAGER ALLENS CREEK PROJECT
GE-NEBG MANAGEMENT
GE-NF&SD MANAGEMENT
QUALITY ASSURANCE MANAGER
PROJECT MANAGER
QUALITY ASSURANCE MANAGER



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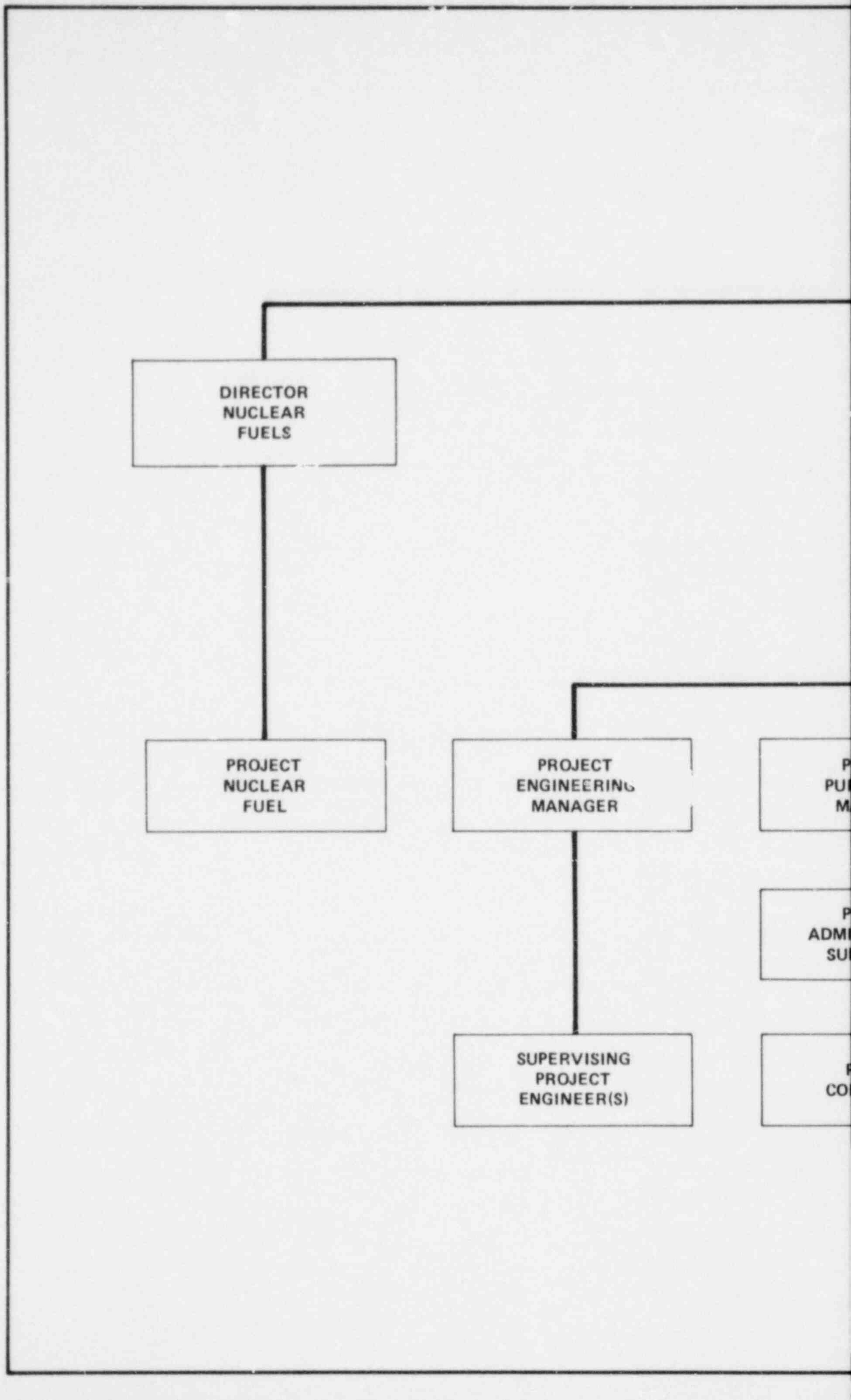
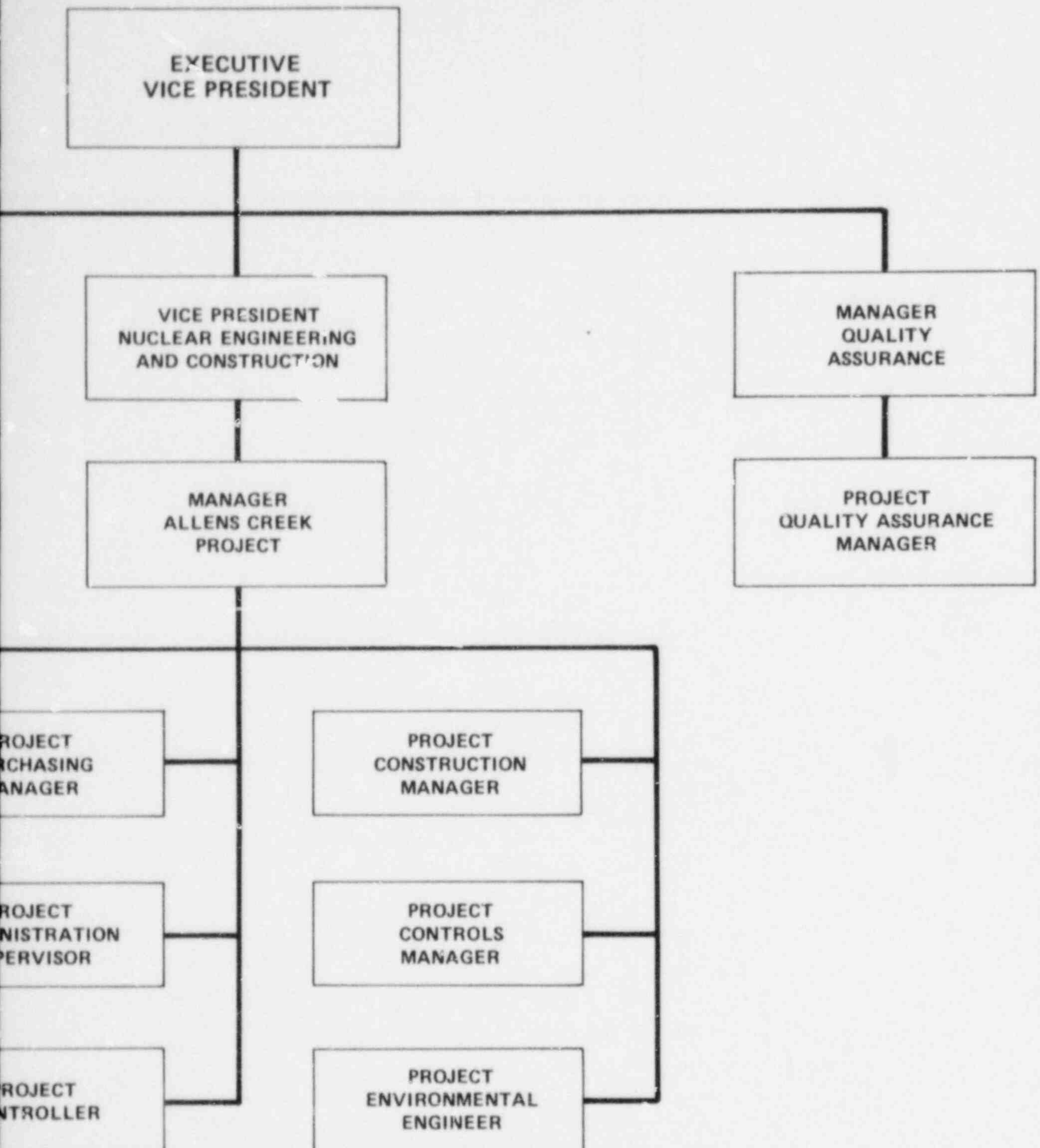
HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

EXTERNAL QA RELATIONSHIPS

FIGURE 17.1.1A–1


DIRECTOR NUCLEAR FUELS
PROJECT NUCLEAR FUEL
PROJECT ENGINEERING MANAGER
SUPERVISING PROJECT ENGINEER(S)



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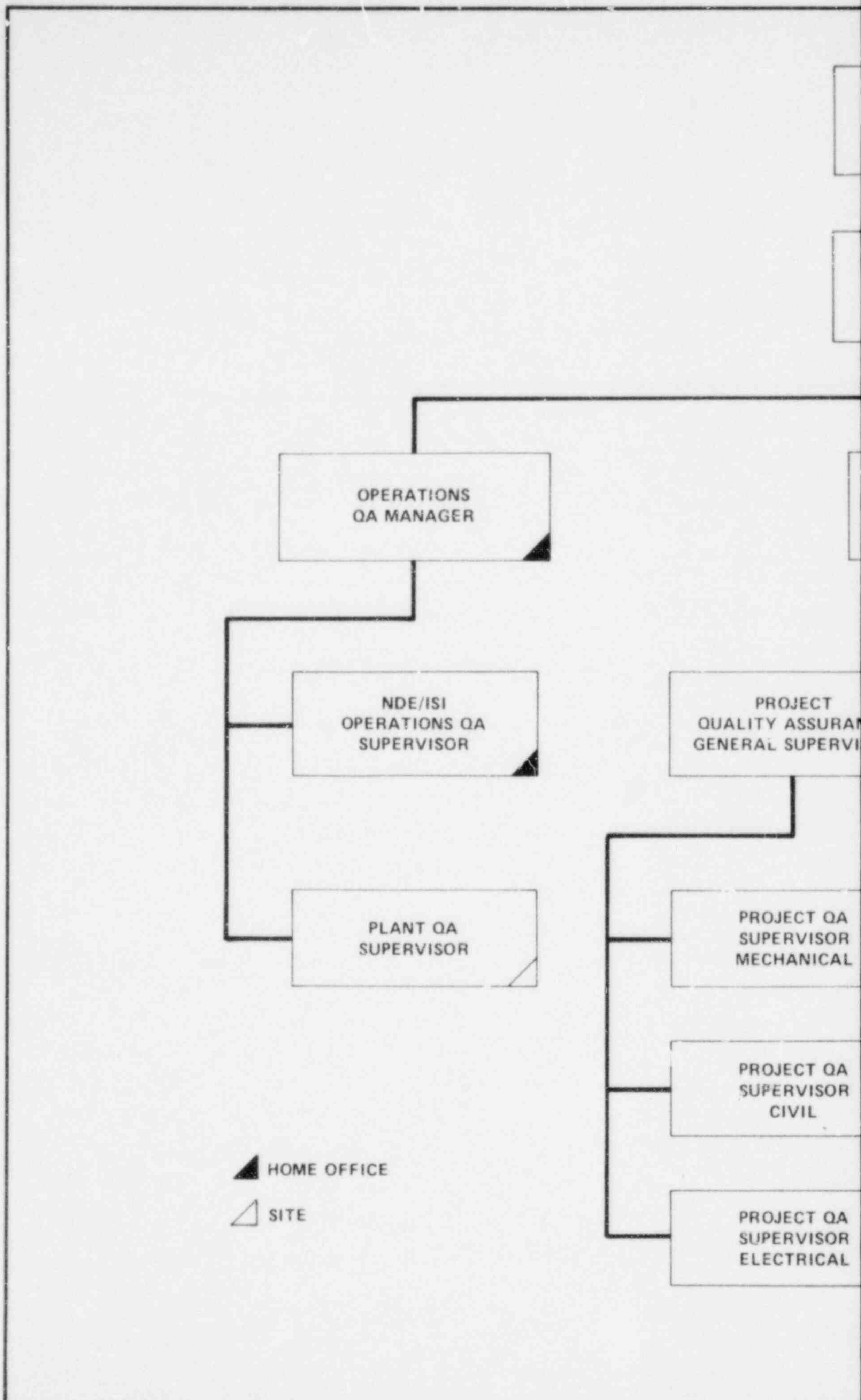
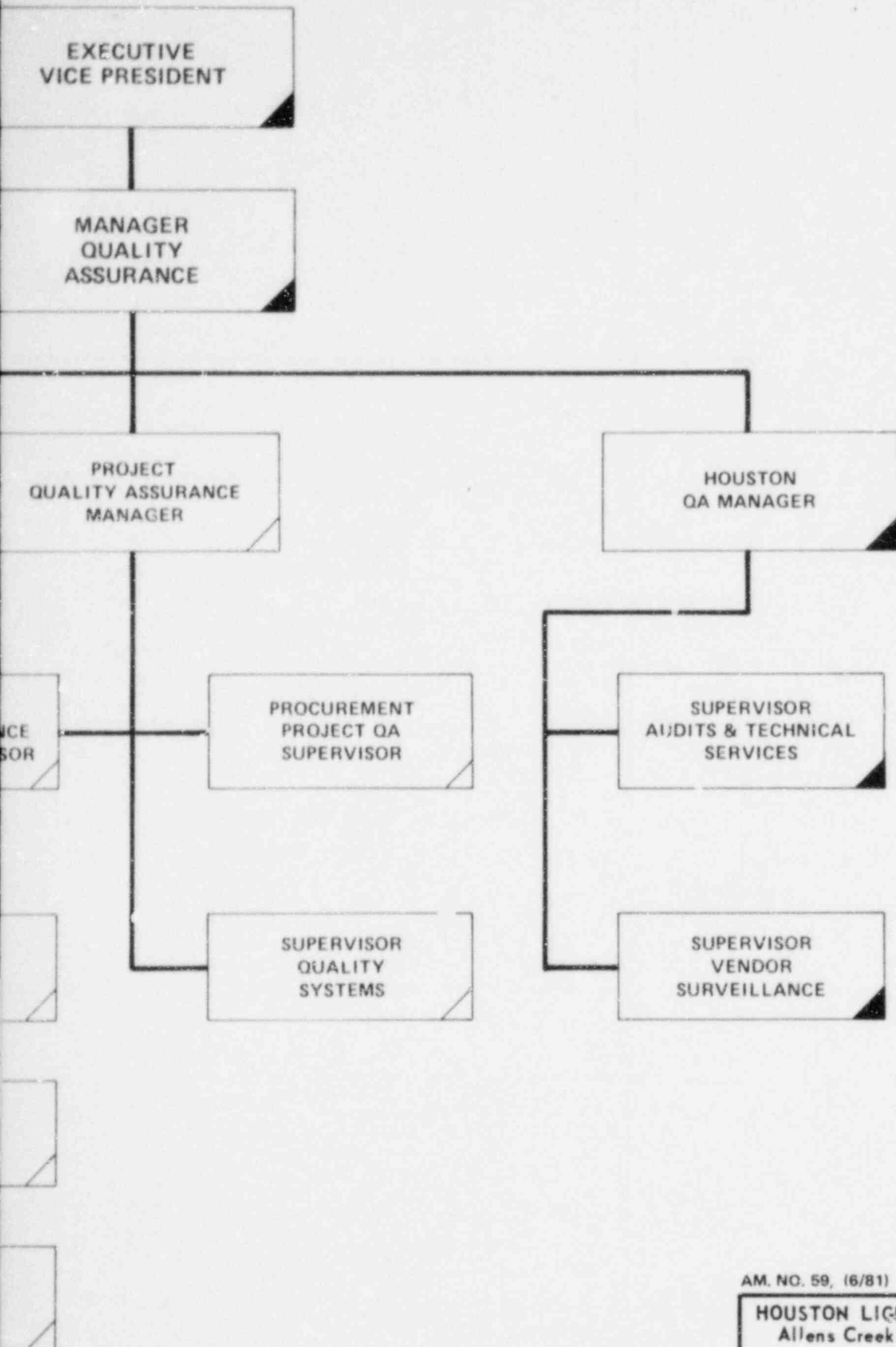
HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

ALLENS CREEK
PROJECT ORGANIZATION
FIGURE 17.1.1A–2


OPERATIONS QA MANAGER
NDE/ISI OPERATIONS QA SUPERVISOR
PLANT QA SUPERVISOR
PROJECT QUALITY ASSURAN
GENERAL SUPERVI
PROJECT QA SUPERVISOR MECHANICAL
PROJECT QA SUPERVISOR CIVIL
PROJECT QA SUPERVISOR ELECTRICAL
HOME OFFICE
SITE



AM. NO. 59, (6/81)

HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

HL&P PROJECT
QA ORGANIZATION

FIGURE 17.1.1A-3

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REGULATORY GUIDE 1.97 42(U)

Rev. 2, 12/80 57

INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT CONDITIONS DURING AND FOLLOWING AN ACCIDENT.

Applicant's Position: 42(U)

The ACNGS design will meet Regulatory Guide 1.97, Rev. 2, except for the following items listed in Table 1 of that guide:

-BWR Core Thermocouples: The primary indicator of core cooling for the BWR is reactor vessel water level, of which multiple indication is available in the main control room. Thermocouples are not reliable devices, and their potential to confuse the operator with erroneous or ambiguous information is deemed to outweigh any perceived benefits of having them available. The detailed arguments 57
against thermocouples are presented in attachment 1 of a letter from D. B. Waters (BWROG) to D. E. Eisenhut (NRC) titled "BWR Emergency Procedures Guidelines, Rev. 1 and responses to related questions", dated 1/31/81.

-Radiation Exposure Rate:

HL&P will develop a plan for the selection and location of radiation monitors in containment penetration areas and in areas where access to service safety equipment is required. This plan will be developed in conformance with the provisions of Reg. Guide 1.97, identify any exceptions and provide justification for any exceptions noted. This plan will be submitted to the NRC prior to the procurement of any of these monitors. 59

-Cooling Water Temperature to ESF components: A range of 30-120°F is provided versus the 30-200°F of the guide. ACNGS uses lake water to cool ESF components. It is inconceivable that the lake could heat up to 200°F, and arbitrarily extending the instrument range diminishes its accuracy over the expected range, decreasing its usefulness to the operator. Thus, the 30-120°F range was selected.

The following are clarifications to the items listed in Table 1 of the guide:

57

-Drywell Sump Level and Drywell Drain Sump Level: These are interpreted to mean the Low Purity Drywell Sump and High Purity Drain Tank Drywell.

-Containment and Drywell Oxygen Concentration: These are interpreted to be required only for pre-inerted containments.

-Suppression Chamber Spray Flow: This is interpreted to mean Containment Spray flow. 59

-Drywell Spray Flow: ACNGS has no drywell sprays.

-Isolation Condenser System Shell Side Water Level: ACNGS has no isolation condenser. 57

-HPCI Flow: This is interpreted to mean HPCS flow.

-Core Spray System Flow: This is interpreted to mean LPCS flow.

-High Radioactivity Liquid Tank Level: This is interpreted to mean Liquid Radwaste System Collection Tanks Level.

-Type A parameters for ACNGS are given in Section 7.5.1.4.2.

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APPENDIX O

RESPONSES TO NUREG 0718, "LICENSING REQUIREMENTS FOR PENDING APPLICATIONS FOR CONSTRUCTION PERMITS AND MANUFACTURING LICENSES" (MARCH, 1981)

NUREG 0718 lists the TMI Action Plan (NUREG 0660) items that the NRC Staff will require to be addressed by near-term construction permit (NTCP) applications. The Action Plan items are divided into five categories of level of detail to be provided in the NTCP submittal. This appendix gives the ACNGS responses to the Action Plan items by NUREG 0718 category. 57

NUREG 0718 CATEGORY 1

"A requirement of a type not applicable to the pending CP or ML applications for any of the following reasons:

a. It can only be addressed in operating license applications or by licensees;

b. It is not directed to CP or ML applicants; 57

c. It does not apply to plants of the type now pending;

d. It has been (or will be) superseded by a more restrictive requirement in the Action Plan or in the regulations;

e. It has already been completed."

RESPONSE

No response is required for NTCP applications.

NUREG 0718 CATEGORY 2

"A requirement of the type customarily left for the operating license stage. The applicant should indicate its recognition of the need for development of operating license or final design requirements and should provide a commitment to implement such requirements in connection with its application for approval of the final design."

RESPONSE 57

The Action Plan items in NUREG 0718 Category 2 concern mostly operations. HL&P has reviewed these items and has concluded that there is nothing to preclude their implementation at the OL stage of licensing. These will be addressed in detail in the FSAR.

NUREG 0718 CATEGORY 3

"Studies (and other research and development activities to provide design development information) of the type customarily left for review at the final stage. However, to satisfy 50.35(a)(3) the staff believes that items in this category should be completed as early as is practicable so that the results can be most effectively taken into account in developing final design details. The applicant should provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and a program to assure that the results of such studies are factored into the final design."

RESPONSE 57

With the exception of Items II.B.8.1 (PRA study) and II.K.3.24 (Adequacy of HPCI and RCIC space cooling), the NUREG 0718 Category 3 items are being resolved between the NRC and the BWR Owners Group (BWROG). These items are being addressed generically and are being conducted so as to envelope all plants participating in the studies. HL&P is participating in the BWROG efforts in this regard and concurs with the positions and recommendations of the Owners Group on the Category 3 items. HL&P commits to incorporate into the ACNGS design the resolution of these items agreed to between the NRC and BWROG.

A summary of the BWROG work done to date on each item is given herein.

II.B.8 RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

NUREG 0718 REQUIREMENT

"Applicant shall:

(1) commit to performing a site/plant-specific probabilistic risk assessment and incorporating the results of the assessment into the design of the facility. The commitment must include a program plan, acceptable to the staff, that demonstrates how the risk assessment program will be scheduled so as to influence system designs as they are being developed.
The assessment shall be completed and submitted to NRC within two years 57
of issuance of the construction permit. The outcome of this study and the NRC review of it will be a determination of specific preventive and mitigative actions to be implemented to reduce these risks. A prevention feature that must be considered is an additional decay heat removal system whose functional requirements and criteria would be derived from the PRA study.

It is the aim of the Commission through these assessments to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. Applicants are encouraged to take steps that are in harmony with this aim."

RESPONSE

HL&P commits to performing an ACNGS specific Reliability Analysis Program (RAP) study. The program status of this study, including schedule, is given
in Appendix 15B. 59

ITEM II.K.2.16 IMPACT OF RCP SEAL DAMAGE FOLLOWING SMALL-BREAK LOCA WITH LOSS OF OFFSITE POWER

NUREG 0718 REQUIREMENT

Applicants shall address the requirements set forth in the Commission Orders issued to operating B&W plants in May 1979 and set forth in Item B.4 of NUREG 0626 regarding the impact of reactor coolant pump seal damage following a small break loss-of-coolant accident with loss of offsite power. Applicants with B&W-designed plants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

SECY-81-20B ITEM II.K.3.25 EFFECT OF LOSS OF AC POWER ON PUMP SEALS

Applicants with BWR plants shall address the requirements set forth in Item B.4 of NUREG-0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs. 57

NUREG 0626 ITEM B.4

The licensees should determine by analysis or experiment, on a plant specific basis, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating current power for at least two hours. Adequacy of the seal design should be demonstrated.

(a) Nature of Study

This concern relates to the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. Adequacy of the seal design should be demonstrated.

The recirculation pump design incorporates a dual mechanical shaft seal assembly to control leakage around the rotating shaft of the recirculation pump. Each assembly consists of two seals built into a cartridge that can be replaced without removing the motor from the pump. Each individual seal in the cartridge is designed for full pump design pressure and can adequately limit leakage in the event that the other seal should fail.

Even though General Electric uses two different recirc pump configurations, the seal designs are essentially the same. Both designs use hydrostatically balanced mechanical shaft seals. Subsequent discussion in this memorandum is applicable to both pump designs.

The recirc pump seals require forced cooling due to the temperature of the primary reactor water and due to the friction heat generated in the sealing surfaces. For all BWR/6 Reactors two systems accomplish this forced cooling: (1) the equipment protection closed cooling water system and (2) the seal purge system. Cooling water, provided by the equipment protection closed cooling water (EPCCW) system, flows through a heat exchanger around the seal assembly. This EPCCW flow cools primary reactor water which flows to the lower seal cavity thereby maintaining the seals at the correct operating temperature. The seal purge system injects clean, cool water from the control rod drive system into the lower seal cavity. This seal purge flow also provides an efficient cooling function for the seals.

The seal cooling system described above is examined to determine the consequenses of a total loss of cooling on the effectiveness of recirculation pump shaft sealing.

(b) Conduct of Study

Under normal conditions, with the primary reactor system at or near rated temperature and pressure and the recirc pumps either operating or secured, both EPCCW and seal purge are operating. These two systems maintain the seal temperatures at approximately 120°F. 57

Recirculation pump vendor test data have shown that the pump seals may begin to deteriorate when seal temperatures exceed 250°F. If an event occurs where both closed cooling water to the pump seal heat exchanger and control rod drive seal purge flow are totally lost, the recirc pump seals will heat up. Vendor test data, taken while operating at approximately 530°F/1040 PSIA, indicate that the seals will heat up, reaching 250°F approximately 7 minutes after the total loss of cooling.

Similar test data indicate that if either one of the seal cooling systems is operating, the seal temperatures remain well below 250°F and no seal deterioration should occur.

If both closed cooling water and seal purge are totally lost, and if the seals heat up to exceed 250°F, seal deterioration may occur, resulting in primary coolant leakage to the drywell. In order to evaluate the fluid loss through a degraded seal, an analysis was performed using the RELAP-4 computer program (see Reference 1).

This analysis modelled the fluid leakage path as a series of fluid volumes with interconnecting junctions, each having appropriate initial conditions. Also, the model assumed gross degradation of the mechanical seals. Gross failure of these seals encompasses warpage, fractures and grooving of the seal faces due to excessive thermal gradients and dirt.

The results of this leakage analysis show that, even with gross degradation of the seals, the leakage would be less than 70 gallons per minute. This amount of leakage is within normal reactor fluctuations and the normal vessel water level control systems will easily compensate for it. Also, 70 GPM is much less than the bounding values of loss-of-coolant accident analyses, hence there are no adverse effects on LOCA analyses.

(c) Completion Date

The study is complete, and was transmitted to the NRC in Reference 1. 57

(d) Program for Implementation of Results

The study concluded that the leakage through a grossly failed RCP seal is of no consequence to any of the LOCA analyses. Therefore, no changes are required to implement the results.

REFERENCES

1. NEDO-24083, "Recirculation Pump Shaft Seal Leakage Analysis", November 1978. (Licensing Topical Report)

ITEM II.K.3.13 SEPARATION OF HPCI AND RCIC SYSTEM INITIATION LEVELS - ANALYSIS AND IMPLEMENTATION

NUREG 0718 REQUIREMENT

Applicants with BWR plants shall address the requirements set forth in Item A.1 of NUREG-0626 as they apply to HPCS and RCIC systems. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

NUREG 0626 ITEM A.1

Currently, the reactor core isolation cooling (RCIC) system and the high pressure coolant injection (HPCI) system both initiate on the same low water level signal and both isolate on the same high water level signal. The HPCI system will restart on low water level but the RCIC will not. The RCIC system is a low-flow system when compared to the HPCI system. The initiation levels of the HPCI and RCIC system should be separated so that the RCIC system initiates at a higher water level than the HPCI system. Further, the RCIC system initiation logic should be modified so that the RCIC system will restart on low water level. These changes have the potential to reduce the number of challenges to the HPCI system and could result in less stress on the vessel from cold water injection. Analyses should be performed by GE to evaluate these changes. The analyses should be submitted to staff and changes should be implemented if justified by the analyses. 57

RESPONSE

a. Nature of Study

This concern covers two aspects of the HPCS and RCIC systems. The first concern is with the initiation levels of these two systems, and requests analysis to determine if benefit could be obtained from allowing the RCIC system to initiate from a higher water level than the HPCS. The second concern is with automatic restart of the RCIC system, and requests analysis to determine if benefit could be gained by introducing this feature.

As previously confirmed in discussions with the NRC the fundamental issue of the separation of initiation setpoints (water level) is the potential benefit of reducing the number of thermal cycles on the reactor vessel and internals resulting from HPCI operation. It is noted that the Allens Creek plant employs HPCS which does not inject via the feedwater nozzle, consequently the fatigue usage on this component is reduced. Thus the study of this issue, which was based mainly on the BWR/4 HPCI arrangement is conservative for Allens Creek.

Analysis was also made to evaluate the proposed logic change for the RCIC system which permits this system to restart automatically following isolation from high water level. This evaluation considered the logic changes involved, effect on system availability, impact on design reliability and the operator/equipment interface.

b. Conduct of Study

a) Setpoint Separation

The analyses conducted are for typical BWR/3 and 4 designs where the HPCI and RCIC systems inject via the feedwater spargers. Later plant designs (BWR/5 and 6) have a separate injection location for HPCS and are less limiting in comparison to the typical BWR/3 and 4 configuration. Differences in the thermal fatigue analyses are identified were appropriate.

The discussion of the study addresses the potential for reducing the thermal cycles due to HPCI and RCIC initiation. The transients considered are those cited in PSAR Chapter 15. Two classes of transients can cause RCIC and HPCI initiation:

1. Initiation of HPCI and RCIC on low water level after feedwater is tripped on high reactor water level. For these transients, the inventory is slowly lost due to decay heat steam generation.

2. Initiation of HPCI and RCIC following a sudden loss of feedwater. For these transients, inventory loss is rapid with HPCI and RCIC initiation occurring approximately 20 seconds after event initiation. 57

The details of this study are provided in Reference 1.

b) Automatic Restart of RCIC System

NUREG-0626, Item A.1, requires evaluation of changes to the Reactor Core Isolation Cooling System to allow automatic restart following a trip of the system at high reactor vessel water level. The evaluation of this change showed that it would contribute to improve system reliability and that it could be accomplished without adverse effect on system function and plant safety. The recommended change would be to relocate the existing high level trip from the RCIC turbine trip valve to the steam supply valve. Once the level reaches a predetermined high level the steam supply valve would be closed. One additional relay in the logic circuitry would be required to accomplish the new function. Closure of the steam supply puts the system in a partial standby configuration because of the existing interlocks associated with closure of this valve. Very little modification to the logic circuitry is required to automate realignment of the system in preparation for low water level initiation. This approach was one of several options considered.

The details of this study are provided in Reference 2.

c. Completion Date

Completed.

d. Program for Implementation of Results

a) Separation of HPCS AND RCIC Setpoints

The results of the analyses for this issue indicate that no significant reduction in thermal cycles can be achieved by separation of these setpoints. It is therefore proposed that the current design values be retained.

b) Automatic Restart of RCIC System

The results of the analyses for this issue indicate that the proposed 57
logic change would contribute to improved system reliability, be of assistance to the plant operator and generally enhance safety. This change can be incorporated into the design, and will be upon NRC approval of the BWROG study, and will be described in the FSAR. 59

References:

1. Letter from R.H. Buchholz (GE) to D.G. Eisenhut (NRC) dated October 1, 1980 and titled "NUREG-0660 Requirement II.K.3.13". 57

2. Letter from D.B. Waters (BWR Owners' Group) to D.G. Eisenhut (NRC) dated December 29, 1980 and titled "BWR Owners' Group Evaluation of NUREG-0737 Requirements".

ITEM II.K.3.16 REDUCTION OF CHALLENGES AND FAILURES OF RELIEF VALVES - FEASIBILITY STUDY AND SYSTEM MODIFICATION

NUREG 0718 REQUIREMENT

Applicants with BWR plants shall address the requirements set forth in Item A.4 of NUREG-0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

NUREG 0626 ITEM A.4

The record of relief valve failures to close for all BWRs in the past three years of plant operation is approximately 30 in 73 reactor years (0.41 failures/reactor year). This has demonstrated that the failure of a relief valve to close would be the most likely cause of a small-break LOCA. The high failure rate is the result of a high relief valve challenge rate and a relatively high failure rate per challenge (0.16 failures/challenge). 57
Typically, five valves are challenged in each event. This results in an equivalent failure rate per challenge of 0.03. The challenge and failure rates can be reduced in the following ways:

(1) Additional anticipatory scram on loss of feedwater,

(2) Revised relief valve actuation setpoints,

(3) Increased emergency core cooling (ECC) flow,

(4) Lower operating pressures,

(5) Earlier initiation of ECC systems,

(6) Heat removal through emergency condensers,

(7) Offset valve setpoints to open fewer valves per challenge,

(8) Installation of additional relief valves with a block or isolation valve feature to eliminate opening of the safety/relief valves (SRVs), consistent with the ASME code,

(9) Increasing the high steam line flow setpoint for main steam line isolation valve (MSIV) closure,

(10) Lowering the pressure setpoint for MSIV closure,

(11) Reducing the testing frequency of the MSIVs,

(12) More stringent valve leakage criteria, and

(13) Early removal of leaking valves.

GE should investigate the feasibility and contraindications of reducing challenges to the relief valves by use of the aforementioned methods. Other methods should also be included in the feasibility study. Those changes which are shown to reduce relief valve challenges without comprising the performance of the relief valves or other systems should be implemented. Challenges to the relief valves should be reduced substantially (by an order of magnitude).

RESPONSE

a. Nature of Study

This report documents a study performed in response to NUREG-0626 item A.4 which requires an evaluation of the feasibility and contraindications of reducing challenges to the relief valves by various methods in BWRs. The report reviews potential methods of reducing the likelihood of stuck open relief valve (SORV) events in BWRs and estimates the reduction in such events that can be achieved by implementing these methods.

Reducing the likelihood of S/RV challenges will directly reduce the likelihood of a SORV. In addition, attention is also given to modifications which could reduce spurious SRV blowdowns and to modifications which could reduce the probability of SRVs to stick open when challenged. 57

b. Conduct of Study

Although the study was precipitated by the consideration of reducing challenges to the Safety/Relief Valves (SRVs), it was recognized that the true objective was to reduce the incidence of Stuck Open Relief Valve (SORV) events. In line with this approach the study also considered reducing the causes of spurious blowdowns and reducing the probability of SRVs to stick open when challenged. The goal of the study was to identify feasible modifications to BWR design and operation, which reduce the frequency of uncontrolled blowdowns by a factor of ten relative to the BWR/4 case, which was used as a base for evaluation.

The details of this study are provided in Reference 1. For the BWR/6 plants such as ACNGS it was concluded that no changes are required to achieve a factor of ten reduction (relative to operating experience) because:

1. design features which reduce SRV challenges are already incorporated.

2. the two-stage Crosby valves to be used are less likely to stick open due to design differences from the three-stage Target Rock valves on which the operating experience is based.

c. Completion Date

Complete

d. Program for Implementation of Results

The study indicates that the required factor of ten improvement relative to operating experience is met by the present design. Thus, no changes are required to implement the results. 57

References

1. Letter from D.B. Waters (BWROG) to D.G. Eisenhut (NRC) dated March 31, 1981 and titled "BWR Owners Group Evaluation of NUREG 0737 Requirements."

ITEM II.K.3.18 MODIFICATION OF ADS LOGIC - FEASIBILITY STUDY AND MODIFICATION FOR INCREASED DIVERSITY FOR SOME EVENT SEQUENCES

NUREG 0718 REQUIREMENT

Applicants with BWR plant shall address the requirements set forth in Item A.7 of NUREG-0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

NUREG 0626 ITEM A.7

The ADS actuation logic should be modified to eliminate the need for manual actuation to assure adequate core cooling. A feasibility and risk assessment study is required to determine the optimum approach. One possible scheme which should be considered is ADS actuation on low reactor vessel water level provided no HPCI or HPCS system flow exists and a low pressure ECC system is running. This logic would complement, not replace, the existing ADS actuation logic.

RESPONSE

a. Nature of Study

A feasibility and risk assessment study was made to examine possible modifications to the ADS initiation lo 'c, which would eliminate the need for manual initiation to assure adequate core cooling. For some non-line break events which are further degraded by assuming non-availability of all high pressure injection systems, manual depressurization of the reactor is required in order to employ the low pressure injection systems. This study examines the advantages and disadvantages of a number of possible ADS initiation logic modifications. 57

b. Conduct of Study

Five ADS logic alternatives were considered: the current design, and four logic modifications. These four modifications are 1) elimination of the high drywell pressure trip, 2) addition of a timer that bypasses the high drywell pressure trip requirement after a certain length of time, 3) addition of a suppression pool temperature trip in parallel with the high drywell pressure trip, and 4) the addition of high pressure system flow measurement and logic in parallel with the high drywell pressure trip.

Each of the options is evaluated on the basis of whether it assures adequate core cooling without operator action for isolations and SORV's. Each option is also evaluated for its capability to assure adequate core cooling without operator action. For these analyses it is assumed that all high pressure systems have failed and the ADS must depressurize the vessel and allow the low pressure systems to inject. The modeling used in these analyses is the same as that used in NEDO-24708.

The details of this study are provided in Reference 1.

c. Completion Date

The study is complete and was transmitted to the NRC by Reference 1.

d. Program for Implementation of Results

The BWROG concluded that an ADS modification which adds a bypass timer on ECCS initiation level or removal of the high drywell pressure trip would be beneficial. These changes would not have any major impacts on the plant design. They can be readily incorporated, and will upon NRC/BWROG resolution of the item. 57

REFERENCES

1. Letter from D.B. Waters (BWROG) to D.G. Eisenhut (NRC) dated March 31, 1981 and titled "BWR Owners Group Evaluation of NUREG 0737 Requirements."

ITEM II.K.3.21 RESTART OF CORE SPRAY AND LPCI SYSTEMS ON LOW LEVEL - DESIGN AND MODIFICATION

NUREG 0718 REQUIREMENT

Applicants with BWR plants shall address the requirements set forth in Item A.10 of NUREG 0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

NUREG 0626 ITEM A.10

The core spray and LPCI system flow may be stopped by the operator. These systems will not restart automatically on loss of water level if an initiation signal is still present. The core spray and LPCI system logic should be modified so that these systems will restart if required to assure adequate core cooling. Because this design modification affects several core cooling modes under accident conditions, a preliminary design should be submitted for staff review and approval prior to making the actual modification.

RESPONSE 57

a. Nature of Study

In this item, the NRC Suggested Certain Modifications to the Core Spray (CS) and the Low Pressure Coolant Injection (LPCI) Emergency Core Cooling Systems (ECCS) that are provided as part of the BWR ECCS network. The NRC suggestions center on incorporating additional control system logic to provide automatic system restart from a low reactor water level signal following actions by the operators to terminate system operation. The NRC concern is that the reactor operators may terminate ECCS operation when a high reactor water level condition exists but may neglect to reinitiate the systems if a low condition recurs. The study, which covers the Allens Creek plant design, includeds the LPCI and both the low and high pressure core spray systems.

Intuitively, it might appear that additional ECCS automation would be purely beneficial since this would supposedly provide added protection against operator errors and omissions. However, these perceived benefits of extended system automation must be measured against the very real penalties of increased system complexity, reduced system reliability and restricted operator flexibility for dealing with unanticipated events. These considerations are not amenable to precise quantification and control system design decisions must of necessity involve judgements as to relative importance of these competing influences.

b. Conduct of Study

In order to determine if any overall benefit is to be derived from the postulated design changes it is necessary to consider the integral nature of the ECCS network, and how the ECCS interacts with other plant systems. The study provides an overview discussion of the generic GE ECCS design philosophy and design practices as they govern ECCS initiation and operator control of these systems. The need for operator override is identified, and how this feature provides for improved overall system reliability. Considerable significance is attached to the complexity of logic and hardware, which would be required to deal with relatively long-term transients involving core and containment cooling, on a purely automatic basis. Several long-term transient scenarios are presented to support this contention.

The details of this study are provided in Reference 1. 57

c. Completion Date

The study is complete and was transmitted to the NRC Reference 1.

d. Program for Implementation of Results

The study concluded that while changes to the LPCI/LPCS logic would not have a net positive safety effect, modifications to the HPCS logic to assure a restart on low reactor water level would. This can be readily incorporated into the ACNGS design and will upon NRC/BWROG resolution of this item.

REFERENCES

1. Letter from D.B. Waters (BWR Owners' Group) to D.G. Eisenhut (NRC) dated December 29, 1980 and titled "BWR Owners' Group Evaluation of NUREG-0737 Requirements".

ITEM II.K.3.24 CONFIRM ADEQUACY OF SPACE COOLING FOR HPCI AND RCIC SYSTEMS

NUREG 0718 REQUIREMENT

Applicants with BWR plants shall address the HPCI and RCIC systems requirements set forth in Item B.3 of NUREG 0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

NUREG 0626 ITEM B.3

Long-term operation of the RCIC and HPCI systems may require space cooling to 57
maintain the pump room temperatures within allowable limits. The licensees should verify for each plant the acceptability of the consequences of a complete loss of alternating current power. The RCIC and HPCI systems should be designed to withstand a complete loss of alternating current power to their support systems, including coolers, for at least two hours.

RESPONSE

The HPCS and RCIC systems are designated as safety-related for ACNGS, and as such, are designed to operate independent of offsite power*. As with all safety systems, HPCS and RCIC are serviced by a safety-related cooling system, which is itself independent of offsite power. The cooling system is sized and designed to maintain a suitable environment for HPCS and RCIC components following a loss of offsite power.

*The BWR Owners Group received clarification that the intent of this item was 59
to assume loss of offsite power only. This was confirmed by letter, D B 57
Waters (BWROG) to D.G. Eisenhut (NRC) "Clarification of NUREG 0737 Items
II.K.3.24 and II.K.3.25, dated 1/23/81. 59

ITEM II.K.3.28 VERIFY QUALIFICATION OF ACCUMULATORS ON ADS VALVES

NUREG 0718 REQUIREMENT

"Applicants with BWR plants shall provide information to assure that the ADS valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation while taking no credit for nonsafety-related equipment or instrumentation. Air (or nitrogen) leakage through valves must be accounted for to assure that enough inventory of compressed air (or nitrogen) will be available to cycle the ADS valves. Applicants shall commit that these requirements will be met in the final design at the OL stage.

In addressing this item prior to CP issuance, applicants should note that safety analysis reports claim that air (or nitrogen) accumulators for the ADS valves provide sufficient capacity (inventory) to cycle these valves open five times at design pressures. Also, General Electric has stated that the emergency core cooling systems are designed to withstand a hostile environment and still perform their functions for 100 days following an accident." 57

RESPONSE

The present ADS air accumulators are sized to cycle the ADS valves twice against 70% of containment design pressure (or five times against containment atmospheric pressure) plus component leakage for seven days. Post-accident access to replenish the air supply (assuming that the supply compressors are inoperative) is being confirmed as part of the post-accident shielding study in response to Item II.B.2. The radiation environmental qualification for the ADS air accumulators and associated components for at least 100 days will be confirmed by this study as well.

HL&P is participating in the BWR Owners Group efforts to agree with the NRC on a uniform design basis for ADS air accumulator sizing. The results of this effort will be adopted for ACNGS and design changes made if necessary.

ITEM II.K.3.44 EVALUATION OF ANTICIPATED TRANSIENTS WITH SINGLE FAILURE TO VERIFY NO SIGNIFICANT FUEL FAILURE

NUREG 0718 REQUIREMENT

Applicants with BWR plants shall address the requirements set forth in Item A.14 of NUREG 0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

NUREG 0626 ITEM A.14

For anticipated transients combined with the worst single failure and assuming proper operator actions, licensees should demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery. Transients which result in a stuck-open relief valve should be included in this category.

RESPONSE

a. Nature of Study

Analyses of the worst anticipated transient (loss of feedwater event) with the worst single failure (loss of a high pressure inventory makeup or heat removal system) were performed to demonstrate adequate core cooling capability. It is shown that, for the BWR/2 through BWR/6 plants, adequate core cooling is maintained for these worst-case conditions. Analyses of further degraded conditions involving a stuck-open relief valve in addition to the worst transient and single failure were also performed. The results show that, with proper operator action, the core remains covered and therefore adequate core cooling is achieved. 57

b. Conduct of Study

Of the two alternate criteria allowed in Item A.14 of NUREG 0626, the study demonstrated that for the combination of anticipated transients with the worst single failure, the reactor core remains covered until stable conditions are achieved. The following assumptions are also made:

a. A representative plant of each BWR product line, BWR/2 through BWR/6, is used to represent all of the plants of that product line. The BWR/6 analyses are applicable to ACNGS.

b. The anticipated transients as identified in NRC Regulatory Guide 1.70, Revision 3 were considered.

c. The single failure is interpreted as an active failure.

d. All plant systems and components are assumed to function normally, unless identified as being failed.

For a BWR/6 plant such as Allens Creek the study indicates that the worst combination is a loss of feedwater with failure of the HPCS. The study further shows that even for the further degraded condition of a stuck open relief valve in addition to the worst single failure/worst transient combination, the core can be kept covered.

The details of this study are provided in Reference 1.

c. Completion Date

The study is complete, and was transmitted to the NRC in Reference 1. 57

d. Program for Implementation of Results

The study concluded that there are no anticipated transient/single failure combinations which result in significant fuel damage. Therefore, no changes are required to implement the results.

REFERENCES

1. Letter from D.B. Waters (BWR Owners' Group) to D.G. Eisenhut (NRC); dated December 29, 1980 and titled: "BWR Owners' Group Evaluation of NUREG-0737 Requirements".

ITEM II.K.3.45 EVALUATE DEPRESSURIZATION WITH OTHER THAN FULL ADS

NUREG 0718 REQUIREMENT

Applicants with BWR plants shall address the requirements set forth in Item A.15 of NUREG-0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

NUREG 0626 ITEM A.15

Analyses to support depressurization modes other than full actuation of the ADS (e.g., early blowdown with one or two SRVs) should be provided. Slower depressurization would reduce the possibility of exceeding vessel integrity limits by rapid cooldown.

RESPONSE

a. Nature of Study

This feasibility study addresses NUREG-0626, Item A.15, and provides an evaluation of alternate modes of reactor depressurization than full actuation of the Automatic Depressurization System (ADS). The study includes the BWR/6 product line and therefore the Allens Creek plant. 57

b. Conduct of Study

Depressurization by full ADS actuation constitutes a depressurization from about 1050 psig to 180 psig in approximately 3.3 minutes. Such an event, which is not expected to occur more than once in the lifetime of a plant, is well within the design basis of the reactor pressure vessel. This conclusion is based on the analysis of several transients requiring depressurization via the ADS valves. Results of these analyses indicate that the total vessel fatigue usage is less than 1.0. Therefore, no change in the depressurization rate is necessary. However, to comply with NUREG-0626, Item A.15, reduced depressurization rates were analyzed and compared with the full ADS actuation. The alternate modes considered cause vessel pressure to traverse the same pressure range in 1) depressurization case 1 (ranges from 6-10 minutes depending on plant size and ADS capacity and 2) depressurization case 2 (ranges from 15-20 minutes). The case 2 depressurization bounds the possible increase in depressurization time by producing an undesirably long core uncovered time. The case 1 depressurization gives the results of an intermediate depressurization. These modes are achieved by opening a reduced number of relief valves.

The details of this study are provided in Reference 1.

c. Completion Date

The study is complete and was transmitted to the NRC in Reference 1.

d. Program for Implementation of Results

The study concluded that there is no benefit to be derived from the use of reduced blowdown rates. Therefore, no changes are required to implement the results. 57

REFERENCES

1. Letter from D.B. Waters (BWR Owners' Group) to D.G. Eisenhut (NRC); dated December 29, 1980 and titled: "BWR Owners' Group Evaluation of NUREG-0737 Requirements".

NUREG 0718 CATEGORY 4

"A requirement to demonstrate that any additional design, development and implementation necessary to satisfy the requirement (or to satisfy the goals of the task whose requirements are to be developed in the future) will be satisfactorily completed by the operating license stage. This is the type of information customarily required at the construction permit stage to satisfy 50.35(a)(2), or to satisfy ALAB-444 with respect to generic issues."

RESPONSE

Responses to the applicable Category 4 items are given herein, including PSAR level of information and detail where design is involved. It should be noted that for most of these items there are no questions as to the ability to implement the requirement prior to issuance of an operating license.

The following Category 4 items are not applicable to ACNGS:

II.E.1.2 Auxiliary Feedwater System Automatic Initiation and Flow indication - applicable to PWRs only. 57

II.E.3.1 Reliability of Power Supplies for Natural Circulation - applicable to PWRs only.

II.E.5.1 B&W Reactors Design Evaluation - applicable to B&W NSSS plants only.

II.E.5.2 B&W Reactor Transient Response Task Force - applicable to B&W NSSS plants only.

II.G.1 Power Supplies for Pressurizer Relief Valves, Block Valves and Level Indicators - applicable to PWRs only.

II.K.2.9 Procedures and training to initiate and control AFW independent of integrated control system - applicable to B&W NSSS plants only.

II.K.2.10 Hard-wired safety grade anticipatory reactor trips - applicable to B&W NSSS plants only.

II.K.3.11 Control use of PORV supplied by Control Components Inc - applicable to PWRs only.

ITEM I.A.4.2 LONG-TERM TRAINING SIMULATOR UPGRADE

NUREG 0718 REQUIREMENT

Applicants shall describe their program for providing simulator capability for their plants. In addition, they shall describe how they will assure that their proposed simulator will correctly model their control room. Applicants shall provide sufficient information to permit the NRC staff to verify that they will have the necessary simulator capability to carry out the actions described in this Action Plan item as well as Action Plan Item II.K.3.54. Applicants shall submit, prior to the issuance of construction permits, a general discussion of how the requirements will be met. Sufficient details shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses. 57

RESPONSE

HL&P intends to use the Black Fox simulator for operations training in support of ACNGS. The simulator will be used for:

(1) continuing training of licensed personnel

(2) initial attainment of cold licenses

(3) supplemental training for advancement from RO to SRO

(4) training of home office and plant staff engineering personnel and auxiliary operators.

In addition, the licensed operator training program will meet the requirements of the following documents:

1. ANS 3.1, 4/10/81, Standard for Qualification and Training of Personnel for Nuclear Power Plants.

2. 10 CFR Part 55, Operators Licenses.

3. REG. GUIDE 1.149, Nuclear Power Plant Simulators for Use in Operator Training. 59

These requirements will be accomplished in a timely manner to support startup and operation. Table I.A.4.2-1 provides an estimate of the manpower schedule to support operator training and assignment. The ACNGS license candidate training program will be typical of that defined in ANS 3.1, Appendix A.

The control room at both Black Fox and ACNGS are Nuclenet 1000 advanced control room designs (Reference Item I.D.1 - Control Room Design Reviews). The control room floor plans for the primary control panels are the same (see Figures I.A.4.2-1 and I.A.4.2-2). A comparison of functions is found in Table I.A.4.2-2. A general discussion of the similarities and differences in the panels is discussed below.

I.A.4.2

Panels P680, P601, P678

Panel P680 is the principal plant operations control console, and the day-to-day normal plant operations functions, short response functions, and reactivity control functions are located on this panel.

The functions/systems on this panel are the same for both plants. The layouts are essentially the same, except for the condensate pumping system, feedwater pumping system, and generator controls. These systems are the responsibility of the utility, which makes them utility unique. The other systems are designed by General Electric. 59

Panel P601 is the reactor core cooling benchboard and contains the NSSS safety systems and NSSS long response functions. The operator uses this panel during abnormal conditions and testing. The functions/systems on this panel are the same for both plants. The layouts are essentially the same, except for the minor differences in layouts of the RHR system. The panel is designed by General Electric.

Panel P678 is the Standby Information Panel. It is used as a support function for the P680 panel. It contains no control functions. The functions/systems are the same; however, the layouts are different due to utility preferences. The differences are minor and do not interfere with operator performance.

During simulator training operators using these panels will learn how to use the CRT displays and their interaction with the associated controls. In addition, the operators will learn to use the benchboards during the transients and accidents. The control rooms are essentially duplicated for these very important functions so that the operators will gain experience directly applicable to ACNGS through the simulator training. 57

Panels P870, P800, P868, P869, P877

Panel P870 is the BOP control benchboard and contains the BOP long response functions and non-frequent use functions. The panel is designed by the individual utility to be plant specific. The functions/systems on this panel are essentially the same; however, those functions/systems that are not on P870 are found on other panels in the control room. The layouts of the panels are different because of differences in utility philosophy.

Panel P877 contains controls for the standby diesels and support systems. The functions/systems are the same; however, the layouts are different because this panel is designed by the utility. 59

Panel P800 is the BOP auxiliary control panel for Allens Creek and contains controls for the HVAC systems.

Panel P800 is the electrical auxiliaries panel for Black Fox. Electrical auxiliaries are found on ACNGS panel P870. This panel is also designed by the utility.

Panels P868 and P869 are the containment isolation panels for ACNGS. Similar information is found on P870 for Black Fox. 59

The benchboards utilized for balance of plant control are similar in function although not identical in layout between the plants. 57 From this functional similarity the operators will gain experience useful for ACNGS operations through the simulator training.

Currently it is expected that four utilities with plants having Nuclenet 1000 control rooms will use the Black Fox simulator for training. It is anticipated that adequate time can be made available to meet the needs of all these utilities. A plan to ensure that adequate time for training and operator licensing examinations will be available on the Black Fox simulator 59 or a plant specific simulator will be developed.

General Electric, the owner of the Black Fox simulator, currently intends to keep the training program updated with NRC requirements on simulator training. The simulator also will have the capability to simulate small break LOCAs and other transients as required. 57

In summary, the Black Fox simulator is a good representation of the Allens Creek Control Room as both are advanced control rooms on a BWR 6 plant. Second, adequate training time is expected to be available for those utilities with these type control rooms, and third, the training program is expected to be kept current with training requirements.

TABLE I.A.4.2-1

MANPOWER ESTIMATE DURING CONSTRUCTION
(in equivalent number of men)

Year	Nuclear Operations	SRO and RO Certification
Start of Construction	4	0
1	6	0
2	13	0
3	18	0
4	29	0
5	39	20
6	59	10
7	38	10
8	21	8
Total	227	48

59

TABLE I.A.4.2-2

COMPARISON OF ALLENS CREEK AND BLACK FOX CONTROL ROOM DESIGN

P - Panel Number

Insert Group XXX - Removable section of panel upon which system instrumentation/controls are located

System/Function	Allens Creek	Black Fox
1) Reactor water clean system	P680, insert group 1	P680, insert group 1
2) Condensate pumping system		
a) condensate pumps	P680, insert group 2	P680, insert group 2
b) condensate booster pumps	P680, insert group 2	P680, insert group 2
c) condensate demineralizer	P680, insert group 2	P680, insert group 2
d) heater drain pumps	P680, insert group 2	P870
3) Feedwater pumping and reactor level control system		
a) TDRFP controls	P680, insert group 3	P680, insert group 3
b) MDRFP controls	P680, insert group 3	
c) reactor level controls	P680, insert group 3	P680, insert group 3
4) Reactor recirculation system	P680, insert group 4	P680, insert group 4
5) Rod control and information system	P680, insert group 5	P680, insert group 5
6) Neutron monitoring system	P680, insert group 6	P680, insert group 6
7) Steam bypass and pressure regulator system	P680, insert group 7	P680, insert group 7
8) Main turbine control system	P680, insert group 8 P680, insert group 9	P680, insert group 8
9) Generator control system	P680, insert group 9	P680, insert group 9
10) Performance monitoring system	P680, insert group 10	P680, insert group 10
11) HPCS		
a) HPCS water system	P601, insert group 16	P601, insert group 16
b) HPCS power supply	P601, insert group 16	P601, insert group 16
12) RCIC	P601, insert group 21	P601, insert group 21
13) LPCS	P601, insert group 21	P601, insert group 21
14) RHR Train A		
a) Div 1 Essential Services Cooling Water	P601, insert group 20	P601, insert group 20
b) Div 1 Reactor Water Level Monitoring	P601, insert group 20	P601, insert group 20

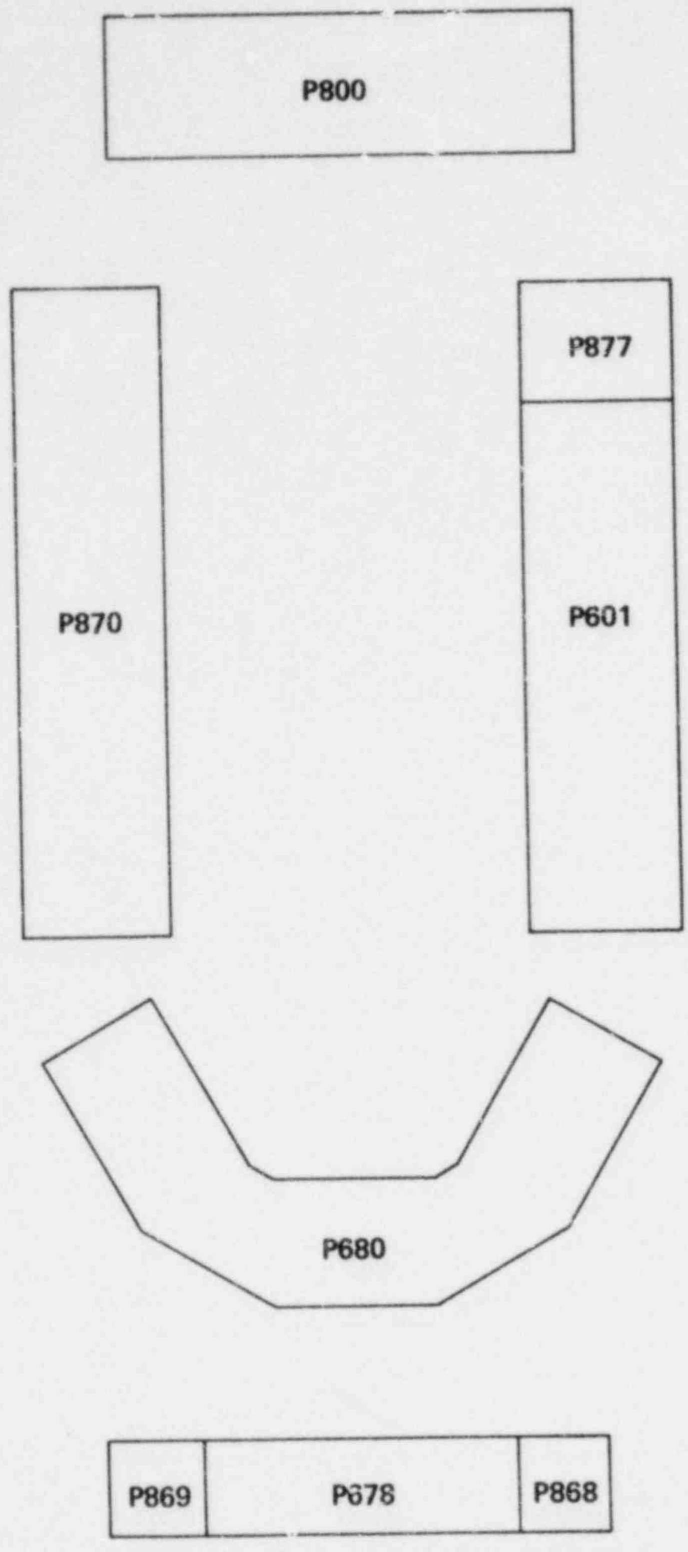
59

TABLE I.A.4.2-2 (Cont'd)

System/Function	Allens Creek	Black Fox	
15) RHR Train B			
a) Div 2 Essential Services Cooling Water	P601, insert group 17	P601, insert group 17	
b) Div 2 Reactor Water Level Monitoring	P601, insert group 17	P601, insert group 17	
16) MSIV Leakage Control System	P601, insert group 18 P601, insert group 19	P601, insert group 18 —	
17) Standby Liquid Control System	P601, insert group 18 P601, insert group 19	P601, insert group 18 P601, insert group 19	
18) ADS Air System	P601, insert group 18 P601, insert group 19	— —	
19) Nuclear Steam Supply Shutdown System	P601, insert group 18	P601, insert group 18	
20) Main steam line	P601, insert group 19	P601, insert group 19	
21) ADS/Relief Valves	P601, insert group 19	P601, insert group 19	
22) Div 1 Containment Isolation	P868	P870	
23) Div 2 Containment Isolation	P869	P870	59
24) Drywell and Containment Sumps	P601, insert group 22	P601, insert group 22	
25) CRD	P601, insert group 22	P601, insert group 22	
26) Turbine vibration monitoring	P678	P678	
27) DCS support indication	P678	P678	
28) Auxiliary electrical systems	P870	P800	
29) Turbine auxiliaries	P870	P870	
30) Steam System	P870	P870	
31) Air evacuation system	P870	—	
32) Condensate & feedwater system	P870	P870	
33) Off-gas system	P870	P870	
34) Circulating water system	P870	P870	
35) Process radiation monitoring	P870	back row panel	

TABLE I.A.4.2-2 (Cont'd)

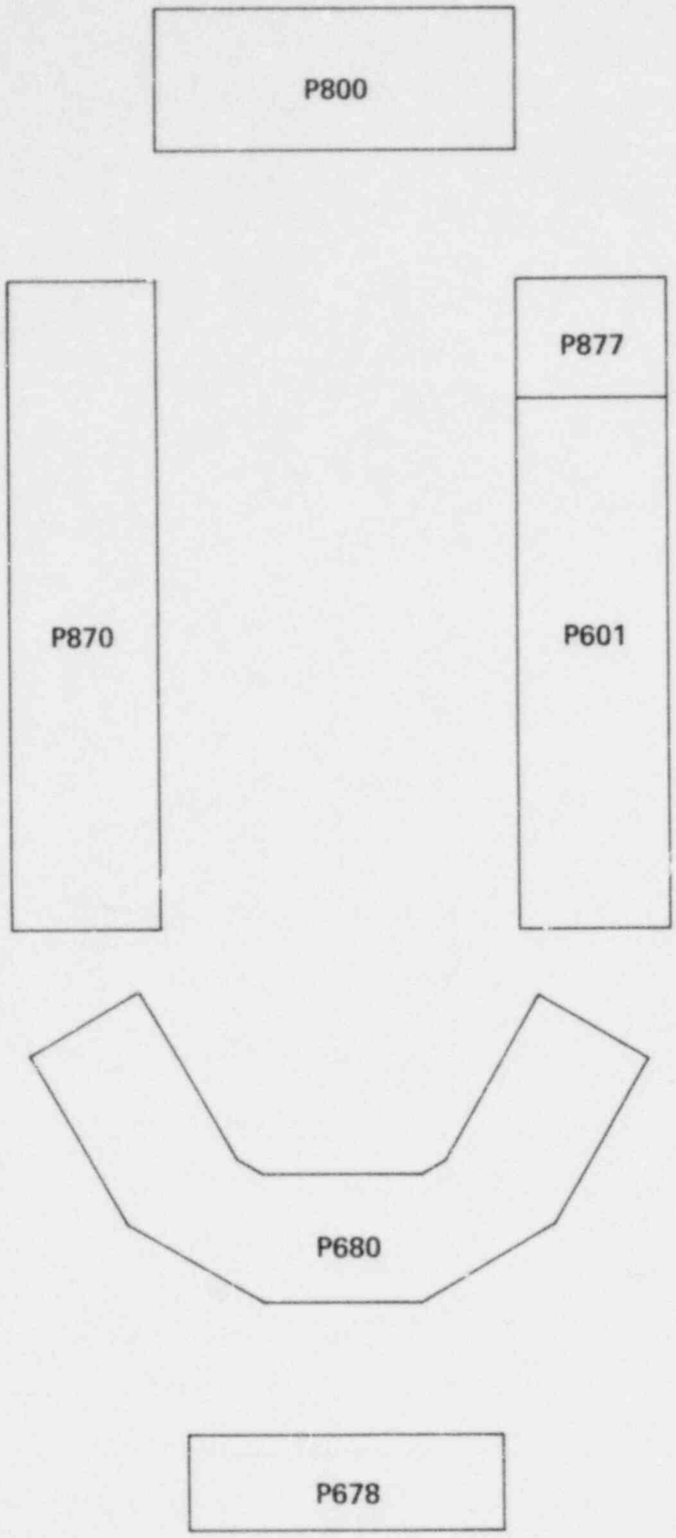
System/Function	Allens Creek	Black Fox	
36) Miscellaneous water systems	back row panel	P870	
37) HVAC	P800	back row panel	59
38) Diesel generators	P877	P877	



AM. NO. 59, (6/81)

HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

ALLENS CREEK CONTROL RM. LAYOUT
(FRONT ROW PANELS)
FIGURE I.A.4.2–1



AM. NO. 59, (6/81)

HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

BLACK FOX CONTROL ROOM LAYOUT
(FRONT ROW PANELS)

FIGURE I.A. 4.2–2

ITEM I.C.9 LONG-TERM PROGRAM PLAN FOR UPGRADING OF PROCEDURES

NUREG 0718 REQUIREMENT

"Applicants shall describe their program plan which is to begin during construction and follow into operation for integrating and expanding current efforts in the area of plant procedures. The scope of the program should include emergency procedures, reliability analysis, human factors engineering, crisis management and operator training. Applicants shall also insure that their program will be coordinated, to the extent possible, with INPO and other industry group efforts. Applicants will submit, prior to the issuance of construction permit eneral discussion of how the requirements will be met. Sufficient d iall be presented to provide reasonable assurance that the requireme. will be implemented properly prior to the issuance of operating licenses."

RESPONSE

The plant staff will begin procedure development approximately three years prior to fuel load. In Section 16.6.8, the following plant procedures are outlined:

Normal Operation

Refueling

Response to Abnormal and Emergency Conditions

Maintenance Procedures 57

Surveillance

Security

Emergency Plan

Radiation Control Procedures

Additional procedures describing the proper interrelationships of the operations, maintenance, technical and training sections and review, approval, and revision of procedures will be developed.

The Plant Operations Review Committee (PORC, formerly PNSRC as described in Section 16.6.6.1 of the PSAR) has responsibilities which include review of procedures and procedure changes thereto determined to affect nuclear safety. The PORC and the Nuclear Safety Review Board (NSRB, formerly CNSRB) will review the Security and Emergency Plans and submit recommended changes to these plans. Procedures will be written to govern the actions of the PORC and the NSRB. Minutes will be kept of all PORC and NSRB meetings.

The PORC will be formed prior to initiating Procedure Development. This committee will exist throughout the life of the plant and meet on a regular basis as will be described in the FSAR. Six months prior to fuel load, plant procedure development should be complete.

The procedure writing/revising program will require four specific evaluations of each ACNGS operating procedure, including Emergency Plan Implementing Procedures and Emergency Operating Procedures. These four areas to be reviewed on each procedure include:

1. Safety Evaluation - determine whether the procedure of activities covered by the procedure constitute an unreviewed safety question or require a Technical Specification change. 57

2. Fire Protection Review - determines need for Fire Hazard Evaluation.

3. Environmental Review - determines need for environmental evaluation.

4. ALARA Review - determines the need for a detailed ALARA evaluation.

HL&P is implementing an organizational element with responsibility for collecting and disseminating the work of many industry groups and other outside organizations (see response to Item I.C.5). This group will thus be one of the primary sources of information for initiating new procedures and revision of existing procedures. The many studies in the areas of emergency procedures reliability analysis, human factors engineering, crisis management, and operator training from the industry and NRC will be collected by this group for evaluation and incorporation in the ACNGS procedures package, as appropriate.

Operating procedures will be written and revised with the assistance of
licensed personnel. Improvements to the procedures may arise from the ACNGS
Training Program, including plant walk through drills. Other improvements may
arise from the ACNGS Plant Staff review of industry studies and industry
efforts on procedures. Major findings regarding reliability analysis will be
reviewed by the ACNGS Plant Staff for modification of the ACNGS Training
Program and applicable results of the reliability analyses and risk assessment
performed for 10CFR50.34(e)(1)(i)/(II.B.8.1) will be used to upgrade
procedures. Human factors studies will be reviewed to determine possible 59
modifications to equipment and HL&P will apply the pertinent results of the
human factors review of the control room, performed for 10CFR50.34(e)(2)(iii)/
(I.D.1) to the development of the operating procedures. Emergency Procedure
improvements will follow closely the efforts of the BWR Owners Group Emergency
Procedures Guidelines. The assistance of INPO and NSAC will be utilized as it 57
becomes available to verify that new methods are incorporated and that plant
procedures and the Emergency Plan Implementing Procedures are consistent with
industry standards. HL&P will document the emergency procedures (for
training) with references that identify the analysis basis or technical basis
that demonstrate conformance to safety requirements of the reactor plant 59
documentation. Experience gained in implementing operating phase procedures

at the South Texas Project Electric Generating Station (STPEGS) is expected to be very valuable to the ACNGS Plant Staff. It is anticipated that the STPEGS Operating Staff will be providing advice and assistance to the ACNGS Staff in development of ACNGS procedures.

The Independent Safety Engineering Review Group will perform additional evaluation of all procedures important to the safe operation of ACNGS for technical adequacy and clarity. Through the combination of all of these reviews, HL&P plans to have well developed, thorough, technically adequate, clear, concise, and safe user oriented procedures. 57

With procedure development beginning approximately 39 months prior to fuel loading, adequate time for interraction exists between the training program and the startup test program, which are scheduled to begin approximately 36 months and 15 months respectively prior to fuel loading.

ITEM I.D.1 CONTROL ROOM DESIGN REVIEWS 59

NUREG 0718 REQUIREMENT

"Applicants shall provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Applicants shall provide a general discussion of their approach to control room designs that comply with human factor principles by specifying the design concept selected and the supporting design bases and criteria. Cosmetic revisions to conventional (1960 technology) designs is unacceptable. Appli-
cants shall also demonstrate that the design concept is technically feasible 57
and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses. Applicants shall commit to control room designs complying to human factors principles prior to issuance of a CP or ML and shall supply design information for approval prior to committing to fabrication or revision of fabricated control room panels and layouts."

RESPONSE

The response to this question is in two parts. The first describes how the NUCLENET control complex was designed utilizing a methodology virtually identical to that set out in Appendix B of NUREG-0659, and how human factors engineering principles were used in balance of plant features of the control complex. HL&P believes that the manner in which the control complex was actually designed meets the intent of NUREG-0659, Appendix B.

The second describes the Allens Creek Nuclenet control complex design review 59
which has already been conducted to identify areas for human engineering enhancements, and notes how the results are now being utilized to make enhancements. HL&P believes that this control complex design review meets the intent of NUREG-0659 and applicable portions (CP status) of NUREG-0700 (to be issued).

The control complex design will be provided to the NRC for review prior to fabrication of the main control boards.

I. SYSTEMS/OPERATING ANALYSIS TECHNIQUES USED IN CONTROL FROM LAYOUT

ACNGS will use the General Electric Company NUCLENET/1000 Control Complex.
NUCLENET as an advanced, generic design, was developed through a methodology 57
virtually identical to that set out in Appendix B to NUREG-0659. This process, similar to the systems/operations analysis process presented in military specification MIL-H-46855, included an analysis of all functions necessary to operate the plant safely, an allocation of functions between operator and machine, and a qualitative verification of the functional allocation.

59

GE assembled a team of experts to design the NUCLENET/1000 Control Complex which included experts in: controls and control systems design, computer
technology, industrial design, operator training, power plant test and opera- 57
tions, and behavioral science.

The premise upon which the design is based is that optimum control is achieved when there is an allocation of control functions between the operator and machine which recognize that each performs certain functions better than the other, and that, once the allocation is made, the design permits efficient and effective manipulation of controls by the operator.

The balance of plant controls were integrated into the NUCLENET complex using human factors principles. A survey was then conducted of a full-scale mockup 57
of the control room with an interdisciplinary team, human engineering discrepancies were identified, and enhancement activities are underway. 59

1. Functional Analysis

1.1 Definition of Objectives

An essential step in the methodology was to define and evaluate the functions and actions to be performed in meeting system objectives. In determining these functions, it was essential to first identify the many activities necessary to operate the plant safely, under normal conditions to produce electricity, those essential for safe operation during transients, and those necessary for safe shutdown during accident conditions.

1.2 Definition of Function 57

Once having identified these activities, the next step was to combine activities under functional groupings, chosen in such a manner that they would be both understandable to the operator and allocated and distributed in such a way to permit effective operator action.

For normal operation the activities were grouped under the following functions: 59

1. Provide and maintain normal core coolant
2. Control reactivity
3. Monitor performance of the core
4. Control reactor pressure
5. Utilize steam for power conversion
6. Convert mechanical power to electrical power

This functional grouping is very similar to those listed in Appendix to NUREG-0659 (p. B-15) which are: 57

1. Nuclear reactor reactivity control
2. Reactor core cooling
3. Reactor coolant systems integrity
4. Primary reactor containment integrity

5. Radioactive effluent control

6. Power generation

7. Power transmission 57

More detailed identification of plant system control functions has been made by considering operational situations and events that will or may confront operators in the Control Complex. The operational situations and events considered consist of: 59

1. All events required to be assessed by Section 15, "Accident Analyses," of the ACNGS PSAR

2. Normal operation of the plant

3. Failures in systems, subsystems, and components, and human errors

4. Anticipated operational occurrences, including startup and shutdown of the plant.

5. (Task Action Plan I.C.1, NUREG-0660 and NUREG-0737)

57

1.3 Decision/Information Requirements

For each significant activity within a functional group, a display was developed which measured each activity against criteria which indicated the type of information essential for making decisions regarding the man-machine interface. This is illustrated on Table I.D.1-1. 59

Human factor engineering design objectives were also developed to reflect the goals to be achieved in a new Control Complex design. These objectives are:

1. Provide a more efficient, coordinated control of the BWR than that attained with a conventional Control Room.

2. Integrate planned operation functions for steam supply and power conversion systems into a single operator station. 57

3. Improve operator response time and reduce operator errors by determining the optimum quantity of data and number of display devices which the operator must continuously survey, analyse and comprehend.

4. Improve operator performance requirements by determining how best to centralize and integrate an optimum number of control devices which the operator must manipulate. 59 57

5. Incorporate efficient hardware and software display techniques in order to present timely, useful information which is meaningful to the operator.

6. Provide for factory testing and evaluation of the entire Control Complex.

1.3.1 Design Criteria

The following design criteria were developed to achieve the design objectives:

1. General

a. Functions in the Control Complex shall be assigned to three types of panels:

(1) Primary Operator interface panels

(2) Secondary Operator interface panels

(3) Back row panels

b. Major power generation systems shall be integrated for planned operations to centralize and minimze the primary Operator interface by: 57

(1) Separating the Operator's short response functions from the long response functions

(2) Making frequently used functions and normal reactivity controls readily accessible from or at the operator's normal duty station

(3) Providing a Display Control System for bringing operational data to the Operator.

c. The planned operating functions of the Core Standby Cooling Systems shall remain integral with the appropriate cooling system, and their direct support systems in order that the design of the benchboard used for Operator interface with these engineered safety features shall not affect licensability of the Control Complex. The other safety systems (e.g, safety related HVAC) are grouped together so that system level actuation and operating status is available on a front row panel (P800), with component level control and all system parameters available on back row panels.

d. Integration of the Nuclear Steam Supply and Balance Of Plant functions shall not degrade the capability for power generation.

1.4 Functional Integration and Interactions

The relationships and interactions between control functions have been defined and evaluated to ensure that all plant operations and safety objectives can be achieved. These relationships and interactions provide a basis for the development of Control Complex design requirements, and, can serve for future design modifications if necessary. 57

1.4.1 Human Factors Application

a. The arrangement of panels shall ensure that each panel defined as primary operator interface will have control, display, and annunciator areas visible to an Operator from his normal duty station.

b. The distance from the operator's normal duty station to the most remotely located function on a primary operator interface panel shall not exceed 30 walking-line feet. 59

c. Normal operations functions shall be placed within the reach span of a single operator without compromising the integrity of those systems having multifunctional capability.

d. Align each system's information devices and controls vertically, with information devices above controls.

e. Align system's operations horizontally, or vertically in the order of the flow path. 57

f. Arrange control functions in an array which is meaningful to the operator. Provide mimic of complex control systems representing the systems process flow, and component orientation as an aid to the operator's job performance. 59

g. Maintain system functional integrity in the human-machine interface to aid operator's comprehension of process behavior.

h. Use miniature devices for controls without sacrificing safety or reliability. 57

i. Provide a Display Control System which presents normal operations information in pre-defined formats, determined by operational analyses, as well as presenting Alarm Initiated Displays (AID). Incorporate: Color and Shape Coding. 57

j. Display by exception, where too much information is not meaningful to the Operator and could cause sensory overload.

k. Provide means for power variation and safe shutdown in the event of catastrophic failure of the Display Control System, yet maximize its availability) ≥ 99.5%. 59

2. Allocation of Functions

A systems analysis was conducted to determine which systems vital to operation of the plant could be controlled from the single operator station designed to be the primary operator interface with the control of the plant. It was determined that these systems were:

(1) REACTOR WATER CLEANUP SYSTEM

(2) CONDENSATE PUMPING SYSTEM

(3) FEEDWATER PUMPING AND REACTOR LEVEL CONTROL SYSTEM

(4) REACTOR RECIRCULATION SYSTEM

(5) ROD CONTROL AND INFORMATION SYSTEM

(6) NEUTRON MONITORING SYSTEM

(7) STEAM BYPASS AND PRESSURE REGULATOR SYSTEM

(8) MAIN TURBINE CONTROL SYSTEM 57

(9) GENERATOR CONTROL SYSTEM

Human factors engineering principles and criteria were used to evaluate human-machine interfaces in analyzing performance requirements for plant control functions and for the allocation of functions to categorize these nine systems. Allocation categories consisted of:

(1) Automatic operation by plant systems equipment

(2) Manual operation by control room Operators and/or plant Technicians

(3) Some combination of (1) and (2)

The design evaluation allocation criteria considered the capabilities and limitations of the Operator(s) and Systems, along with cost-benefit considerations of automating in those instances where the Operator and system could perform a given task approximately equally well. Factors comparable to those listed in Table B-1 of Appendix B to NUREG-0659 were used in making the allocation of functions.

2.1 Operator/Technician Processing Capabilities

Plausible human roles of Operators, Technicians, and Supervisors (e.g., control manipulator, instrument monitor, supervisor, decision maker, communicator, equipment repairer, coordinator) have been defined. Qualitative information processing capability in terms of load, accuracy, rate, and time delay have been prepared for each Operator/Supervisor information processing function.

2.2 System Processing Capabilities

Plausible system roles of Control Complex equipment (automatic control of reactor flux, reactor trip system, engineered safety feature) have been defined. Information processing capabilities and control function response times of control systems equipment have been defined considering load, accuracy, rate, and time delay for processing and response.

2.3 Responsibility for Plant Safety

The overall responsibility for the top-level assessment of plant operating and safety status has been allocated to the human Operator(s). The rationale for this allocation is based on the cognitive abilities of humans, which cannot be duplicated by a machine. The information requirements to exercise this responsibility determine methods for transfer of plant systems data and information to the operator(s) in the Control Complex. 57

2.4 Results of Allocation

A summary description of the allocation of functions follows:

Reactor Water Cleanup System - This system is operated manually. This is an instance where the operator or machine can perform approximately equally well, and the system objective is achieved by manual operation. This operation does not overload the Operator, and it was not cost-beneficial to automate.

Condensate Pumping System - This system is primarily manual. The
operator must reach a decision on when and how much water to pump.
The decision is based on the Operator's ability to observe a wide
variety of stimuli and to reach a judgment based on those observa-
tions. This is an activity in which the Operator is superior to the 57
machine. Once the Operator takes the manual action, other actions in
the system are carried out automatically, such as maintaining the hot
well water level. The automatic operation is best suited to the
machine.

Feedwater Pumping and Reactor Level Control System - During power 59
operation this system operates automatically since its function is to
monitor and perform the routine task of maintaining proper reactor
water level.

Reactor Recirculation System - This system can be operated semi-auto-
matically or manually, and is another example of approximately equal
capability between the operator and the machine to perform a task.
Manual operation, when used, does not overload the operator. 57

Rod Control and Information System - The Operator manually initiates
the action for operation of this system, based on a judgment of when
it should be operated. This judgment is reached after considering a
wide variety of information, a task in which the operator excels.
Once control action is initiated by the operator, the system 59
functions automatically to ensure rods do not exceed established
limits while being withdrawn. This automatic function is ideal for
the machine. The portion of the RCIS which controls Control Rod
sequences and patterns during Startup, Shutdown and power operation
namely the Rod Pattern Control System is not initiated by the
Operator. This system is a hard wired scheme for which the Operator
has limited bypassing ability for a limited number of control rods. 57

Neutron Monitoring System - The Operator must insert and withdraw 59
IRM's and SRM's during startup and shutdown. Also during these 57
phases of operation the Operator must change IRM ranges. Once the
limits within which this system must operate are established by the 59
Operator, the system performs its monitoring and trip functions
automatically. This is a monitoring function in which machines excel.

Main Turbine Control System - This system combines manual and automa-
tic operation. The Operator manually initiates system operation; the
system then operates automatically up to predetermined hold points,
to permit the Operator to monitor the system's performance and reach 57
a judgment on whether automatic operation should be continued to the
next hold point. This system thus combines the most desirable as-
pects of operator and machine control.

Steam Bypass and Pressure Regulator System - The Operator sets the
limits appropriate for the phase of operation, and the system 59
operates automatically within those limits.

Generator Control System - Synchronizing of the unit could be operated approximately equally well either manually or automatically. Manual operation was chosen as preserving the greatest flexibility for integrating balance of plant controls for this system into the Control Complex. 57

Load control is automatically within the limits of the Reactor Recirculation Control.

There are other systems essential to safe operation which are not included in the primary operator interface with plant control, such
as ECCS, the long-term residual heat removal system and most other 59
safety systems. Since NRC requirements dictated that these systems 57
operate automatically, an allocation of functions was not performed
for these systems, except for those used for surveillance testing and 59
manual intervention.

The Combustible Gas Control, Spent Fuel Pool Cooling and Spent Fuel Pool Service Water Systems are the only safety systems which are manually actuated. These systems are simple to operate and their immediate operation is not required. In addition, the control rod drive hydraulic system is manually operated because it is such a simple system that automation is not justified.

The systems on the BOP control benchboard (P870), including Turbine and Generator Lube Oil, Steam Supply and Drains, Circulating Water, Condenser Off-Gas and Condensate and Feedwater Auxiliary Systems, are long-response systems. They are generally manually initiated during startup then operate automatically.

3. Verification of Functional Allocation

The verification of functional allocation is a detailed assessment and analysis of each allocation to ensure that the correct functional allocation has been made. 57 The verification of functional allocation defines the design requirements and specifications for the systems required by the Control Complex as well as the specifications for quantity of operators, for the interface between operators and a system, for the operational procedures (including emergency procedures), and for maintenance requirements.

3.1 Verification of Functions Allocated to Machines

For each system function allocated to a machine, the performance requirements of the system, or equipment to execute the function, have been defined. The performance requirement considers such characteristics as response time, accuracy, reliability, and operator interface or display requirements. Points regarding the design of Control Complex systems are:

1. Display Systems

The design requirements for display systems consider established design criteria. Futhermore, the design requirements for display systems contain criteria to display signals that directly and accurately reflect the information to be transferred to the operator. These signals are to the extent practicable a direct measurement of the desired variable. Displayed parameters are selected from which the operator can determine if the systems are 57
performing their design functions or are responding to operator commands.

2. Control Systems

The design of control systems considers the design criteria presented in Appendix A of 10CFR50: General Design Criteria 20-39. Utilizing this analysis data and the design criteria previously described, primary and secondary Operator Interface panels were defined as:

a. Primary Operator Interface

i	Control Console	(P680) 59
ii	Standby Information Panel	(P678)
iii	Reactor Core Cooling Benchboard	(P601)
iv	Standby Diesel Panel	(P877) 57
v	BOP Control Benchboard	(P870)
vi	BOP Auxiliary Control Benchboard	(P800)
vii	Supervisory Monitoring Console	(P679) 59

b. Secondary Operator Interface 57

All other panels on which are located controls or displays 59
which must be overtly employed by the operator, as opposed to the maintainer.

The design criteria were then applied to the Operator Interface panels.

57

Function Placement

1. NUCLENET Control Console (P680)

a. Normal (after prestart) plant operations functions

b. Short response functions

c. Frequently used and/or reactivity controls

d. Reactor Protections System Operator Interface

Note 1: Only non-divisional systems related to a, b and c above, except Nuclear Steam Supply Shutoff Manual Initiation at System level.

Note 2: Exclude functions not related to above.

2. Standby Information Panel (P678)

a. Support Information of the Display Control System 57

b. No Process Control

3. Reactor Core Cooling Benchboard (P601)

a. NSS Safety systems

b. NSS Long response functions

c. Standard design with no licensing impact

d. Maintained divisional integrity

4. BOP Control Benchboard (P870) 59

   a. BOP Long Response functions 57
   b. Non-frequent use functions
   c. Maintained divisional integrity

5. Standby Diesel - Generator Panel (P877) 59

   a. Safety related diesel generators
   b. Support systems for a. 57
   c. Maintain divisional integrity

6. BOP Auxiliary Control Panel (P800) 59

   a. All other safety systems
   b. BOP long response functions 57
   c. System level monitoring and control of a. and b.
   d. Maintain divisional integrity

7. Supervisory Monitoring Console (P679) (See page 0-50). 59

Table I.D.1-1 illustrates the form for recording the application of these major criteria to each system, leading to a conclusion as to the assignment of functions to the Operator interface panels. Table I.D.1-2 shows the panel assignment conclusions for the BWR process systems which perform the various functional objectives.

With the Systems assignment to panels determined, the next step was to determine the order of placement of those Systems on the panels. Working on each panel individually, applying the design criteria, a logical order of placement of Systems, upon that panel was deduced.

3.2 Verification of Functions Allocated to Humans 57

The most critical portion of the analysis is the verification of functions allocated to humans. Detailed analysis of functions assigned to humans has determined the suitability of the human-machine interface for the performance of the assigned function. Evaluation of the Operator's workload has determined if Operator overload conditions exist. The product resulting from the analysis of functions allocated to humans should determine requirements for:

1. Operator training

2. Operator procedures

3. Optimal Control Complex human-machine interface and control room configuration

4. Control Complex staffing.

Initial Work Station Layout

1. Control Console (P680): 59

"The most critical controls and displays, should be placed in the center of the Operator's work station."

In a Nuclear Power Plant the most critical controls and displays are those which are used to control and monitor the intended performance of the reactor core. In the BWR, these are the Rod Control & Information System, the Reactor Protection System and the Neutron Monitoring System. These were, therefore, placed nearest the center of the Console. 57

There must be water to act as moderator for the fission process and cool the core. In the BWR, steam is generated within the core and, after being scrubbed and dried, carried off to directly drive the Turbine-Generator. 59

With the reactor core at the center of the Console, as the point of reference, if water comes in, and steam goes out, there exists the left to right expectancy of: water into the core; water and steam in the core; and, steam out of the core. 57 59

Therefore, the water system groups were placed on the left side of the Console, and the steam system groups on the right.

The Reactor Water Recirculation System controls reactivity, as a function of flow. It was placed on the left side of the Console, nearest the center. The Condensate Pumping and Feedwater Pumping and Level Control Systems indirectly control reactivity. They were placed next to the Recirculation System. The remaining water system, the Reactor Water Clean-Up System, which bears a functional relationship with another system was placed on the far left side of the Console. (This functional relationship will be explained during the discussion of the other system). 57

The reactor's pressure control is performed by the Steam Bypass and Pressure Regulator System. Pressure directly affects reactivity; therefore this system was placed on the right side of the Console, nearest the center. The Turbine ElectroHydraulic Control (EHC) System controls steam utilization by the Turbine. It was placed next to the Steam Bypass and Pressure Regulator System. The Generator is directly coupled to the Turbine, and was therefore placed next to the Turbine.

There are two more systems which were placed on the Console. One of which had been included in the previous analysis. The Performance Monitoring System is an operational aid which provides the capabilities of:

a. NSS performance calculations, Sequence of Events, Status alarm, and, Post-incident data recall

b. BOP performance calculations and logs

c. Displays of NSS performance calculations results

d. Means of displaying operations information to supervisory personnel

e. Means of generating new display formats, in the field, for both computer systems.

The Performance Monitoring System's Operator Interface was placed on the far right side of the Console.

The other system was the new Display Control System (DCS), so named because it was to be used to provide, per General Design Criteria 1.b.3, (Section 1.3.1) information displays which bring operations data to the Operator. 57

There were ten color CRTs placed on the Console, one to be associated with each of the System groups, and one to be used primarily by the Performance Monitoring System, with switching capability to the DCS. The Operator's controls for the DCS were located on the Console, in a manner to be described later.

2. Reactor Core Cooling Benchboard (P601):

   Order of system placement on P601 was based on the sequence and frequency of operation, as well as the relationship of a particular system to other systems.

Of those system assigned to P601, there is one system which bears a functional relationship with the RWCU. It is the CRD Hydraulic System.

During fueling of the reactor, there are times when it is neither desirable nor practicable to operate the Control Rods. Since the Control Rods are hydraulically operated via controlled leakage carbon seals, when the CRD Hydraulic System is operated, water inventory in the reactor vessel is increased, if not compensated for. One of the functions of the RWCU is to compensate for water level increases, during reactor startup, by providing a controlled drain. When the operator starts up the CRD Hydraulic System after an outage, he must control reactor water level through the RWCU. This functional relationship establishes the need for the CRDHS and the RWCU to be in close proximity to each other, even though on two separate panels.

Hence, the CRD Hydraulic System must be located on P601 at the end closest to the Console, and that end of P601 must be located in close proximity to the left side of the Console. Panel arrangement and Key 57
plan are both anchored by this relationship.

In a nuclear plant, the integrity of the Nuclear Steam Supply is of vital importance. Leakage from both controlled and uncontrolled sources must be monitored to verify the degree of that integrity. Controlled leakage is collected in Equipment Drain Sump(s) before
being pumped to the Clean (low conductivity) Radwaste. Uncontrolled 59
leakage is collected in Floor Drain Sumps before being pumped to the Dirty (high conductivity) Radwaste. The frequency of monitoring and
recording the leakage collected and pumped out to Radwaste dictates 57
that the information should be as close as possible to the Operator. 57
This function is therefore located on P601 next to the CRDHS.

The next most frequently used functions are those of the Main Steam System: Safety/Relief Valves; Main Steam Line Isolation Valves; and, the Steam Line Drains. These functions are located next to the CRDHS and Drain Sumps. The Standby Liquid Control System has very few Operator Interface devices, and, in point of fact, has never been
deliberately operated to inject negative reactivity into the core. 57
The SLC System controls and displays were located next to the Main Steam System.

Core standby cooling is functionally allocated to: the Residual Heat Removal System (RHR); the Low Pressure Core Spray System (LPCS); the Reactor Core Isolation Cooling System (RCIC); and, the High Pressure Core Spray. These systems were assigned to locations on P601 in that
order. 59

3. BOP Control Benchboard (P870):

   Order of system placement on the P870 was also based on the sequence and frequency of operation, as well as the relationship of a particular system to other systems.

During power generation, the in-house electrical loads are usually 57
either totally or partially supplied from the Generator output. Switching of the power source to these Auxiliary Electric Systems normally occurs immediately after the Turbine-Generator is synchronized to the Grid and loaded. There is a relationship therefore between the Generator and the Auxiliary Electric System.

Hence, the Auxiliary Electric System must be located on P870 at the end closest to the Console, and that end of P870 must be located in close proximity to the right side of the Console. As before, panel arrangement and Key plan are both anchored by this relationship. 57

This logical order of placement of systems was continued to locate the remainder of the systems on P870 in the following order: Turbine Test; Turbine-Generator Auxiliaries; Steam Systems, Condensate/ Feedwater; Air Removal System; Off Gas System; Circulating Water System; and Process Radiation Monitoring. 59

BOP Auxiliary Control Panel (P800)

Systems located on Panel P800 are automatic with the exception of the control room emergency filtration inlet selection, as explained in the control room habitability Section 6.4. The panel contains indication to monitor if a system is performing its design function (cognitive task) and controls to start or stop a system (Reg. Guide 1.62) and where required select a mode of operation. The panel has a general layout from left to right

Fuel Pool Service Water 57
Fuel Pool Cooling
Suppression Pool Cooling
Upper Pool Dump to Suppression Pool
HVAC System Level Controls in an arrangement similar to back row HVAC panels P863, P864

Diesel Generator Panel P877

The Diesel Generator Panel contains controls for the Division 1 and 2 standby diesels and support systems such as fuel oil transfer. The operation of the standby diesels is automatic as loss of power or LOCA. This panel contains displays to verify the system is operating according to design such as voltage and frequency. The panel contains controls to synchronize the diesels on to the Auxiliary Electrical Distribution System. The panel is located next to the HPCS diesel which is on one end of P601. This allows the operator to address the entire plant standby power system at one station. 59

Containment Isolation Panel P868, P869 57

The primary purpose of these panels is two fold: 1) To supply the operator with a display by which one can determine that for a given event the portion of the Containment Isolation system required to operate has indeed operated; 2) To supply a location of control for the safety related isolation valves of non-safety systems, such as service water, in a location readily accessible to the operator without violating the requirements of separation between safety related components and non-safety related components (Reg. Guide 1.75).

It can be seen from the above that the back row panels have a support function to the front row panels and contain displays and controls 57
for equipment addressed by the operator on a less frequent basis than front row controls and displays.

Back row Panels

59
Combustible Gas Control Systems located on Panels P871 and P872 are for control of hydrogen that may be generated as a result of an accident. The operator will be required to address this panel anywhere from 30 minutes into an accident to several days into an accident. This is a very long response compared with the other systems. The systems that make up the Combustible Gas Control System are for control, monitoring and recording of the Hydrogen concentrations in the containment.

Other Back row Panels

The safety system back row panels contain controls for individual components with the various BOP safety system. These systems, with the exception of the Combustible Gas Control System, are automatically initiated and operated. The operator addresses the back row panel for testing components or systems or in the case that the operator may wish to operate the particular system in an arrangement different from its normal alignment. The Accident Monitoring Panel and ESF Status Panel contain specifics of the general information displayed on the front row panels, for example the back row 57
Post-Accident Monitoring Panel displays five localized Suppression Pool temperature and Bulk Suppression Pool temperature and the front row Panel displays the same Bulk Suppression Pool temperature.

Front Row/Back Row Interface

The interface betwen the front row and back row panels is different for various modes of operation.

a) During system setup previous to reactor criticality the operator will align service water and service steam systems.

b) Once the reactor has reached criticality the operator has all the controls needed for normal start-up, operation and shutdown on the front row panel group. Testing of the various plant systems can be done from the back row panel during this mode of operation.

c) In the event of an accident as assessed by "Accident Analyses"
Chapter 15 of the ACNGS PSAR or the sequence of failure events 57
for transients and accidents analyzed to develop upgraded emer-
gency procedures no operator initiated control is required for at
least the first 20 minutes. An operator need only monitor that 59
the systems are performing their function. If an accident exists 57
where hydrogen may be generated an operator will go to panels P871 59
and P872 to monitor and initiate systems to control combustible 57
gas control systems. Control of all other BOP safety systems at
the system level (Reg Guide 1.63) is on panel P800. An operator 59
may choose to go to the Post Accident Monitoring Panels P880, 57
P885, P892, P896 to verify a display located on one of the front
row panels. An operator may choose to manipulate controls for 59
safety related HVAC systems which are located on panels P847,
P848, P863, and P864. Manipulation of component controls is not
necessary for the system to perform its safety function. This is
true even in the event of a single random failure.

3.2.1 Subfunction and Task Definition

For each function allocated to humans, all subfunctious and tasks including cognitive tasks that must be performed to achieve the function have been defined and arranged in sequence of performance. Manual tasks are specific with regard to actions and information transfers from system to human required to complete the task. The plant procedures used by the control room Operator/Technicians have been reviewed to determine that they provide adequate guidance to perform the plant control functions according to the allocation of functions. 57

3.2.2 Operator Task Analysis

All requirements for Operator tasks have been analyzed to ensure that they do not exceed human capabilities. All time-critical functions allocated to the Operator have been analyzed to define the time requirements needed to successfully perform each task.

These analyses serve as the basis for specifying the size of the operating crew required, the human performance characteristics required for normal and emergency operations, the operational procedures required for abnormal and emergency operation, and the training requirements for Operators.

Based upon the data just derived, the anthropometric data of the intended user population, and the criteria previously stated, a full-scale mockup of the Primary Operator Interface panels was constructed. Sheet styrofoam was used to form the panels. The front surfaces representing the control and display areas were covered with a material whose texture is compatible with the use of "velcro" fasteners.

Systems analysis had determined, in meeting the systems design objectives, which functions were allocated to the Operator, and which were allocated to the control system. The manner of implementation of those allocations was yet to be tested.

The assumed control and display functional devices, selected for consistency with the design criteria, were photographically reproduced. Small pieces of "velcro" fastener material were adhered to the backs of the devices, to permit their placement (and rearrangement) on the mockup.

The system's Operator Interface Devices were placed on the Console and Benchboards in accordance with the design criteria, and in the same order in which they were selected for location on the panel. The devices were rearranged many times, to provide as nearly as possible, the optimum Operator orientation.

3.2.3 Critical Task Analysis

Operational analyses was then performed, by simulating operation of each system, using system Operating Procedures. The System Operating Procedures used were those in effect in a plant having nearly identical system(s) design. 57

Operational analysis was then performed for integrated plant operation, using the plant procedures. As a result of these analyses, device location and arrangement were more nearly optimized.

A task analysis was conducted for those tasks and modes of operation that are likely to have an adverse effect on plant safety if not accomplished in accordance with system requirements. These tasks are identified as critical tasks. An analysis of critical tasks was done to identify:

1. information required by Operator/Technician, including cues for task initiation

2. information available to Operator/Technician

3. evaluation process

4. decision reached after evaluation

5. action taken

6. body movements required by action taken

7. workspace envelope required by action taken

8. workspace available

9. location and condition of work environment 57

10. frequency and tolerances to action

11. time base and time margins (time margins must be adequate to cover variances in human responses)

12. feedback informing Operator/Technician of the adequacy of the actions taken

13. tools and equipment required

14. number of personnel required, their specialty, and experience 59

15. job aids or references required 57

16. communication required, including types of communication 59

17. special hazards involved

18. Operator interaction where more than one operator is involved

19. operational limits of personnel (performance) 57

20. operational limits of machines and systems

The critical task analysis also included analyzed accident conditions 59

During the operational analyses, careful notation was made of the Operator's information needs for each phase of system operation. This data would be used to select input variables 57 to the Display Control System (DCS), and, to help assign the variables to the various system formats. The immediate use of the data, however, was as a basis for assignment of hardwired, backup information devices to the Standby Information Panel.

1. Standby Information Panel (P678) 57

Until the calculated DCS reliability (≥.995) can be verified 59
operationally, it is necessary to provide sufficient hard-wired information displays (as well as a DCS Configuration/Status Display) to allow continued steady state power operations, reasonable power maneuvers in the Run Mode, or a safe shutdown, without reliance on the DCS. The Standby Information Panel serves no other purpose. There are no process controls or annunciators on the panel. There are no displays which were not determined to be necessary, as a result of the operational analyses.

The Standby Information Panel stands behind the Control Console. Initially, it was intended to be in the direct view of a standing operator. It was later determined that the front silhouette of the Control Console could provide a visual path for the seated operator. The standby information displays for each system controlled from the Control Console were located, accordingly, on the panel.

The Standby Information Panel is located four feet behind the Control Console to allow clearance for CRT removal from, and replacement in, the Control Console, but still maintain the information displays within the visual range of a licensed 57
Operator.

2. Supervisory Monitoring Console (P679)

The Supervisory Monitoring Console allows supervisory personnel access to the same data available to the Operator, without creating a disturbance for the Operator, by looking over his shoulder. The DCS and the PMS have communications links, therefore, all data in the DCS is available to the PMS.

Supervisory personnel wishing to access DCS data may do so on two color CRTs, communicating via a free-standing, multi-function keyboard which is identical, in all but physical appearance, to the keyboard supplied the Operator, for PMS communication.

The Supervisory Monitoring Console is centrally located between, but at the opposite end of, the Benchboards from the Control Console. This provides supervisory personnel with independent visual access to all of the Primary Operator Interface.

3. Display Control System (DCS)

The total design for the DCS required approximately 35 man-years of effort. Some software enhancements continued for almost 7 years after the design initiation.

Display format research and development extended over a period of more than 3 years. As a result of studies performed by General Electric, the DCS formats employ the following color coding:

a. Green - Used only for lines and symbols in process diagrams to represent static system components, i.e., pumps, motors, valves, and, piping which are not dynamically presented in the given format. Selected for this association because the display elements make up the larger part of the display, and, a green hue has been demonstrated to be the least visually fatiguing of the available hues.

b. Cyan - Used as a supporting hue and applied to alphanumeric identification, scales, and borders. 57

c. Yellow - Applied to all dynamic process variable display elements, such as bar graphs and digital data. Selected for this application because of the intensity of its hue. Yellow allows the Operator to scan the display and easily identify dynamic information.

d. Red - Restricted to use as a visual cue for abnormal conditions. Should any variable exceed process limits, the data (bar graph and/or digital) normally displayed in yellow, changes to red. Selected because of the traditional, pre-established psychological associations (populational stereotype) with such conditions, and because intensity allows minimal visual search.

e. White - Used as a reference mark on scales, adjacent to bar graphs, to indicate process limits, or, to present low confidence data.

f. Magenta - May be used in place of red.

g. Dark Blue - Shall not be used, due to its visual loss against the normal background color.

h. Black - Normal background color.

Initial format definition began from the data gathered during the operational analyses. A family of 63 formats was generated for the process System Groups, depicting various levels of each system's operation. Further analysis was performed to determine the relationship of these formats to reactor operations phases.

The DCS design was a continuing process, as stated above. At this point, however, the Operator's controls for retrieval of operational information via the DCS could be defined and located. Each of the ten CRTs, on the Control Console, would have two multi-position selector switches. One switch would serve for System selection and one for Format selection, thus providing capability of displaying any System Format on any CRT. Two momentary push-buttons would provide a Menu Display and Format Change Enable. It is not necessary that the computer system, which drives the displays, attempt to follow Format Selection until the operator has placed the Format Select Switch in the position of the Format desired. The Operator informs the computer that the System and Format selected are those desired for viewing by depressing the Format Change Enable switch. This group of four switches is mounted next to each of the CRTs which they control, including the CRT which is normally assigned to the PMS. Included, for the PMS CRT is a fifth switch (momentary push-button) for assignment of that CRT to the DCS, when necessary. 57 59

One of the positions of the Format Select Switch is designated "Master". When any, or all, of the Format Select Switches are in this position, the Operator has simultaneous control of those CRTs from a "Master Display Select Matrix" located at his left hand, when seated at the center of the Control Console. The informational needs data, derived from the operational analyses, showed what information the Operator needed to either overtly employ, or have available to him, during which phase of plant operation. The Master Display Select Matrix is used, by the operator, to inform the computer which phase of reactor operation he is performing. The computer then displays those System Formats determined to be most meaningful to that phase of operation. Thus the Operator is only required to perform a single action to have appropriate data retrieved, and displayed to him. 57

3.2.4 Work Station Design Analysis 57

For each work station in the control complex, the time 59
sequence of operator activities and the time required for
information exchange or transfer to the operator has been
defined. The analysis verified that the Operator is capable
of completing all tasks and that all tasks are capable of
being performed using the work station design.

3.2.5 Operational Sequence Analysis

An analysis and evaluation of Control Complex sequences of
operations, flow of decisions, physical transmissions of data
and information, receipts of information, storage of informa-
tion, monitoring of systems and interactions among operational 57
crew members, work stations, and systems has been conducted.
The purpose of the analysis was a validation of the Control
Complex capability to successfully complete the intended
functions of the design, in both the time and space domain.

3.2.6 Workload Analysis

A workload analysis for all critical functions was conducted
to appraise the extent of the Control Complex operator work- 59
loads. The analysis was based on the sequential accumulation
of task times. Application of this technique permits an 57
evaluation of the capability of the control complex operator(s) 59
to perform all assigned tasks in the time required to maintain
plant safety.

The detailed workload analysis divided the Operator's tasks
into categories corresponding to perceptual-motor channels
such as vision, left hand, right hand, feet, cognition, audi-
tory, and voice channels. The purpose of this level of detail 57
was to ensure that the Operator is not required to perform
more than one task at a time if two or more tasks require the
simultaneous use of a single perceptual-motor channel nearly
75 percent of the time.

3.2.7 Human-Error Analysis

A human-error analysis was conducted for each perceptual-motor channel workload of 75 percent or greater as defined by the results of the workload analysis.

The purpose of the human-error analysis was to investigate the probability of error during high workload conditions and to evaluate the consequences resulting from these errors.

3.2.8 Work Station Link Analysis

A work station link analysis has been conducted for each work station used by the Operator to perform critical tasks. The analysis defined the frequency and criticality associated with each of the interactions occurring between Operator and equipment and/or between one Operator and another. The defined frequency and criticality of the interactions are then used to 57
evaluate the design adequacy of the work station layout in terms of time and space utilization. This analysis achieves a near optimal design for the work station, such as the spatial correlation of displays with controls to provide the Operator with feedback information as required by General Design Criterion 13, Instrumentation and Control.

SYSTEM VS. CRITERIA

Major Criteria / System	Response Short	Response Long	Frequent Use	Controls Reactivity	Rx Core Coolant	Divisional Yes	Divisional No	Safety Related	Normal Operational (After Prestart)	CONCLUSION REGARDING PANEL ASSIGNMENT P680	P601	P870	OTHER	
Nuc. Boiler Proc. Instr.	x		x		(x)		x		x	x	(x)			
Recirc. Syst.	x		x	x			x		x	x				
CRD Hydraulic		x					x				x			
Rod Control & Information	x		x	x			x		x	x				
FW Lv Control	x		x		x		x		x	x				
SLC		x		x		x		x			x			
Neutron Monit.	x		x			x		x	x	x				
Rx Prot. Sys	x			x		x		x	x	x				57
Rx Press. Contr	x		x	x			x		x	x				
Perf. Mon. Sys		x	x				x		x	x				
RHR		x			x	x		x	x		x			
LPCS		x			x	x		x			x			
HPCS/Power Sup		x			x	x		x			x			
RCIC		x			x	x		x			x			
Press. Relief						x		x			x			
RWCU	x		x		x		x		x	x				
Main Steam		x				x	x	x	x		x			
Condensate Pumping	x		x		x		x		x	x				
FW Pumping	x		x		x		x		x	x				
FW Heaters, Vents & Drains		x					x		x			x		
Condenser Air Removal		x					x		x			x	x	
Turbine - Generator	x		x				x		x	x				
Circ Water		x					x					x		
Service Water		x					x						x	
EP CCW		x					x						x	
Off Gas System		x	x				x		x			x		
Condensate System		x	x				x		x			x		
F.W. System		x	x				x		x			x		59
Radiation Monitoring		x	x				x					x		
Steam System		x	x				x		x			x		
Aux Elec Distribution		x	x			x	x	x				x		
Spent Fuel Pool Clg		x	x			x		x	x				P800	57

TABLE I.D.1-1 (Cont'd)

Major Criteria	Response		Frequent	Controls	Rx	Divisional		Safety	Normal Operational	CONCLUSION REGARDING PANEL ASSIGNMENT			
	Short	Long	Use	Reactiv-ity	Core Coolant	Yes	No	Related	(After Prestart)	P680	P601	P870	OTHER
System													
Supp Pool Clg		x	x			x		x	x				P800
Control Room HVAC		x	x			x		x	x				P800
SGTS		x				x		x					P800
ECCS Area Exhaust		x				x		x					P800
ECCS Room Clgs		x				x		x	x				P800
Upper Pool Dump to S.P.		x				x		x					P800
Switch Gear Area HVAC		x				x		x	x				P800
UHS HVAC		x				x		x	x				P800
Diesel Blg HVAC		x				x		x	x				P800
Diesel Oil Blg HVAC		x				x		x	x				P800
Hydrogen Analyzer		x				x		x					P871& P872
Hydrogen Mixing		x				x		x					P871& P872
Hydrogen Recombiner		x				x		x					P871& P872

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TABLE I.D.1-2

PANEL ASSIGNMENT CONCLUSIONS

Panel	System
NUCLENET* Control Console	Nuc. Boiler Process Instr.
	Recirc
	Rod Control & Information
	FW Level Control
	Neutron Monitoring
	Rx Protection System
	Rx Pressure Control
	Performance Monitoring System 57
	Rx Water CleanUp (RWCU)
	Condensate Pumping
	FW Pumping
	Turbine - Generator
Rx Core Cooling Benchboard	CRD Hydraulic System
	Standby Liquid Control System (SLC)
	Residual Heat Removal (RHR)
	Low Pressure Core Spray (LPCS)
	High Pressure Core Spray & Power Supply (HPCS)
	Rx Core Isolation Cooling (RCIC)
	Pressure Relief valves
	Main Steam System
BOP Control Benchboard	FW Heaters, Vents & Drains 59
	Condenser, Air Removal
	Off Gas System
	Turbine Auxiliaries & Test
	Circ Water
	Condensate System
	Feedwater System
	Radiation Monitoring
	Steam Supply & Drain System
	Auxiliary Electric Distribution
BOP Aux Control Benchboard	Spent Fuel Pool Cooling
	Suppression Pool Cooling 57
	Control Room HVAC
	Standby Gas Treatment System
	ECCS Area Filtered Exhaust
	ECCS Area Fan Coolers
	Upper Pool Dump to Suppression Pool
	All other Safety-Related HVAC Systems
Combustible Gas Control Panel	Drywell Hydrogen Mixing
	Hydrogen Recombiners
	Hydrogen Monitoring System

II CONTROL ROOM DETAILED HUMAN FACTORS REVIEW

BACKGROUND

The Allens Creek Control Room Evaluation Task Team performed a preliminary assessment of the Allens Creek Nuclear Generating Station Unit 1 control room for the purpose of identifying additional human factors engineering design conditions that could provide a basis for further improvements.

Houston Lighting & Power Company's instrumentation and controls engineers assigned to Allens Creek built a full size mockup of the front row panels of the Nuclenet 57
1000 Control Complex. A human factors engineering evaluation was performed on the mockup. The control room was evaluated as is, without regard to planned changes in layout and changes required as a result of Three Mile Island.

The evaluation consisted of the application of human factors engineering design checklists. The checklists were developed and prepared by the General Electric Boiling Water Reactor (BWR) Control Room Owners Group.

As defined by the checklists, the purpose of the control room survey is to review and assess the adequacy of the arrangement and identification of important controls and displays, the usefulness of audio and visual alarm systems, plant status information provided, procedures and training with respect to limitations of existing instrumentation, information recording and recall capability, the control room layout and environment, and other areas of human factors engineering that potentially impact operator effectiveness. The ultimate objective is to identify essential modifications at the operator-control room interface to minimize the potential for human error.

The control room survey methodology as currently recommended by the BWR Control Room Owners Group consists of four phases. The four phases are: (1) control room review utilizing checklists which compares the engineering aspects of the control room with established human factors engineering criteria, (2) operator interviews, (3) LER analyses, and (4) emergency procedures walk-through.

Phase I of the control room survey is conducted by the survey team using checklists which are titled A) Panel Layout and Design, B) Instrumentation and Hardware, C) Annunciators, D) Computers, E) Procedures, F) Control Room Environment,
G) Maintenance and Surveillance, and H) Training and Manning. 59

This essentially agrees with the ten major topics which will be included in Draft NUREG-0700. The majority of the topics in Draft NUREG-0700 are found in BWR Owners Checklist A and B as there are distinct subsections within the BWR Owners Group checklists for the Draft NUREG-0700 topics.

NUREG	BWR Owners Group 57
1) Control Room Workspace	Panel Layout and Design
2) Workplace Environment	Control Room Environment
3) Annunciators and Auditory Signals	Annunciators
4) Controls	Instrumentation and Hardware
5) Visual Displays	Instrumentation and Hardware
6) Panel Layout	Panel Layout and Design
7) Control/Display Integration	Panel Layout and Design Instrumentation and Hardware
8) Labels and Location Aids	Panel Layout and Design
9) Process Computers	Computers
10) Data Recording and Retrieval	Instrumentation and Hardware

Houston Lighting & Power Company's survey team has performed the control room review phase. The checklists that were considered applicable to the Allens Creek mockup were A) Panel Layout and Design, B) Instrumentation and Hardware, and C) Annunciators. The other phases of the survey (2-4) cannot be performed on the Allens Creek control room at this time because they involve operating history and emergency procedure availability.

Each checklist item is presented in the form of a question for consideration by each survey team member. As each specific question is evaluated, the team member actually doing the evaluation of that question indicates the relative degree of compliance (no compliance, somewhat compliant, mostly compliant, full compliance, not applicable). Following each checklist item is space for the person performing the evaluation to enter comments. For each specific checklist item, these comments will identify items or components of non-compliance, the scope of review, or any qualifying statement judged to be appropriate to the evaluation.

Evaluation Approach

The evaluation of the Allens Creek control room mockup was conducted in three phases, described as follows:

- Phase 1 - Documentation Collection
  This phase included collection and review of control room panel drawings, control and display design conventions, human factors engineering reference material (MIL STD 1472B, NUREG CR-1580) board profiles and dimensions, piping diagrams, control panel instrument lists, control wiring diagrams, and flow diagrams.

- Phase 2 - Data Collection 57
  During this phase, checklists containing human factors criteria were applied. These checklists addressed:
  - control room layout
  - panel layout and design
  - annunciators
  - controls
  - instrumentation
  - displays and mimics

- Phase 3 - Analysis and Reporting
  The final phase of the evaluation consisted of (a) review of the data collected, (b) identification of the items needing human factors engineering enhancement, (c) prioritization of the items needing human factors engineering enhancement, and (d) reporting of the results. The items identified as needing human factors engineering enhancement were consolidated into the following groups:
  - annunciators
  - labels
  - panel layout
  - displays
  - instrumentation

The items needing human factors engineering enhancement were prioritized according to the following:

- Category I - corrective action is recommended to minimize the potential for error
- Category II - corrective action should be considered to reduce the potential for error
- Category III - no corrective action is necessary

Results -- Identification of Human Factors Engineering Discrepancies

The results of the control room survey will be objectively evaluated by Houston Lighting & Power Company to determine what changes should be made for minimizing the potential for operator error as a result of human factors engineering considerations. The schedule for this evaluation is such that the changes deemed necessary will be incorporated prior to fabrication of the control room.

The following is a summary of the general human engineering discrepancies that are applicable in some degree to the panels reviewed by the evaluation task team. 57

- Labels

  1) Labels are not consistent in nomenclature and abbreviations.
  2) Labels are not size coded in a hierarchical system for components, major systems, and associated subsystems.
  3) Labels are not consistently positioned.

- Panel Layout

  1) There is a lack of consistency in color coding.
  2) The use of lines of demarcation needs to be expanded.
  3) Mimics need to be added to enhance the system flow path.
  4) Lines of demarcation are needed to separate primary and secondary systems.
  5) Limits on anthropometric design are exceeded.
  6) Layouts are not consistent in operational sequence.

- Displays

  1) Indicators are not scaled in process units that relate to system operation.
  2) There is no normal range or setpoint markings on indicators. There are no markings to show safe or unsafe ranges and expected or unexpected range of operation.

- Instrumentation

  1) There is a lack of consistency in switch coding for pumps, fans, dampers, valves, and breakers.
  2) All switch positions are not clearly marked.
  3) Switches for emergency or abnormal use are not clearly identifiable.
  4) Keylock switches require use of keys for normal operation. The key is the switch handle.
  5) Backup indication is not easily correlated with indicators.

- Annunciators

    1) Abbreviations used on annunciator windows are inconsistent with the abbreviations used on instrument nameplates.
    2) Alarm prioritization does not exist for balance of plant systems.

Results -- Enhancements to Resolve Human Factors Engineering Discrepancies

Based on the preliminary assessment of the Allens Creek control room, the following enhancements will be made to resolve the human factors engineering discrepancies:

1) A standard list of abbreviations is being developed to include instrumentation on the panels, annunciators, and the computer system.
2) Hierarchical labeling will be implemented.
3) Specific guidelines are being developed for labels and will be implemented. This will include label consistency, accuracy, and board placement relative to labeled components.
4) Lines of demarcation will be added to functional control and display groups.
5) Mimics and lines of demarcation will be added to improve flow paths.
6) Panel layout will be improved in terms of control/display relationships, operational sequence, functional relationships, and anthropometric standards.
7) All switch positions will be identified, and switch coding will be consistent throughout the control room.
8) Instruments required for emergency and abnormal conditions will be identified and clearly marked.
9) A color coding standard is being developed for indicating lights, mimics, and computer system displays. 57
10) The annunciator system will be improved in terms of prioritization using color coded windows and proximity to associated controls and displays.
11) Indicators will be scaled in process units relating to system operation, and ranges of operation will be provided.

Control Room Evaluation Task Team

The Allens Creek Control Room Evaluation Task Team consisted of two licensed operators, five instrumentation and controls engineers, and a human factors engineering consultant.

The team was broken into four groups for the purpose of the survey. A minimum of two groups evaluated each of the six front row panels. Although each person filled out his own checklist, the two people on each group were not required to agree on the assessment of the panel.

Conclusion

The Allens Creek control room uses advanced technology which incorporates human factors engineering principles in design and operation. A control room design review has been conducted based on a full-scale mockup, deficiencies identified, and enhancement activities are being undertaken. Completion of the enhancements will result in a control room which meets the intent of NUREG/CR-1580 and NUREG-0659. Results of these actions will be forwarded to NRC upon completion. 57

ITEM NO. I.D.2 PLANT SAFETY PARAMETER DISPLAY CONSOLE

NUREG 0718 REQUIREMENT

"Applicants shall describe how they intend to meet the staff criteria contained in NUREG-0696 for the plant safety parameter display console. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses." 57

RESPONSE

A Safety Parameter Display System (SPDS) will be provided, and is described in Section 7.5.1.6. The system will have the capability of displaying the full range of important plant parameters and data trends on demand. The system will also indicate when plant parameters are approaching or exceeding process limits. The SPDS will be designed in conformance with the guidance of NUREG-0696, February 1981, Section 5. 59

The design concept of the SPDS is known to be technically feasible and within the state-of-the-art. HL&P has no questions or concerns as to the ability to implement the SPDS design prior to OL issuance. The SPDs will be a computer-based system of high quality and reliability. It will be capable of functioning properly in the environments that are present during transient and accident conditions. 57 59

ITEM I.D.3 SAFETY SYSTEM STATUS MONITORING

NUREG 0718 REQUIREMENT

"Applicants shall describe how their design conforms to Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, appli- 57
cants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses."

RESPONSE

Regulatory Guide 1.47 was a pre-TMI design basis for ACNGS, and the design meets the guide without any exceptions. The design includes automatic indication of the bypassed and operable status of safety systems. To the 59
extent practical, inputs to the Safety System Status Monitoring system will be direct measurements of the desired variables. Details are given in Section 7.2.2.2.2.2, and systems covered by the guide are shown on Table 7.1-2. 57

ITEM II.B.1 REACTOR COOLANT SYSTEM VENTS

NUREG 0718 REQUIREMENT

"Applicants shall modify their plant designs as necessary to provide high point reactor coolant system and reactor vessel head vents that can be remotely operated from the control room. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to 57
meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses."

RESPONSE

Venting capability of the ACNGS reactor vessel is addressed in two parts (refer to Figure 5.1-3a):

1. Up to the main steam line nozzles: The presence of non-condensible gases in the vessel below the main steam line nozzles could interfere with continued core cooling, so the capability for venting this region is essential. This can be accomplished by opening any one of the 19 safety relief valves on the main steam lines (which may be open already depending on the mode of core shutdown cooling in use). These valves and their operators are safety grade, seismically and environmentally qualified for accident conditions, have positive position indication in the control room (see Item II.D.3), and are powered from the onsite electrical system 59
and operable from the control room. Eight of the valves have a safety
related air supply, providing redundant venting capability. 57

   In addition, this region of the vessel can be vented through the RCIC steam supply line which connects to main steam line A, without opening 59
the SRV's. This path is through the RCIC steam turbine exhaust, which discharges to the suppression pool.

2. Above the main steam nozzles: The presence of non-condensible gases in the vessel above the main steam line nozzles will not interfere with continued core cooling, and as such venting this region of the vessel is 57
not considered to be a safety concern. Even so, there are two means of venting this space:

   a. Normally open 2" reactor head vent line and valve B21-F005, which 59
discharges to main steam line A (which can be vented to the suppression pool/containment via any one of three safety relief 57
valves).

   b. Normally closed 2" reactor head vent line and series valves B21-F001 59
and B21-F002, which discharges to the drywell high purity drain tank.

   These valves are safety grade and their operators are Class 1E, 57
seismically and environmentally qualified, but are not powered from the onsite electrical system. They are operable from the Main Control Room.

Consideration has also been to the potential for the accumulation of non-condensible gases interfering with the operation of the ECCS. In the post-LOCA condition, it is possible for non-condensible gases to come out of solution while operating the RHR system. It is expected that these gases would be swept through the RHR system, but some gases could potentially accumulate in the upper portions of the RHR heat exchanger during the steam condensing mode of RCIC operation should substantial amount of non-condensibles be generated. The upper portion of the RHR heat exchangers are provided with separate 1" vent lines to the suppression pool for the removal of non-condensible gases. The isolation valves on these lines are Class 1E, and are operable from the Main Control Room. Procedures for the use of these lines will be summarized in the FSAR. 59

All of the above venting paths lead to the containment via the suppression pool, which is the basis for hydrogen mixing analyses. The control of large amounts of hydrogen in containment is discussed in the response to Item II.B.8.3.

The above supplements the PSAR information on capability for RCS venting, and is consistent with preliminary design information normally required at the CP stage of review. There is no new, novel design, and there are no concerns regarding technical feasibility, state-of-the-art or ability to implement the intended RCS venting design. 57

ITEM II.B.2 PLANT SHIELDING TO PROVIDE ACCESS TO VITAL AREAS AND PROTECT SAFETY EQUIPMENT FOR POST-ACCIDENT OPERATION

NUREG 0718 REQUIREMENT

"Applicants shall (1) perform radiation and shielding design reviews of spaces around systems that may contain highly radioactive fluids and (2) implement plant design modifications necessary to permit adequate access to vital areas and protect safety equipment. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses." 57

RESPONSE

A post-accident radiation shielding review is being conducted for ACNGS, and is scheduled for completion in December, 1981. Its purpose is to ensure that:

1. access and required occupancy is possible to areas where plant personnel must perform post-accident functions, such as the control room, onsite technical support center, sampling station, sample radiochemical analysis laboratory, etc., such that the GDC-19 specified dose design basis will not be exceeded. 59

2. radiation levels for which safety-related equipment is qualified to function post-accident are not exceeded. 57

The basic assumptions of the study are:

1. 100% of the core inventory of noble gases and 25% of the halogens are dispersed into the drywell and containment air volumes. 50% of the halogens and 1% of all other fission products (except noble gases) are dispersed into the reactor and suppression pool water. The core inventory of isotopes available for release is given in Table II.B.2-1. Instantaneous release and mixing are assumed. 59

2. The following systems are instantaneously filled with contaminated fluid per 1 above: 57

   a. Residual Heat Removal (Suppression Pool Cooling, Containment Spray and Low Pressure Coolant Injection).

   b. High Pressure Core Spray

   c. Low Pressure Core Spray

   d. Reactor Core Isolation Cooling

e. MSIV Leakage Control

f. Standby Gas Treatment

g. Hydrogen Analyzer 57

h. Containment Gas Sampling

i. Post-Accident Liquid Sampling

j. ECCS Filtered Exhaust

k. Drywell and Containment

Systems listed in NUREG-0737, Item II.B.2, Section (2) which are not considered as sources terms are listed as follows with justification:

a. Hydrogen Recombiner System: ACNGS utilizes thermal recombiners which are completely internal to the containment. 59

b. Gaseous and Liquid Radwaste Systems: The radwaste systems are isolated from the containment and other systems which may contain primary coolant after an accident.

c. Chemical and Volume Control System: This is a PWR system.

The study will verify the adequacy of the existing design and indicate where changes will need to be made. If changes are required to meet acceptable operator and/or equipment dose levels in certain locations, the following options are available: 57

1. move the offending radiation source to a less sensitive location

2. move the target equipment or operator control/work station to a location with an acceptable radiation field.

3. place additional shielding around the offending radiation source.

4. place local shielding around the target equipment or operator control/work station.

5. purchase equipment designed to withstand the newly specified radiation environment.

In selecting the option to be used emphasis will be placed on minimizing building structural modifications, since the buildings potentially affected are mostly designed and are early in the construction sequence.

If problems are encountered as a result of the shielding analysis, they are expected to be of a physical or design detail nature rather than questions of technical feasibility or state-of-the-art. Since the shielding study will be finished in 12/81 and a CP for ACNGS is not expected before then, HL&P is assured that any necessary changes can be effectively implemented prior to construction, and that the specifications of NUREG 0737, Section II.B.2 will be implemented prior to OL issuance. 59

TABLE II.B.2-1

CORE INVENTORY OF ISOTOPES FOR ACNGS UNIT #1 AT SHUTDOWN

Isotope	Inventory in Ci	
I 131	1.05×10^{8}	
I 132	1.49×10^{8}	
I 133	1.72×10^{8}	
I 134	2.22×10^{8}	
I 135	1.76×10^{8}	
Br 82	1.39×10^{5}	
Br 83	1.07×10^{7}	
Br 84	1.68×10^{7}	
Br 85	2.33×10^{7}	
Xe 133*	6.99×10^{6}	
Xe 133	2.02×10^{8}	
Xe 135*	6.08×10^{7}	59
Xe 135	3.50×10^{7}	
Xe 137	1.68×10^{8}	
Xe 138	1.59×10^{8}	
Xe 139	1.41×10^{8}	
Xe 140	6.93×10^{7}	
Xe 141	2.11×10^{7}	
Kr 83*	1.08×10^{7}	
Kr 85*	2.32×10^{7}	
Kr 85	1.07×10^{6}	
Kr 87	4.19×10^{7}	
Kr 88	6.05×10^{3}	
Kr 89	7.13×10^{7}	
Kr 90	7.31×10^{7}	
Kr 91	4.71×10^{7}	
Cs 134	6.96×10^{6}	
Cs 136	3.21×10^{6}	
Cs 137	1.28×10^{7}	
Cs 138	1.85×10^{8}	
Rb 88	6.11×10^{7}	
Rb 89	7.82×10^{7}	
Sr 89	8.05×10^{7}	
Sr 90	8.82×10^{6}	
Sr 91	1.03×10^{8}	
Sr 92	1.11×10^{8}	
Y 90	9.27×10^{6}	
Y 91	1.10×10^{8}	

TABLE II.B.2-1 (Cont'd)

Isotope	Inventory in Ci
Zr 95	1.67×10^8
Zr 97	1.65×10^8
Nb 95	1.69×10^8
Mo 99	1.81×10^8
Ru 103	1.46×10^8
Ru 106	6.83×10^7
Ag 110*	9.72×10^4
Te 129*	4.65×10^6
Te 129	2.81×10^7
Te 132	1.49×10^8
Te 134	1.69×10^8
Ba 139	1.78×10^8
Ba 140	1.69×10^8
Ba 141	1.74×10^8
Ba 142	1.40×10^8
La 140	1.80×10^8
Ce 141	1.77×10^8
Ce 143	1.49×10^8
Ce 144	1.26×10^8
Pr 143	1.49×10^8
Nd 147	6.01×10^7
Tc 99*	1.59×10^8
Tc 101	1.70×10^8

59

ITEM II.B.3 POST-ACCIDENT SAMPLING

NUREG 0718 REQUIREMENT

"Applicants shall (1) review the reactor coolant and containment atmosphere sampling system designs and the radiological spectrum and chemical analysis facility designs, and (2) modify their plant designs as necessary to meet the requirements. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the
requirements by specifying the design concept selected and the supporting 57
design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses."

RESPONSE

The capability for post-accident sampling of reactor coolant and the containment atmosphere, along with onsite analysis capability, will be
provided, consistent with the requirements of NUREG 0737, II.B.3. Details are 59
as follows:

A. Sample Collection 57

1. Liquid: The capability to collect liquid samples from the reactor
coolant system (sample location for the RCS is shown on Figure 59
II.B.3-1) and suppression pool will be provided. The length of the sample lines will be as short as possible to minimize plateout.
Sample collection will not require an isolated auxiliary system to be 57
placed in operation. The sampling operation under post-accident conditions utilizing Regulatory Guide 1.3 source terms will not
result in a personnel dose of greater than the dose specified in 59
GDC19 (5 rem whole body, 75 rem extremeties). See Item II.B.2. The
sample can be collected within one hour of the request for a sample. 57
The sample return line will be to the suppression pool. Provisions
will be incorporated to minimize radioactive release and the spread 59
of contamination from the sample station.

2. Gaseous: The capability to collect containment atmosphere samples will be provided through the containment/drywell H_2 sampling system described in Section 7.5.1. The same dose to workers criteria and
sampling time as for liquid sampling will be met for the containment 57
atmosphere sampling operation. The sample return line will be to the containment atmosphere.

B. Sample Analysis

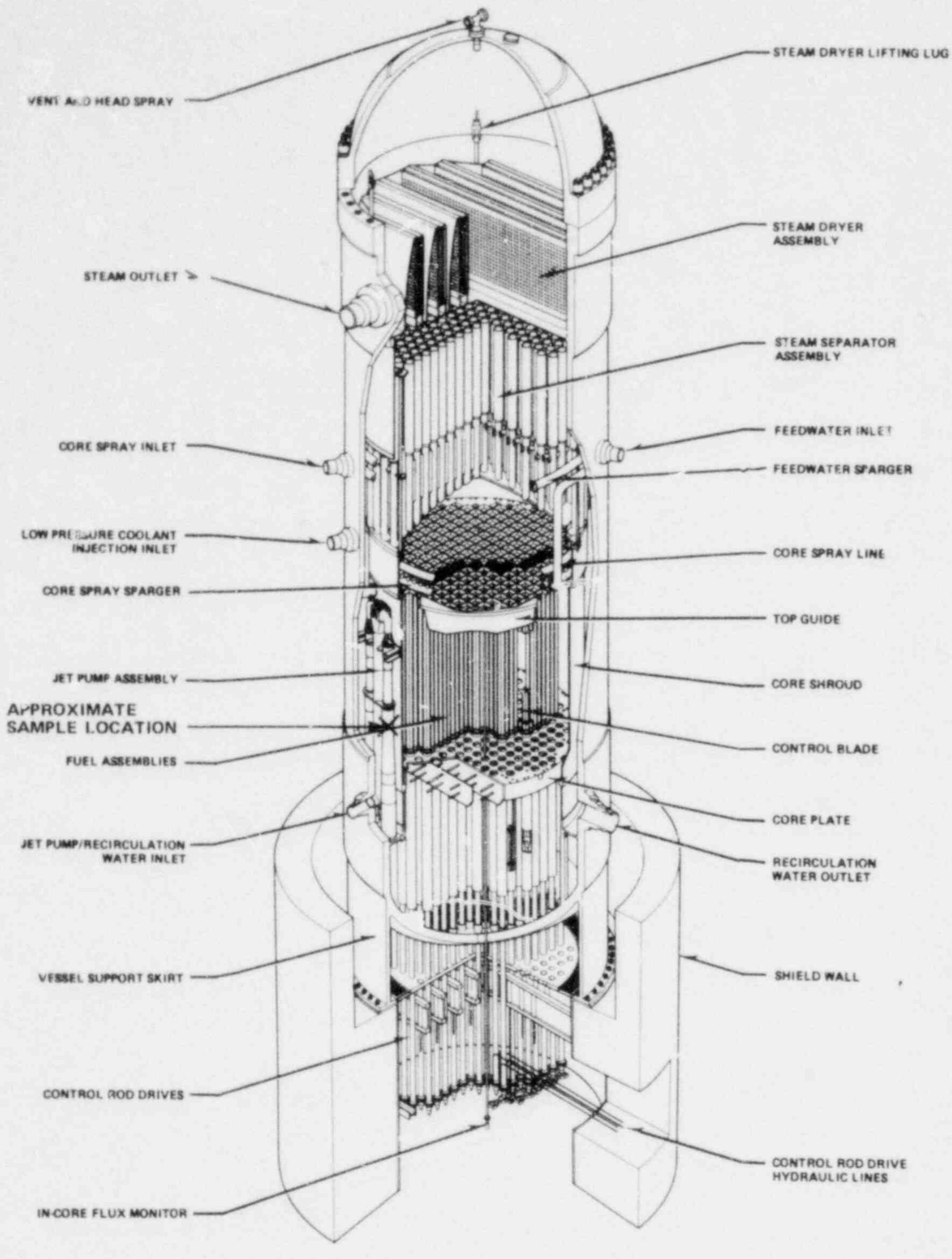
Analysis of the post-accident samples collected per A above will be in the Personnel Access Building which will be designed and shielded such
that the required analyses can be performed without interference from 59
external radiation sources. Radiological analyses for certain radionuclides that are indicators of core damage (e.g. noble gases,
iodines and cesium and non-volatile isotopes) will be performed in the 57

counting room and chemical analyses in the laboratory. The chemical sample analysis stations are equipped with fume hoods which are exhausted to the outside through HEPA filters. Doses to workers involved in sample analysis will not exceed those specified for the sample collection and transportation operation. Time for the sample analyses will not exceed the following: 59 57

- radiological: two hours
- boron: two hours, if boron injection was initiated 59
- chlorides: twenty-four hours 57
- total dissolved gas or hydrogen: two hours
- dissolved oxygen: verification that dissolved oxygen is $<$ 0.1 ppm if chloride concentration exceeds 0.15 ppm 59

Accuracy, range and sensitivity will be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant system. 57

There are no questions regarding technical feasibility or state-of-the-art regarding the post-accident sampling capability, nor are there any concerns as to the ability to implement the design prior to OL issuance.



AM. NO. 59 6/1/81 (D) - DESIGN

HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

LOCATION OF POST-ACCIDENT
RCS SAMPLE POINT
FIGURE II.B.3 – 1

ITEM II.B.8 RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

NUREG 0718 REQUIREMENT

"Applicant shall:

(3) provide a system for hydrogen control capable of handling hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction."

RESPONSE

HL&P commits to provide a hydrogen control system. Currently a number of different methods are being considered throughout the industry and it is expected 57
that these efforts will, in the future, produce valuable data upon which to select an optimum means of hydrogen control. Further, it is expected that the pending rulemaking on degraded cores will determine the necessity for such a system. For the purposes of meeting the stated requirement, a post-accident inerting system using CO_2 as an inerting agent is proposed. However, for the reasons given above the basic design and need for this system will be under continuing review.

The post-accident inerting system is to be capable of handling the hydrogen generated by the equivalent of oxidation of 100% of the active fuel cladding. This system assures that containment integrity is not endangered due to the combustion of hydrogen generated by the reaction between the fuel cladding and the reactor coolant. Control of hydrogen combustion will be accomplished by the injection of carbon dioxide into the containment after event initiation, but before significant hydrogen transport to the containment. The carbon dioxide concentration will be sufficient to render all mixtures of hydrogen and air inert, and incapable of sustaining combustion.

The following criteria will be used to design the Post-Accident Inerting System:

(a) The hydrogen from a transient resulting from the reaction of up to 100 percent of the active fuel cladding with the reactor coolant is assumed to start evolving from the suppression pool surface in no less than 45 minutes from the reactor scram. Analyses will be performed and submitted to the NRC for review two years from construction permit issuance in order to demonstrate that the 45 minute period for hydrogen evolution from the 59
suppression pool encompasses a majority of those sequences that are major contributors to risk (as identified in the PRA performed as described in App. 15B) in ACNGS (including at least analysis of the most probable small break accident).

(b) The system will have the capability for providing adequate mixing of the carbon dioxide with the containment atmosphere to assure the prevention of 57
the combustion of hydrogen in the containment atmosphere.

(c) The system will have provisions for both long term sampling of carbon dioxide and oxygen concentrations, and the addition of more carbon dioxide as necessary.

(d) Alternating current power will not be required for the system to perform its inerting function.

(e) The system will be single active failure proof, for either intended or inadvertent operation.

(f) Inadvertent full inerting, excluding seismic and design basis and other loadings, will not produce stresses in the steel containment in excess of the limits set forth in II.B.8(4)(d).

(g) Inadvertent full inerting of the containment will not effect the safe shut down of the plant.

(h) The system will be protected from tornado and external missile hazards.

(i) The quantity of carbon dioxide to be injected will be limited to the amount required to provide 61 volume percent carbon dioxide concentration within the drywell and containment for the accident condition, plus the additional allowance for carbon dioxide solubility in water. 57

(j) The containment isolation valves associated with the system will be classified as intermediate with no system integrity isolation function.

(k) Design and operation of the system will be such that buildup of ice on the nozzles, valves or in the lines will be prevented, and the isolation valves operability following injection will be assured. 59

Functional Description

The Post-Accident Inerting System (PAIS) will be designed so that the PAIS components located inside the containment can withstand conditions produced by the PAIS design basis transient.

The inerting system will be initiated manually. The system will annunciate audibly and visually in the main control room and by special tone alarm in the containment. A time delay will be provided for containment evacuation, and continued restoration of water level, thus preventing unnecessary operation. A second audible and visual annunciation in the main control room plus special tone containment alarm will be given to confirm system operation based on the detection of carbon dioxide flow in the main header. The system will be designed so that an inert condition is reached after a 15 minute discharge time. 57

In general, maintenance on equipment of significant importance to safety are not undertaken during plant operation. If such maintenance is performed, it will be as allowed by the Technical Specifications and under technical specialist supervision. The ability of personnel to complete such tasks under an inerting situation or following many other possible conditions is considered in developing the Technical Specifications. 59

The charge of carbon dioxide will be stored in liquid phase in three (3) storage tanks outside the Reactor Containment Building in a new building located at grade elevation. Each tank will contain one-third of the necessary 57 charge. A fourth tank will provide any necessary makeup. This fourth tank will be under separate manual control. Each of the tanks is equipped with fill connections, relief valves, level gauges, and pressure and temperature gauges. The building will require electrical lighting, heating and ventilating. Low temperature storage is maintained by redundant refrigeration compressors and coils for each tank, maintaining the tank pressure between approximately 290 and 305 psig at $0^{o}F$. Under emergency conditions when the mechanical refrigeration is not functioning, the unit is self-refrigerating due to the cooling effect of small amounts of vapor released through a bleeder valve. The system can remain in this self-refrigeration mode for at least 24 hours without losing full functional capability.

Redundant, back seated, pressure actuated discharge valves on each tank will feed the carbon dioxide to the main header. One containment penetration with redundant isolation valves on both the inboard and outboard sides (see Figure 1) will feed the flow to an interior network of nozzles which assure adequate carbon dioxide distribution. The discharge and isolation valves will close on low header pressure after the injection process is complete. Under normal operation ac power is required to maintain refrigeration, the battery charging
system, HVAC, and electrical lighting. Direct current powered main control 57
panel(s) located in the area of the tanks will be equipped with selfcontained battery units for valve actuation. No alternating current power is required for the system to perform its inerting function.

The system will be designed to deliver approximately 135 tons of carbon dioxide in less than 15 minutes. After this time, periodic sampling and analysis of the containment atmosphere will be performed to detect any reduction in carbon dioxide concentration or any increase in oxygen fraction.

The design for the discharge and isolation valves assures a high probability that no inadvertent dump will occur. The system will require two active failures to inadvertently inject into the containment. Procedures for containment purging and carbon dioxide cleanup following a degraded core event or inadvertent injection will be developed later.

An investigation for the potential of and possible consequence due to the decomposition of carbon dioxide (CO_2) to carbon monoxide (CO) in degraded
core conditions will be conducted and if applicable factored into the design. 59

A conceptual arrangement of the system is shown in Figure II.B.8.3-1.

<u>Design Evaluation</u> 57

Significant core heating and hydrogen generation in a BWR does not start until the water level falls below bottom of the active fuel. This is due to cooling of the core by steam generated in the lower portion of the vessel.

The start time of hydrogen evolution into the containment for transients is at least 45 minutes from event initiation, including the hydrogen transport from the core, through the suppression pool and into the containment. The contribution of hydrogen from radiolysis of water and corrosion of exposed metal surfaces is insignificant, and does not affect inerting system design.

The estimate of the time for the start of hydrogen evolution from the suppression pool is based on the analysis of core uncovery, core heat up and metal-water reaction in postulated transient events. An isolation transient event with the loss of all coolant make up was analyzed and the start of significant hydrogen generation in the core was estimated to occur at 45 minutes. The evolution of hydrogen from the suppression pool would occur later than 45 minutes. This estimate of time for hydrogen evolution is expected to bound most transient events, with partial or complete loss of coolant makeup. These transients represent a large fraction of the postulated 59
loss of coolant events which could potentially lead to core damage.

A concentration of 61% carbon dioxide by volume in air will render hydrogen-air mixtures nonflammable regardless of hydrogen concentration.[1] This quantity of carbon dioxide plus an allowance for potential for loss of the inerting agent, will provide an adequate margin to ensure an inert 57
atmosphere.

HL&P will carry out a program to determine that sufficient CO_2 will be present in the containment to preclude combustion of hydrogen considering the various sources of random ignition including venting of hydrogen from the 59
reactor vessel at elevated temperatures (above autoignition temperature).

Mixing is primarily a function of bulk air movement. Very rapid air movement is caused by the flashing and expansion of the liquid carbon dioxide to a gas and by the discharge of this expanding liquid at an elevated velocity. Nozzle 57
design and selection maximizes this air movement. Nozzle placement is based on attaining both global and local air movement within the containment and on protecting enclosed spaces not within the general injection patterns established. Nozzle placement to assure adequate mixing will be based on extensive experimental data and field experience existing in the fire protection industry. Data will be provided which demonstrates the rapidity and thoroughness of the mixing process provided by the techniques and equipment to be employed in the design. 59

Nozzle Location and Piping Layout

Containment: Based on studies to date a total of 46 nozzles will be located in the containment. To obtain the maximum degree of mixing and recirculation, 57
20 nozzles (single axial orifice) have been located at approximate elevation 272'-0" (above polar crane) along the outer periphery of the wetwell. Of these 20 nozzles, two groups of five nozzles each will be positioned to discharge downward into the containment annulus. In addition, two groups of five nozzles each will be positioned to discharge upward toward the center of the containment dome. These nozzles deliver 82 percent of the total carbon dioxide discharge rate required in the containment.

The remaining carbon dioxide discharge required in the containment will be distributed to 20 individual rooms and below six floor levels to provide assurance of local mixing and inerting. The rates into each room were based on their individual volumes; rates for nozzles located below floor levels will be based on the volumes which existed below those floor areas at that elevation. Further studies may result in modification of the quantity and/or locations of nozzles.

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1 Coward, H.F. and Jones, G.W. "Limits of Flammability of Gases and Vapors" Bureau of Mines Bulletin #503 (1952).

Drywell: A total of 5 nozzles will be located in the drywell. Four nozzles will be positioned near the ceiling to discharge horizontally. One nozzle will be positioned to discharge into the drywell head above the reactor.

The distribution and rate of injection of the agent is calculated using equations based on two-phase fluid flow. This calculation procedure conforms to the method given by NFPA-12, the National Fire Protection Association standard on rbon dioxide extinguishing systems, and is accepted by Underwriters Laboratories and by the Factory Mutual Insurance Corporation and has been validated in test discharges throughout the fire protection industry. 57

System Initiation

(1) A key locked switch is actuated by the operator to open the discharge valves on each of the three (3) main tanks. Parameters which the operator would use to determine when to initiate include reactor water level, ECCS function and reactor pressure. Formal criteria will be defined in the emergency procedures developed for ACNGS.

(2) A timer then actuates on high pressure in the main header, and upon reaching zero time, enables the operator to open the isolation valves via a second key-locked switch.

(3) After discharge, the isolation and discharge valves close on low header pressure.

The development of the initiation signals, logic and backup signals is under continuing development.

The guidelines and signals chosen to form the procedure for manual initiation of the PAIS must balance the necessity for completing the injection of CO_2 before a flammable condition is reached in the containment, and avoiding an unnecessary injection. 59

Reactor water level is considered to be the best indication of adequate core cooling in a BWR and the best indication of the potential for rapid hydrogen generation from metal-water reaction.

Two procedures have been proposed: In the first, a reactor level 1 sustained for (X) minutes signals the operator to open the PAIS discharge valves. Main CO_2 header pressure actuates a containment evacuation alarm and starts a timer set for (Y) minutes. While the timer is operating, the operator continues to ascertain the functioning of ECCS and other water makeup sources to the reactor vessel. If the operator determines that there is inadequate reactor makeup, he then operates four key-locked switches which open the isolation valves as soon as the timer times out and clears the electrical interlock. The isolation valves close automatically on low CO_2 header pressure (which indicates that CO_2 has been discharged).

The second procedure utilizes a lower reactor level setpoint, sensed inside the reactor core shroud at about the core mid-plane. When this level is reached the operator is signaled without delay to open the PAIS discharge valves. As in the first procedure, the main CO_2 header pressure actuates a containment evacuation alarm and starts a timer set for (Z) minutes. When the timer times out and clears the electrical interlock, the operator actuates the four key-locked switches which open the isolation valves. The isolation valves close on low header pressure.

In the first procedure, ECCS or other water makeup sources will restore reactor water level in most cases prior to the expiration of the reactor level time delay. In those cases where the water level is not promptly restored in the reactor vessel-core shroud annulus (where reactor level 1 is sensed), such as for the break of the large recirculation piping, adequate core cooling is indicated if there is minimum ECCS function. However, this procedure is more complicated than the second, and requires more action by the operator. 59

The second procedure is less complicated and requires minimum operator attention, while reducing time available for CO_2 injection.

As a condition for entering either of these procedures, the operator will wait (later) minutes after the start of a significant event to allow him to devote his full attention to reactor emergency core cooling. The procedures described above are designed to minimize the potential for activation during non-degraded core conditions.

Analyses will be performed and submitted to the NRC for review two years from construction permit issuance in order to demonstrate that unique initiation information can be developed which gives reasonable assurance that the system will be actuated when needed and not actuated when not needed.

Testing and Inspection

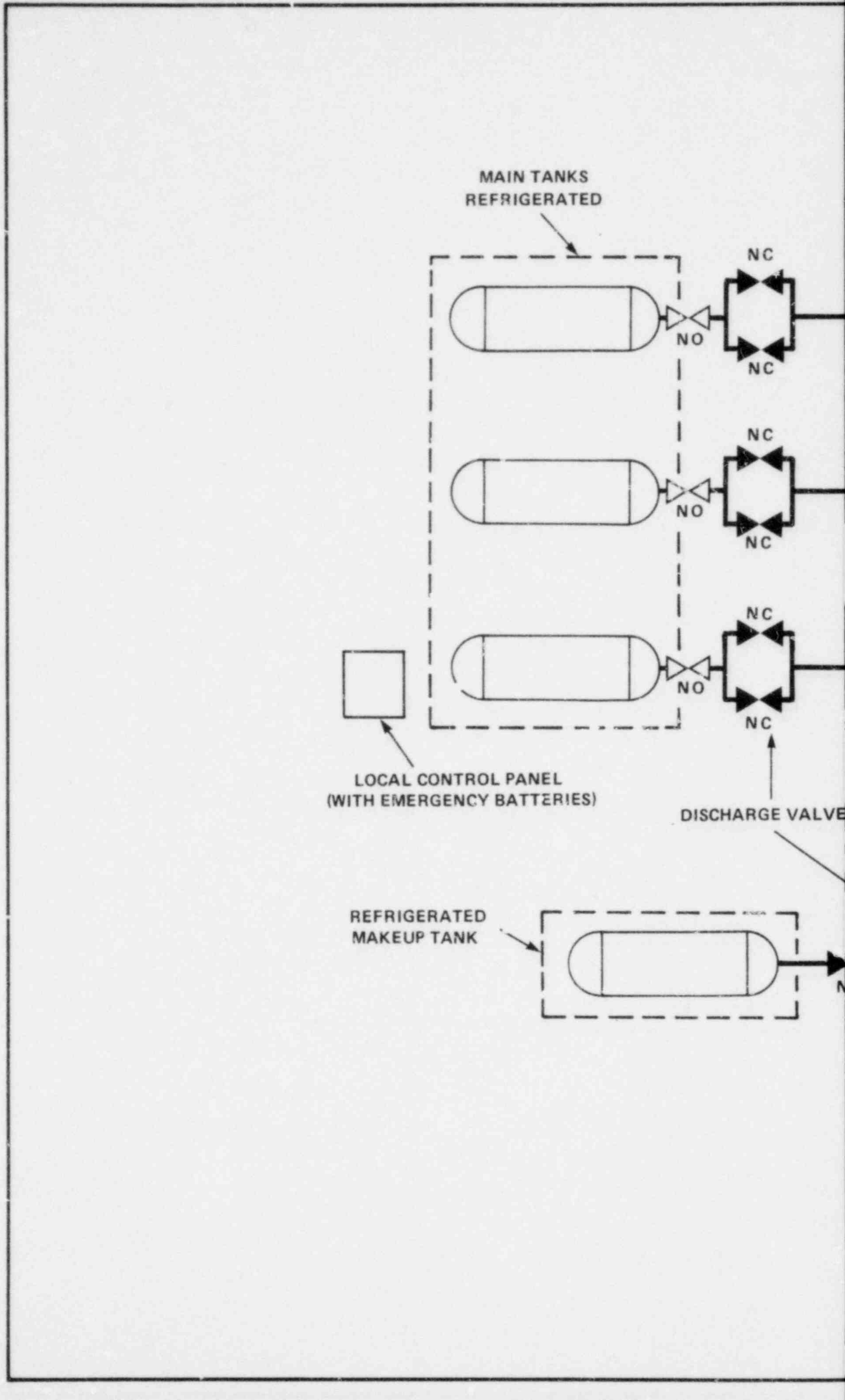
Each discharge valve will be pressure tested and operated. Provisions will be provided so that each active component of the Post-Accident Inerting System can be tested onsite periodically. "Puff" testing can be performed to pressure test and clean out the piping system and instrumentation.

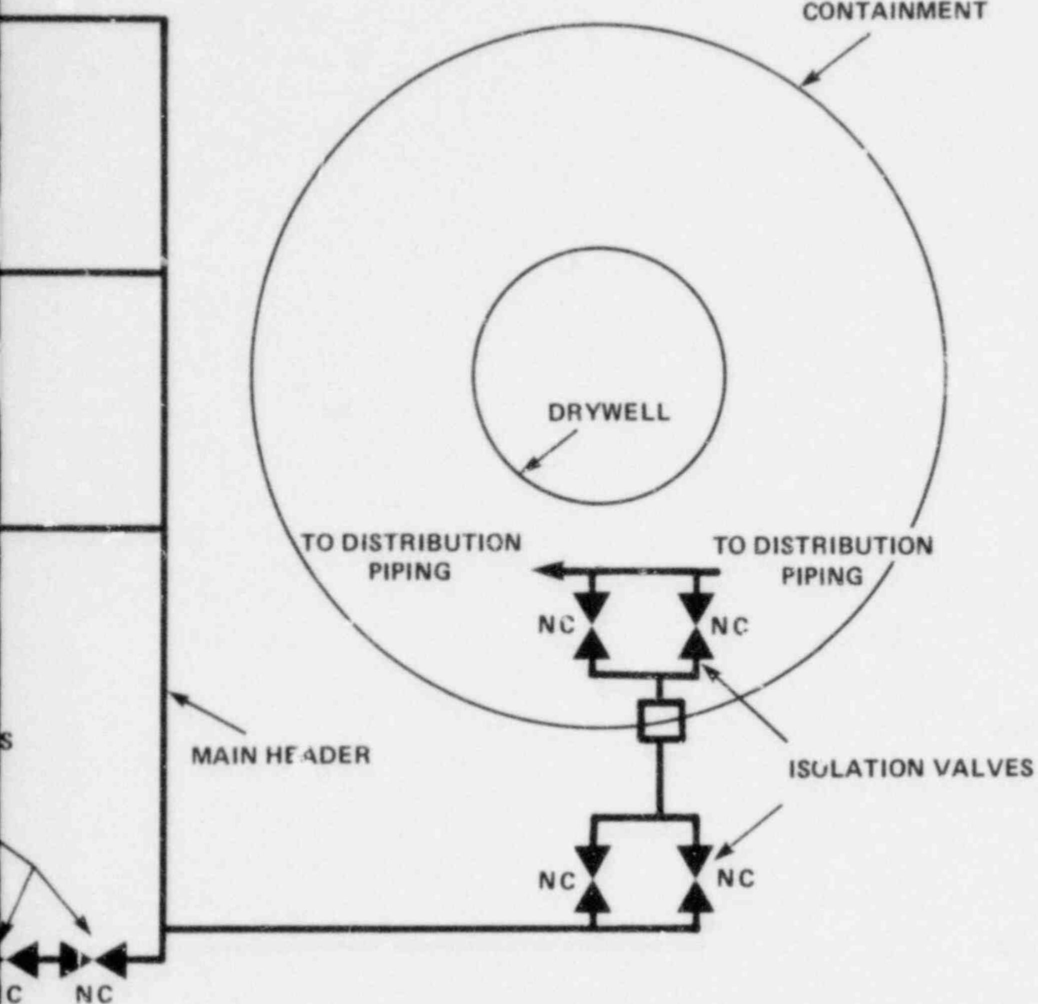
Instrumentation Requirements

Annunciation will occur on system initiation, on detection of flow in the main header, on high pressure or low liquid level in the tank and on high and low temperature in the inerting enclosure. Instrumentation will include discharge valve position, tank pressure, and inerting enclosure temperature. Local instruments monitor level and temperature gauges in the tanks and system battery readiness. 57

Materials

Piping from the tank to the inboard containment valves is ASTM A106, schedule 80, seamless steel pipe. Distribution piping is ASTM A106, schedule 40, seamless steel pipe. Nozzles will be stainless steel: fittings will be standard weight steel. A complete list of materials by commercial name and showing estimated quantities and physical characteristics cannot be completed until the detailed design is established.


MAIN TANKS
REFRIGERATED
NC
NO
NC
NC
NO
NC
NC
NO
NC
LOCAL CONTROL PANEL
(WITH EMERGENCY BATTERIES)
DISCHARGE VALVE
REFRIGERATED
MAKEUP TANK



AM. NO. 59, (6/81)

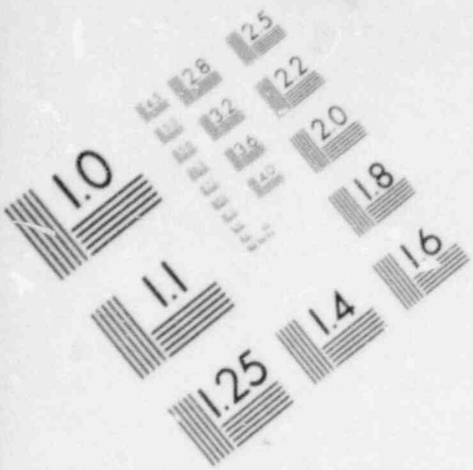
HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

GENERAL ARRANGEMENT OF
CARBON DIOXIDE
INERTING SYSTEM (CDIS)

FIGURE II.B.8.3-1

IMAGE EVALUATION
TEST TARGET (MT-3)

6″



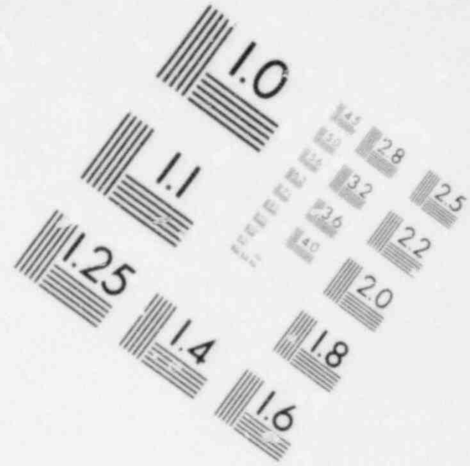
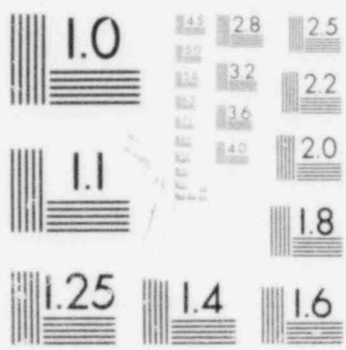
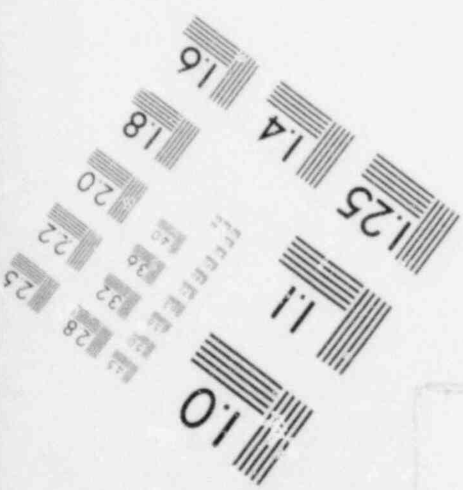
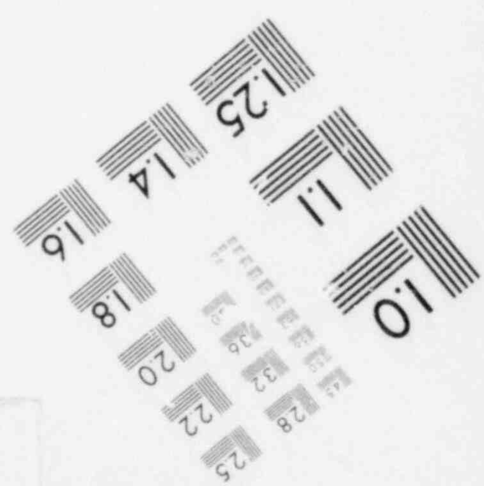


IMAGE EVALUATION TEST TARGET (MT-3)



6″





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ITEM II.D.1 TESTING REQUIREMENTS

NUREG 0718 REQUIREMENT

Applicants and their agents shall plan and carry out a test program and model development to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents. Consideration of anticipated transient without scram (ATWS) conditions shall be included in the test planning. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. Applicants shall submit, prior to the issuance of the construction permits or manufacturing license, a general explanation of how the testing requirements will be met. Sufficient detail should be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Applicants shall (1) demonstrate the applicability of the generic tests conducted under II.D.1 to their particular plants and (2) modify their plant designs as necessary. Applicants shall commit, prior to the issuance of the construction permits or manufacturing license, to comply with these requirements and shall submit within six months following the completion of the generic tests or the issuance of construction permits, whichever is later, a detailed explanation of how the test results will be incorporated in the plant design. Sufficient detail should be presented to provide reasonable assurance that the requirements resulting from the test will be implemented properly prior to the issuance of operating licenses. 57

RESPONSE

Performance testing of BWR safety/relief valves will be done beyond the current qualification requirements. This testing will be sponsored by the utilities of the BWR Owners' Group, in response to NUREG-0578 Requirement 2.1.2.

In July, 1979, the NRC issued its TMI short-term Lessons Learned Report (NUREG-0578). In this report, the NRC required that testing be conducted "to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents". The "expected operating conditions" were to be determined through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70 Revision 2. The discussion accompanying the requirement gave primary emphasis to two-phase and liquid flow conditions.

Reference 1 presents an evaluation of those Regulatory Guide 1.70 Revision 2 events which have the potential for producing liquid or two-phase flow discharge from the safety relief valves. This report is applicable to most

BWRs. It is specifically applicable to all plants that have level 8 trips (high water level) on high pressure inventory maintenance systems (e.g. Feedwater, High Pressure Coolant Injection and Reactor Core Isolation Cooling). Allens Creek is included within this group of plants.

The conclusion reached after a detailed review of all identified events (see Table 2-1 of Reference 1) is that a test which simulates the alternate shutdown cooling mode should be performed. This event is an anticipated operating condition which has been considered in the design analysis of plants. The BWR Owners' Group has committed to perform liquid and two-phase flow safety valve tests for the conditions which can occur for this mode of operation (Reference 3). All other events which were identified are either of sufficiently low probability or low consequence such that no additional testing is warranted.

A description is given in Reference 2 for those tests which will be run on typical S/RV's for BWR/2 through BWR/6 plants to demonstrate ability to perform satisfactorily under the condition in which low pressure (i.e., up to 250 ± 20 psig) water passes through the valve instead of saturated steam. This corresponds to conditions expected during the Alternate Shutdown Cooling Mode, 57
i.e., the mode in which low pressure pumps are injecting cold water into the reactor vessel and this water is vented through the S/RV's back to the suppression pool.

The low pressure water test will serve the following two purposes:

A. To demonstrate the capability of each type of S/RV to operate satisfactorily under the bounding cases of release of low pressure water with resultant, typical BWR pipe loads on the S/RV.

B. To measure the S/RV discharge line (S/RVDL) loads during water discharge through S/RV's.

Six different S/RV's will be tested in the relief mode (normal operating mode for low pressure).

The specimens to be tested consist of 6 x 8 (inlet dia. x outlet dia. in inches) pilot-operated Electromatic Relief Valve, 6 x 10 twoand three-stage pilot-operated Target Rock S/RV's, 6 x 10 and 8 x 10 Crosby direct-acting S/RV's and 8 x 10 Dikkers direct-acting S/RV.

ACNGS will use the Crosby 8 x 10 direct-acting S/RV's, and is thus covered by the testing program. In addition, the data gathered on discharge piping response will be considered in the design of the ACNGS SRV discharge piping, which will be designed for the same two-phase and solid water flow conditions for which the valves are being tested.

REFERENCES

1. "Event Evaluation for BWR Safety Relief Valve Testing Required by NUREG-0578, 2.1.2" enclosed with letter from D.B. Waters (BWR Owners' Group to R.H. Vollmer (NRC) dated September 17, 1980 and titled "NUREG Requirement 2.1.2 - Performance Testing of BWR and PWR Relief and Safety Valves".

2. "NUREG-0578 BWR Safety/Relief Valve Test Description" enclosed with letter from D.B. Waters (BWR Owners' Group) to R.H. Vollmer (NRC) dated September 17, 1980 and titled "NUREG-0578 Requirement 2.1.2 - Performance Testing of BWR and PWR Relief and Safety Valves". 57

3. Letter, T.D. Keenan (BWR Owners' Group) to D.G. Eisenhut (NRC) dated December 14, 1979 and titled "BWR Owner's Group Implementation of NUREG-0578 Requirement 2.1.2".

ITEM II.D.3 RELIEF AND SAFETY POSITION INDICATION

NUREG 0718 REQUIREMENT

Applicants shall modify their plant designs as necessary to provide direct indication of relief and safety valve position in the control room. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to issuance of operating licenses.

RESPONSE

As shown in PSAR Section 7.5.1.4.2.3.3.e, safety relief valve position indication will be determined by pressure measurement in the discharge pipe. This has been verified by the BWROG to be an adequate indication of SRV position indication by studying data from operating plants, which was submitted to the NRC by letter, T. Keenan (BWR Owners Group) to D.G. Eisenhut (NRC) dated October 17, 1979. 57

The actual pressure setpoint to be used at ACNGS will be determined from a combination of analysis and field test data, and will be submitted with the FSAR. Indication in the Main Control Room will be on two light matrices, one for each division of position measurement, on the Reactor Core Cooling Systems benchboard (H13-P601) above the manual control switches for the relief valves. The indication will be redundant, safety grade, seismically and environmentally qualified, and powered from a Class 1E power source. An alarm indicating that an SRV is open will be provided, but will not be safety grade.

There are no questions regarding technical feasibility or state-of-the-art of the SRV position indication design, nor is there any concern that it cannot be implemented prior to OL issuance.

ITEM II.E.4.2 ISOLATION DEPENDABILITY

NUREG 0718 REQUIREMENT

"Containment isolation system designs shall comply with the recommendations of Standard Review Plan Section 6.2.4.

All plants shall give careful consideration to the definition of essential and nonessential systems, identify each system determined to be essential, identify each system determined to be nonessential, and describe the basis for selection of each essential system. All nonessential systems shall be automatically isolated by the containment isolation signal. Revision 2 to Regulatory Guide 1.141 will contain guidance on the classification of essential versus nonessential systems and is due to be issued by June 1981.

For post-accident situations, each nonessential penetration (except instrument lines) is required to have two isolation barriers in series that meet the requirements of General Design Criteria 54, 55, 56, and 57, as clarified by Standard Review Plan, Section 6.2.4. Isolation must be performed automatically (i.e., no credit can be given for operator action). Manual valves must be sealed closed, as defined by Standard Review Plan, Section 6.2.4, to qualify as an isolation barrier. Each automatic isolation valve in a nonessential penetration must receive diverse isolation signals.

The design of control systems for automatic containment isolation valves shall 57
be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action. Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method of meeting this requirement.

Ganged reopening of containment isolation valves is not acceptable. Reopening of isolation valves must be performed on a valve-by-valve basis, or on a line-by-line basis, provided that electrical independence and other single-failure criteria continue to be satisfied.

The containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions. The containment pressure history during normal operation for similar operating plants should be used as a basis for arriving at an appropriate minimum pressure setpoint for initiating containment isolation. The pressure setpoint selected should be far enough above the maximum observed (or expected) pressure inside containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuations due to the accuracy of the pressure sensor. A margin of 1 psi above the maximum expected containment pressure should be adequate to account for instrument error. Any proposed values greater than 1 psi will require detailed justification.

All systems that provide an open path from the containment to the environs (e.g., containment purge and vent systems) must close on a safety-grade high radiation signal. 57

Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979, must be sealed closed as defined in SRP 6.2.4, Item II.3.f 59
during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit state of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by 57
specifying the design concept sleected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses."

RESPONSE

Details of the Containment Isolation System (CIS) are given in Sections 6.2.4 and 7.3.1.1.2. A summary of conformance to NUREG 0660/0718 Item II.E.4.2 Criteria is given below.

(1) Compliance with SRP 6.2.4 (Rev. 1)

The ACNGS CIS is in full compliance with SRP 6.2.4, Revision 1. All lines penetrating containment meet the letter of GDC 55, 56 or 57 as applicable, with the exception of the lines listed below, which meets the alternate provisions permitted by SRP 6.2.4, as follows:

- ECCS Pump Suction Lines (Penetrations 11, 12, 13, 35, 41 and 50):

  Each of these lines has a single isolation valve located outside containment. There is no valve inside containment because it would be submerged in the suppression pool, and, to enhance system reliability, there is no second valve outside containment. The following compensating features are provided for these lines in 59
  conformance with SRP 6.2.4, Acceptance Criteria II.3.e.

  o the systems are closed outside containment.

  o no single active failure will cause loss of the containment function.

  o the closed system outside containment is missile protected, seismic Category I, Safety Class 2 and is designed for pressure and temperature that exceeds the values for the containment.

o the closed system outside containment will be leak tested (see NUREG-0718 Item III.D.1.1. response).

o the valves and piping between the containment and valves are conservatively designed in accordance with SRP Section 3.6.2.

o the capability to detect (Class IE sump level indicators) and terminate leakage from the valve stems is provided (see Section 6.3.2.12).

- ECCS Pump Minimum Flow Recirculation lines (Penetrations 14, 15 16, 39, 44 and 52):

Each of these lines has a single isolation valve located outside containment. To enhance system reliability, there is no second isolation valve, as flow through the minimum flow recirculation line is essential for pump protection when the main discharge path is closed. The same compensating features are provided for these lines as for the ECCS pump suction lines, in conformance with SRP 6.2.4, Acceptance Criteria II.3.e.

- RHR Heat Exchanger Relief Valve Discharge Lines (Penetrations 21 and 24): 59

Each of these lines has a single isolation valve (the relief valve) outside containment. There is no second isolation valve because the ASME B&PV Code does not allow such valves downstream of relief valves to preserve maximum reliability of the relief function. The same compensating features are provided for these lines as for the ECCS pump suction lines in conformance with SRP 6.2.4, Acceptance Criteria II.3.e.

- RCIC Turbine Exhaust Lines (Penetration 56):

This line has a single isolation valve outside containment. There is no valve inside containment because it would be submerged in the suppression pool or located within the pool swell region, and, to enhance system reliability, there is no second valve outside containment. The same compensating features are provided for these lines as for the ECCS pump suction lines, in conformance with SRP 6.2.4, Acceptance Criteria II.3.e.

- RCIC Turbine Exhaust Vacuum Breaker Line (Pentration 51):

This line is provided with two isolation valves outside containment. There is no valve inside containment because it would be located within the pool swell region. The following compensating features are provided for this line in accordance with SRP 6.2.4, Acceptance Criteria II.3.d.

- o the valve closest to containment and the piping between containment and the valve are conservatively designed in accordance with SRP Section 3.6.2. 59

- o the capability to detect (Class IE sump level indicators) and terminate leakage from the valve stems is provided.

- Suppression Pool Cleanup Suction Line (Penetration 125)

This line is provided with two isolation valves outside containment. There is no valve inside containment because it would be submerged in the suppression pool. The same compensating features are provided for this line as for the RCIC Turbine Exhaust Line, except that leakage detection is by measurement of flow mismatch in the suction and discharge lines, in conformance with SRP 6.2.4, Acceptance Criteria II.3.d.

(2) Identification of essential and nonessential systems

Systems penetrating containment are categorized as follows for isolation purposes: 57

a. ESSENTIAL: safety and support system for which credit is taken in the accident analyses (e.g. ECCS). Essential systems are not automatically isolated on accident signals.

b. INTERMEDIATE: systems which could be useful in mitigating an accident, but for which credit was not taken in the accident analysis (e.g., equipment protection closed cooling water). Intermediate systems are automatically isolated by diverse accident signals. Certain intermediate systems are isolated on an additional signal 59 indicating a loss of system integrity. The control room hand switch for containment valves in intermediate lines can reopen the valves when the accident signal is still present, but not when the signal indicating loss of system integrity is present. This permits the 57 operator to use all available systems to cope with an accident while still maintaining the effectiveness of the containment.

In summary, the isolation provisions for intermediate systems have the same essential features as nonessential systems (double barrier isolation, automatic isolation on diverse accident signals). The main difference is that non-essential systems can be manually re-opened by the operator while the accident signal is still present. 59

c. NONESSENTIAL: systems that are not required or useful in mitigating an accident (e.g. drywell low purity sump). Nonessential systems are automatically isolated by diverse accident signals. The control room hand switches for containment isolation valves in nonessential lines cannot reopen the valves when the accident signal is 57 still present (unless the operator holds the spring-return-to-auto switch in the OPEN position continuously).

System designation into the above categories is shown on Table 6.2-12.

(3) Isolation of nonessential systems

a. Each nonessential penetration has two isolation barriers in series that meet GDC 54, 55, 56 or 57 as applicable.

b. Isolation of nonessential penetrations with power operated valves is automatic.

c. Manual barriers (such as manual valves, flanges, etc.) on nonessential penetrations are sealed closed when containment integrity is required.

d. All automatic isolation valves in nonessential systems receive diverse isolation signals.

(4) Reopening of isolation valves on isolation signal resetting

Containment isolation valve logic is such that the valve will not automatically reopen when the automatic isolation signal is reset. Reopening of a containment isolation valve requires deliberate operator action (see also Item (1) above). No administrative provisions are necessary to insure this. 57

(5) Ganged reopening of isolation valves

There is no ganged reopening of automatic containment isolation valves once closed by their respective isolation signal(s). Such reopening requires operator action on a valve-by-valve basis.

(6) Containment pressure isolation setpoint

The drywell pressure setpoint to initiate containment isolation is 2 psig, which allows 1 psig for operational pressure swings and 1 psig for instrument error to minimize the potential for spurious containment isolation.

(7) High radiation isolation of open path lines

All lines which provide an open path from the containment to the environment, e.g. the containment purge and vent lines, will isolate on a safety grade high radiation signal. The radiation monitors are located such that containment atmosphere releases through the purge line prior to isolation from the radiation signal will not result in doses in excess of 10CRF100 guidelines for a spectrum of accidents. 59 In addition, this signal will also isolate certain other nonessential lines, such as the containment and drywell sumps, as shown on Table 6.2-12. 57

(8) Containment purge isolation valves

The containment purge and vent isolation valves will satisfy the operability criteria of CSB 6-4. See Item II.E.4.4 (3).

(9) Level of information 57

The above supplements the PSAR information on the CIS, and is consistent with preliminary design information normally required at the CP stage of review. There is no new, novel design, and there are no concerns regarding technical feasibility, state-of-the-art or ability to implement the intended CIS design.

ITEM II.E.4.4 PURGING

NUREG 0718 REQUIREMENT

"Applicants shall (1) provide a capability for containment purging/venting designed to maximize purging time, consistent with ALARA principles for occupational exposure, (2) evaluate the performance of purging and venting isolation valves against accident pressure, (3) address the interim NRC guidance on valve operability, (4) adopt procedures and restrictions consistent with the revised requirements; and (5) provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses."

RESPONSE

The general safety concern over containment purging stems from the presumption that the purge line provides an open path for accident releases prior to isolation, and further, that the dynamic effects of the accident may interfere with effective isolation of the purge line. 57

These presumptions are not directly applicable to the Mark III containment design. The reactor coolant system piping is enclosed in the drywell, which communicates with the containment only through the suppression pool. Releases from the primary system are subjected to the quenching and scrubbing action of the suppression pool before entering the containment, so the purge system does not provide an open path for primary system releases in the same sense as other containment designs may. Even so, special care is being taken in the purge system design, specifically for valve operability assurance (Items 2 and 3 below).

The specific points of NUREG 0718 are addressed below:

(1) Purging consistent with ALARA

The present design calls for continuous purging of the containment during power operation at 5000 cfm through an 18" line to reduce airborne radionuclide concentrations to a level which permits continuous access. This is in keeping with occupational ALARA considerations, because extensive containment access for routine maintenance is required.

(2) Performance of purge and vent valves against accident pressure

The purge and vent containment isolation valves are not expected to have to close against the containment design pressure even assuming that a DBA LOCA occurs. The purge and vent lines begin to isolate when drywell pressure reaches 2 psig (almost instantaneously, as shown on Figure 6.2-5). 57
Containment pressure is virtually unaffected for the first several seconds of the accident, and does not rise to near the design pressure for many hours. Regardless, the purge and vent containment isolation valves will be designed to close against the containment design pressure of 15 psig.

(3) Interim NRC guidance on valve operability

The purge and vent containment isolation valves will meet the "Guidelines for Demonstration of Operability of Purge and Vent Valves," reproduced in 59
Table II.E.4.4-1.

(4) Procedures and restrictions consistent with revised requirements

There are no additional procedures or restrictions on containment purge deemed necessary. 57

(5) Assurance of purge and vent isolation reliability 59

The inherent design of the Mark III containment and the added conservatism in isolation valve design and testing give a high level of assurance that the purge and vent lines are reliable to isolate under accident conditions. 57

TABLE II.E.4.4-1

GUIDELINES FOR DEMONSTRATION OF OPERABILITY OF PURGE AND VENT VALVES

OPERABILITY

In order to establish operability it must be shown that the valve actuator's torque capability has sufficient margin to overcome or resist the torques and/or forces (i.e., fluid dynamic, bearing, seating, friction) that resist closure when stroking from the initial open position to full seated (bubble tight) in the time limit specified. This should be predicted on the pressure(s) established in the containment following a design basis LOCA. Considerations which should be addressed in assuring valve design adequacy include:

1. Valve closure rate versus time - i.e., constant rate or other.

2. Flow direction through valve; P across valve.

3. Single valve closure (inside containment or outside containment valve) or simultaneous closure. Establish worst case. 59

4. Containment back pressure effect on closing torque margins of air operated valve which vent pilot air inside containment.

5. Adequacy of accumulator (when used) sizing and initial charge for valve closure requirements.

6. For valve operators using torque limiting devices - are the settings of the devices compatible with the torques required to operate the valve during the design basis condition.

7. The effect of the piping system (turns, branches) upstream and downstream of all valve installations.

8. The effect of butterfly valve disc and shaft orientation to the fluid mixture egressing from the containment.

DEMONSTRATION

Demonstration of the various aspects of operability of purge and vent valves may be by analysis, bench testing, in-situ testing or a combination of these means.

Purge and vent valve structural elements (valve/actuator assembly) must be evaluated to have sufficient stress margins to withstand loads imposed while valve closes during a design basis accident. Torsional shear, shear, bending, tension and compression loads/stresses should be considered. Seismic loading should be addressed.

Once valve closure and structural integrity are assured by analysis, testing or a suitable combination, a determination of the sealing integrity after closure and long term exposure to the containment environment should be evaluated. Emphasis should be directed at the effect of radiation and of the containment spray chemical solutions on seal material. Other aspects such as the effect on sealing from outside ambient temperatures and debris should be considered.

The following considerations apply when testing is chosen as a means for demonstrating valve operability:

Bench Testing

A. Bench testing can be used to demonstrate suitability of the in-service valve by reason of its traceability in design to a test valve. The following factors should be considered when qualifying valves through bench testing.

1. Whether a valve was qualified by testing of an identical valve assembly or by extrapolation of data from a similarly designed valve.

2. Whether measures were taken to assure that piping upstream and downstream and valve orientation are simulated. 59

3. Whether the following load and environmental factors were considered:

a. Simulation of LOCA
b. Seismic loading
c. Temperature soak
d. Radiation exposure
e. Chemical exposure
f. Debris

B. Bench testing of installed valves to demonstrate the suitability of the specific valve to perform its required function during the postulated design basis accident is acceptable.

1. The factors listed in items A.2 and A.3 should be considered when taking this approach.

In-situ Testing

In-situ testing of purge and vent valves may be performed to confirm the suitability of the valve under actual conditions. When performing such tests, the conditions (loading, environment) to which the valve(s) will be subjected during the test should simulate the design basis accident.

NOTE: Post test valve examination should be performed to establish structural integrity of the key valve/actuator components.

TABLE II.E.4.4-1(Cont'd)

CLARIFICATION OF SEPT. 27 LETTER TO LICENSEES
REGARDING DEMONSTRATION OF OPERABILITY OF PURGE AND VENT VALVES

1. The ΔP across the valve is in part predicated on the containment pressure and gas density conditions. What were the containment conditions used to determine the ΔP's across the valve at the incremental angle positions during the closure cycle?

2. Were the dynamic torque coefficients used for the determination of torques developed, based on data resulting from actual flow tests conducted on the particular disc shape/design/size? What was the basis used to predict torques developed in valve sizes different (especially larger valves) than the sizes known to have undergone flow tests? 59

3. Were installation effects accounted for in the determination of dynamic torques developed? Dynamic torques are known to be affected for example, by flow direction through valves with offset discs, by downstream piping backpressure, by shaft orientation relative to elbows, etc. What was the basis (test data or other) used to predict dynamic torques for the particular valve installation?

4. When comparing the containment pressure response profile against the valve position at a given instant of time, was the valve closure rate vs. time (i.e. constant or other) taken into account? For air operated valves equipped with spring return operators, has the lag time from the time the valve receives a signal to the time the valve starts to stroke been accounted for?

   NOTE: Where a butterfly valve assembly is equipped with spring to close air operators (cylinder diaphragm, etc.), there typically is a lag time from the time the isolation signal is received (solenoid valve usually de-energized) to the time the operator starts to move the valve. In the case of an air cylinder, the pilot air on the opening side of the cylinder is approximately 90 psig when the valve is open, and the spring force available may not start to move the piston until the air on this opening side is vented (solenoid valve de-energizes) below about 65 psig, thus the lag time.

5. Provide the necessary information for the table shown below for valve positions from the initial open position to the seated position (10^{o} increments if practical).

Valve Position (in degrees - 90^{o} = full open)	Predicted ΔP (across valve)	Maximum ΔP (capability)

6. What Code, standards or other criteria, was the valve designed to? What are the stress allowables (tension, shear, torsion, etc) used for critical elements such as disc, pins, shaft yoke, etc. in the valve assembly? What load combinations were used?

TABLE II.E.4.4-1 (Cont'd)

9. For those valve assemblies (with air operators) inside containment, has the containment pressure rise (backpressure) been considered as to its effect on torque margins available (to close and seat the valve) from the actuator? During the closure period, air must be vented from the actuators opening side through the solenoid valve into this backpressure. Discuss the installed actuator bleed configuration and provide basis for not considering this backpressure effect a problem on torque margin. Valve assembly using 4 way solenoid valve should especially be reviewed.

10. Where air operated valve assemblies use accumulators as the fail-safe feature, describe the accumulator air system configuration and its operation. Provide necessary information to show the adequacy of the accumulator to stroke the valve i.e. sizing and operation starting from lower limits of initial air pressure charge. Discuss active electrical components in the accumulator system, and the basis used to determine their qualification for the environmental conditions experienced. Is the accumulator system seismically designed? 59

11. For valve assemblies requiring a seal pressurization system (inflatable main seal) describe the air pressurization system configuration and operation including means used to determine that valve closure and seal pressurization have taken place. Discuss active electrical components in this system, and the basis used to determine their qualification for the environmental condition experienced. Is this system seismically designed.

For this type valve, has it been determined that the "valve travel stops" (closed position) are capable of withstanding the loads imposed at closure during the DBA-LOCA conditions.

12. Describe the modification made to the valve assembly to limit the opening angle. With this modification, is there sufficient torque margin available from the operator to overcome any dynamic torques developed that tend to oppose valve closure, starting from the valve's initial open position? Is there sufficient torque margin available from the operator to fully seat the valve? Consider seating torques required with seats that have been at low ambient temperatures.

13. Does the maximum torque developed by the valve during closure exceed the maximum torque rating of the operators? Could this affect operability?

14. Has the maximum torque value determined in =12 been found to be compatible with torque limiting settings where applicable?

15. Where electric motor operators are used, has the minimum available voltage to the electric operator under both normal or emergency modes been determined and specified to the operator manufacturer, to assure the adequacy of the operator to stroke the valve as DBA

TABLE II.E.4.4-1 (Cont'd)

conditions with these lower limit voltages available. Does this reduced voltage operation result in any significant change in stroke timing? Describe the emergency mode power source used.

16. Where electric operator units are equipped with handwheels, does their design provide for automatic re-engagement of the motor operator following the handwheel mode of operation? If not, what steps are taken to preclude the possibility of the valve being left in the handwheel mode following some maintenance, test etc. type operation.

17. Describe the tests and/or analysis performed to establish the qualification of the valve to perform its intended function under the environmental conditions exposed to during and after the DBA following its long term exposure to the normal plant environment.

18. What basis is used to establish the qualification of the valve, operators, solenoids, valves? How was the valve assembly (valve/operators) seismically qualified (test, analysis, etc.)?

19. Where testing was accomplished, describe the type tests performed conditions used etc. Tests (where applicable) such as flow tests, aging simulation (thermal, radiation, wear, vibration endurance, seismic) LOCA-DBA environment (radiation, steam, chemicals) should be pointed out. 59

20. Where analysis was used, provide the rationale used to reach the decision that analysis could be used in lieu of testing. Discuss conditions, assumptions, other test data, handbook data and classical problems as they may apply.

21. Have the preventive maintenance instructions (part replacement, lubrication, periodic cycling, etc.) established by the manufacturer been reviewed, and are they being followed? Consideration should especially be given to elastomeric components in valve body, operators, solenoids, etc. where this hardware is installed inside containment.

22. Assess the structural capability of any ducting or piping in the purge system which is upstream or downstream of the valves and is exposed to the flow condition associated with the LOCA and seismic event. In particular consider the effects of loose debris from the pipe or duct system on the closure capability of these valves.

ITEM II.F.1 ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION

NUREG 0718 REQUIREMENT

"Applicants shall comply with the requirements addressed in NUREG-0737. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses."

NUREG 0737 REQUIREMENT

"Item II.F.1 of NUREG-0660 contains the following subparts:

(1) Noble gas effluent radiological monitor;

(2) Provisions for continuous sampling of plant effluents for postaccident releases of radioactive iodines and particulates and onsite laboratory capabilities (this requirement was inadvertently omitted from NUREG-0660; see Attachment 2 that follows, for position); 57

(3) Containment high-range radiation monitor;

(4) Containment pressure monitor;

(5) Containment water level monitor; and

(6) Containment hydrogen concentration monitor

NUREG-0578 provided the basic requirements associated with items (1) through (3) above. Letters issued to all operating nuclear power plants dated September 13, 1979 and October 30, 1979 provided clarification of staff requirements associated with items (1) through (6) above. Attachments 1 through 6 present the NRC position on these matters.

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

(a) the use of this information by an operator during both normal and abnormal plant conditions,

(b) integration into emergency procedures,

(c) integration into operator training, and

(d) other alarms during emergency and need for prioritization of alarms."

RESPONSE

Subparts (1) and (3)-(6):

The additional accident monitoring instrumentation called for in NUREG 0737, Item II.F.1, Subparts (1) and (3)-(6) will be provided as discussed below. 57

(1) Noble gas effluent radiological monitor
(PSAR Section 12.2.4.1.2)

All credible post-accident release points are provided with high range noble gas effluent monitors, as shown on Table 12.2-5a. These monitors meet the design criteria specified in NUREG 0737, Table II.F.1-1.

(3) Containment high range radiation monitor
(PSAR Section 12.1.4)

Two channels of high range ($1-10^7$ R/hr gamma only) radiation monitoring instrumentation will be provided in the containment and in the drywell. These monitors will meet the design specifications in NUREG 0737, Item II.F.1, Attachment 3. 59

(4) Containment pressure monitor
(PSAR Section 7.5.1.4.2)

Four channels of containment pressure instrumentation with a wide range of -5 psig to at least 60 psig (four times design pressure) will be provided. The upper range may be higher (up to 80 psig) depending on the exact range of the transmitter to be purchased. These instruments are in addition to the accident normal range containment pressure monitors, and meet the requirements of Regulatory Guide 1.97, Revision 2 (see Item II.F.3) and NUREG 0737. 57 59

(5) Containment water level monitor
(PSAR Section 7.5.1.4.2)

Four channels of suppression pool water level instrumentation, each covering the range from the top of the ECCS suction strainers to 5' feet above the normal suppression pool level, will be provided. These instruments are in addition to the normal range suppression pool water level monitors, and meet the requirements of Regulatory Guide 1.97, Revision 2 (see Item II.F.3) and NUREG 0737. 57 59

(6) Containment hydrogen concentration monitor
(PSAR Section 7.5.1.4.2.11)

Two channels of containment hydrogen monitoring instrumentation, each covering the range of 0-30% will be provided. These will be in addition to the existing accident normal range (0-10%) hydrogen monitoring instrumentation, and meet the requirements of Regulatory Guide 1.97, Revision 2 (see Item II.F.3) and NUREG 0737. 57 59

The instrumentation in (1) and (3)-(6) will be redundant, safety grade, seismically and environmentally qualified for accident conditions including the span of its own measured parameter range, and powered from the onsite electrical system. This instrumentation is known to be commercially available (with the exception of Item (3), as discussed in the next paragraph), and space has been allocated for transmitter locations in the plant. The display location in the Main Control Room may be in dedicated post-accident panels or adjacent to or integrated with the existing normal range instrumentation display.

The above supplements the PSAR information on additional accident monitoring instrumentation, and is consistent with preliminary design information normally required at the CP stage of review. There are no concerns regarding technical feasibility, state-of-the-art or ability to implement this instrumentation design, with one exception. A fully environmentally qualified high range containment radiation monitor has not yet been found. However, this is not viewed as critical or even significant at this time.

Subpart (2)

The requirement of Subpart (2), Sampling of Plant Effluents, is not monitoring instrumentation per se, but is rather a sample collection and analysis capability. This will be provided in the manner specified in NUREG 0737, as described below. 57

Sample Collection: The same release points with high range noble gas effluent monitors will also have particulate and iodine sampling capability, as specified on Table 12.2-5a. Iodine samples will be taken with a charcoal or silver-zeolite cartridge and particulate samples with a filter which are located in the 3 stage monitoring unit cabinet, as shown on Figure 12.2-1. The post-accident iodine and particulate samples are extracted from the release point via the same sample line as the monitoring line.

Sample Transport: The sample cartridges will be placed in a portable shielded cask and taken to the counting room in the Personnel Access Building.

Sample analysis: Capability for the analysis of sample cartridges will be provided in the PAB counting room. Design of the counting facility will consider the design basis sample.

The precise location of the sample collection station will be selected upon completion of the post-accident shielding study (Item II.B.2), and the location will asure that a worker involved in the sample collection and transport operation will not receive an exposure greater than the GDC-19 specified dose design basis. 59

ITEM II.F.2 IDENTIFICATION OF AND RECOVERY FROM CONDITIONS LEADING TO INADEQUATE CORE COOLING

NUREG 0718 REQUIREMENT

Applicants shall describe their program for developing and implementing procedures to be used by the reactor operators to detect and recover from conditions leading to inadequate core cooling.

Applicants with PWR plants shall incorporate in their plant designs a primary coolant saturation meter and all applicants shall incorporate in their plant designs instrumentation to detect conditions with a potential that may lead to inadequate core cooling. Any additional equipment, including reactor water 57
level instrumentation, that could be used to indicate inadequate core cooling shall be incorporated in the plant designs. Design requirements for core exit thermocouples are described in NUREG-0737.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

RESPONSE

HL&P concurs with the BWR Owners Group position as stated in NEDO 24708A, 59
Revision 1, December, 1980, that no additional instrumentation is needed to monitor inadequate core cooling as discussed in the response to Item II.F.3. Regarding operator recognition of inadequate core cooling, the BWR Owners Group emergency procedure guidelines for recognizing the approach to inadequate core cooling, which were submitted to the NRC by letter, D.B. Waters (BWR Owners Group) to D.G. Eisnehut (NRC) dated January 31, 1981, will be incorporated into the ACNGS plant procedures. Details will be given in the 57
FSAR.

ITEM II.F.3 INSTRUMENTATION FOR MONITORING ACCIDENT CONDITIONS (REG. GUIDE 1.97)

NUREG 0718 REQUIREMENT

"Applicants shall provide in their facility design instrumentation to monitor plant variables and systems during and following an accident in accordance with defined design bases and Regulatory Guide 1.97, Rev. 2, December 1980. Designs are already established for much of the instrumentation that will be required; some of the requirements, however, may involve state-of-the-art designs or designs which have yet to be developed.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses." 57

RESPONSE

Conformance to Regulatory Guide 1.97, Revision 2 is discussed in Appendix C. There are no concerns regarding technical feasibility, state-of-the-art (except for the high range containment radiation monitors) or ability to implement the post accident monitoring instrumentation prior to OL issuance. Qualified transmitters with the required ranges are known to be commercially available, and space for the indicators has been allocated on the control room panels.

Radiation monitors with the required range of the containment high range monitor are available, but uncertainty exists as to their qualification. When a qualified monitor becomes available, it will be incorporated into the ACNGS design. In the meantime, the radiation monitoring computer will be left with provisions to accept these monitors data as inputs, and cables will be routed to the monitor locations to facilitate incorporation of the monitors when they become available.

ITEM II.K.1.22 DESCRIBE AUTOMATIC AND MANUAL ACTIONS FOR PROPER FUNCTIONING OF AUXILIARY HEAT REMOVAL SYSTEMS WHEN FW SYSTEM IS NOT OPERABLE

NUREG 0718 REQUIREMENT

Applicants with BWR plants shall address the requirements set forth in action 57
item 3 of IE Bulletin 79-08. A general explanation of how these requirements will be met is required prior to issuance of the construction permits. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly.

IE BULLETIN 79-08 ITEM 3

Describe the actions, both automatic and manual, necessary for proper
functioning of the auxiliary heat removal systems (e.g., RCIC) that are used 59
when the main feedwater system is not operable. For any manual action necessary, describe in summary form the procedure, by which this action is taken in a timely sense.

57

RESPONSE

Operating procedures for ACNGS have not been written, but the design is such that no manual actions are required initially to mitigate the consequences of a loss of feedwater, although the operator may take anticipatory actions
before automatic actions. These manual actions will be specified in the 59
operating procedures, and will be summarized in the FSAR. The following is a discussion of the response of the BWR-6 to loss of feedwater transients to demonstrate that no manual operator action is required immediately.

The BWR/6 NSSS is designed with self actuating systems to assure core cooling. An isolation event can be totally accommodated initially by automatic operation of engineered safety feature systems and RCIC which are redundant and diverse. These systems restore and maintain system parameters. During the long term, however, there is adequate time for the operator to take appropriate action. The operator need monitor and control only reactor vessel pressure and level. Furthermore, the operator has multiple parameters
available to provide information on system conditions. 57

All the loss-of-feedwater flow cases result in a proportional reduction of vessel inventory causing the vessel water level to drop. Corrective action normally begins as soon as low feedwater flow is sensed (any one or all pumps) and low level alarm (L4) is reached. At this time a reduction of the core recirculation flow is initiated to reduce power and thereby reduce the rate of level decrease. The first automatic protective action is the low level (L3) scram trip actuation. The reactor protection system responds within 1 second after this trip to scram the reactor. The low level (L3) scram trip function meets the single failure criterion.

For the Loss of Feedwater (LOF) and LOF + Stuck Open Relief Valve (SORV) cases main steam line isolation occurs from low steam line pressure.

For the LOF + NO HPCS/RCIC case main steam line isolation occurs from low water level (L1) signal. The main steam line isolation signal also initiates a main steam line isolation valve position scram trip as part of the normal isolation event. The reactor, however, is already scrammed and shut down by this time due to the L3 scram.

Loss of Feedwater

Vessel water level continues to drop reaching the L2 trip at about 20 sec. At this time, the recirculation system is complete tripped, and HPCS and RCIC operation is initiated. After the initiation delay HPCS and RCIC inject into the vessel causing the vessel water level to reach its minimum value about 6.5 feet above the TAF. In addition, operation of both HPCS and RCIC will cause the vessel to depressurize which causes a low pressure isolation to occur (assuming the reactor has remained in the RUN mode). After the HPCS and RCIC have tripped off on high vessel water level or are regulated by operator action, the vessel will repressurize to the setpoint of the lowest set relief valve which will open to limit the pressure rise.

Loss of Feedwater with Stuck-Open Relief Valve

Vessel water level continues to drop reaching the L2 trip at about 20 seconds. At this time, the recirculation system is completely tripped, and HPCS and RCIC operation is initiated. After the initiation delay HPCS and RCIC inject into the vessel causing the vessel water level to reach its minimum value about 6.5 feet above the TAF. In addition, operation of both HPCS and RCIC will cause the vessel to depressurize which causes a low pressure isolation to occur (assuming the reactor has remained in the RUN mode). After the HPCS and RCIC have tripped off on high vessel water level or are regulated by operator action, the vessel will repressurize to the setpoint of the lowest set relief valve which will open to limit the pressure rise. It is assumed in this case, that the relief valve fails to close when the reactor pressure drops below the relief valve reset point, thus remaining stuck open. The stuck-open relief valve causes the reactor to depressurize to the point where the shutdown cooling system can be put into operation.

57

Loss of Feedwater with no HPCS/RCIC

Vessel water level continues to drop, reaching the L2 trip at about 20 seconds. At this time, the recirculation system is completely tripped. With the failure of HPCS and RCIC the vessel water level continues to drop and the level outside the core shroud reaches the low level (L1) trip. At this time the main steam line isolation valve will close. The operator can maintain adequate core cooling by manual actuation of the relief valves or ADS to lower reactor pressure and allow use of the low pressure ECCS in time to prevent core uncovery. In this case it was assumed that the operator performed the manual operation at the low level (L1) trip point.

ITEM II.K.3.23 CENTRAL WATER LEVEL RECORDING

NUREG 0718 REQUIREMENT

Applicants with BWR plants shall address the requirements set forth in Item B.2 of NUREG 0626. Applicants shall implement design modifications as necessary to meet the requirements. Applicants shall submit, prior to issuance of construction permits, a general explanation of how the requirements will be met. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses. 57

NUREG 0626 ITEM B.2

In order to simplify the reading of the water level in the vessel and to provide the operators with a record of water level during transients, all BWRs should have the capability to record vessel water level over the range from the top of the vessel dome to the lowest pressure tap. This range of water level should be available in one location on recorders which meet normal post-accident recording requirements. The recorders should be started on a reactor trip signal.

RESPONSE

Reactor vessel water level instrumentation which meets the requirements of 59
Regulatory Guide 1.97 spanning the range from the bottom of the core support plate to the steam lines centerline will be provided as post-accident monitoring instrumentation, and will be continuously recorded. See Table 57
7.5-0.

ITEM III.A.1.2 UPGRADE LICENSEE EMERGENCY SUPPORT FACILITIES

NUREG 0718 REQUIREMENT

"Applicants shall address the requirements for a Technical Support Center, Operational Support Center and the Emergency Operations Facility. Applicants shall provide preliminary design information in accordance with the functional requirements of NUREG-0696 at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the design bases and criteria. Applicants shall demonstrate that the design concept is technically feasible and within the state-of-the-art, and that
there exists reasonable assurance that the requirements will be implemented 57
properly prior to the issuance of operating licenses."

RESPONSE

The TSC, OSC and EOF are described in Sections 13.3.5.2, 13.3.5.4 and 13.3.5.3 respectively. The information provided in those sections is consistent with
the level normally provided at the CP stage. 59

There are no questions regarding technical feasibility or state-of-the-art for
the emergency support facilities, nor is there any question that they can and 57
will be implemented prior to OL issuance.

ITEM III.D.1.1 PRIMARY COOLANT SOURCES OUTSIDE THE CONTAINMENT STRUCTURE

NUREG 0718 REQUIREMENT

"NRC is studying the need for improved acceptance criteria for systems outside containment that contain (or might contain) radioactive material either during normal operations or following an accident. These studies are to be completed in early 1981, and these matters will be included in the degraded-core rulemaking proceeding.

Applicants shall review the designs of such systems outside containment, and their provisions for leakage control and detection, overpressurization design, discharge points for waste gas venting systems, etc., with the goal of minimizing the possibility of exposure to workers and public during normal opera- 57
tions and in the event of an accident.

In this regard, applicants shall submit, prior to the issuance of construction permits, a general discussion of their approach to minimizing leakage from such systems outside containment, in sufficient detail to provide reasonable assurance that this objective will be met satisfactorily prior to issuance of operating licenses."

RESPONSE

Systems outside containment which may contain primary coolant and the liquid 59
and gaseous radwaste systems are designed to minimize leakage to the maximum extent practical, with the goal of minimizing the possibility of exposure to workers and public during normal operations and in the event of an accident. The means of achieving this are summarized in Table III.D.1.1-1 and described in the following.

1. Leak Reduction and Collection Design Features

The systems outside containment which may contain primary coolant were determined in conjunction with the post-accident shielding study (Item II.B.2). These systems incorporate various leak testing, reduction and/or collection features, including:

(i) welded/seamless piping system;

(ii) pumps with mechanical seals;

(iii) low point drains from the piping system and drains from equipment 57
such as pumps, leakoff from valves, etc. with single (or double) isolation valve(s), are routed to equipment drains and, thus, to sumps;

(iv) high point vents are provided with single (or double) isolation valve(s) and pipe caps;

(v) pressure test connections for temporary (or local) instrumentation are provided with single (or double) isolation valve(s) and pipe caps;

(vi) lantern rings and leakoffs are specified as follows:

a) for all valves having a connection to the leak detection system,

b) for rising stem for 4" and larger gate and globe valves, with a maximum operating temperature of 212°F.

(vii) packless metal diaphragm valves will be utilized for 2" and smaller valves located in the process line.

2. Leak Detection Features

For intersystem leakage detection, the following provisions are incorporated in the design:

(i) radiation monitors with alarm annunciation in the control room are provided for the Service Water systems which remove heat from the RHR, RWCU Non-Regenerative, and Fuel Pool Cooling heat exchangers, thereby enabling detection of radioactive fluid in the non-radioactive side due to heat exchanger tube leaks; 57

(ii) double block valves (and/or check valves) are provided in non-radioactive systems that are utilized for flushing radioactive system outside containment. In addition, relief valves are provided for protection against overpressurization.

(iii) instrumentation to monitor piping system integrity (differential flow transmitters), high differential and high ambient area temperature monitors, and sump level measurement.

3. Periodic System Leakage Testing

The design considerations discussed above are expected to reduce leakage to as low as practical levels for those systems outside containment (shown in Table II.D.1.1-1) that would or could contain highly radioactive fluid.

HL&P recognizes that to be effective such design must be verified initially and then monitored and maintained in that condition. To accomplish the goal of maintaining leakage to as low as practical levels, the following program is proposed for the initial measurement of leakage to establish a base leakage prior to the operating license and surveillance throughout plant lifetime. 59

BASELINE

1. Measure actual leak rates on the system under consideration.

SURVEILLANCE

1. Survey the affected systems with the systems operating at approximately expected pressures in either normal or test modes. 59

2. Systems containing gases are to be tested by use of tracer gases, by pressure decay testing or by metered make up tests.

3. Maintenance priorities will be high on leakage related tasks.

This program shall include periodic integrated leak tests on an interval not to exceed each refueling period. The period for inspection may be revised as operating experience is gained. Details of this testing will be provided in the ACNGS FSAR.

TABLE III.D.1.1-1

SUMMARY OF LEAK REDUCTION CRITERIA

ITEM No.	SYSTEM	SYSTEM LEAK TESTING CAPABILITY	LEAK REDUCTION/ COLLECTION MEASURES	INTERSYSTEM LEAK DETECTION/REDUCTION	REMARKS
1	Residual Heat Removal Systems (RHR)	1) Scheduled System Leak tests in conjunction with hydrotests will be made at intervals not to exceed each inspection interval as per ASME Section XI (para. IWA-5000).	1) Piping joints are of welded/seamless construction.	1) Essential Service Water System (low pressure side of the RHR Heat Exchangers) is provided with radiation monitors with alarms to alert the operator of tube leakages.	
		2) In accordance with the requirements of 10CFR50 Appendix 'J' - containment isolation valves will be tested during each reactor shutdown for refueling but in no case at intervals greater than 2 years.	2) Mechanical seals are provided on all pumps.	2) Double block valves (and/or check valves) are provided in non-radioactive systems that are utilized for flushing/draining radioactive systems outside containment. In addition, relief valves are provided for protection against over-pressurization.	
		3) Valve seat leakage tests and operability (surveillance) testing is performed in accordance with ASME XI (para. IWV-3000)	3) Leakages from equipment, safety relief valve, and low point drains from the piping systems will be piped to equipment hubs in the drainage system to reduce floor contamination.	3) Local test pressure connections are provided to check intersystem leakage through valves; for example, between RHR/RCIC System.	
			4) Lantern rings and leakoffs are specified as follows: i) For all valves having a connection to the leak detection system. ii) for rising stem for 4" & larger gate & globe valves, with a max. operating temp. of 212°F.		
			4A) Packless metal diaphragm valves will be utilized for 2" & smaller valves located in the process line.		
			5) The following leakage detection instrumentation is provided: A) High differential and high ambient area temperatures (inlet & outlet air is compared for differential air temperature).		

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TABLE III.D.1.1-1 (Cont'd)

ITEM No.	SYSTEM	YSTEM LEAK TESTING CAPABILITY	LEAK REDUCTION/ COLLECTION MEASURES	INTERSYSTEM LEAK DETECTION/REDUCTION	REMARKS
1	(RHR) (Cont'd)		B) Differential flow transmitter (will confirm the integrity of the system piping). C) ECCS sump level measurement and level alarms D) Two pressure transmitters are available to monitor leakage through the containment isolation valves. 6) Pressure test connections are provided for containment isolation valve leak rate testing per 10CFR50 Appendix J and ASME XI (IWV-3000). 7) Pressure test connections for temporary (or local) instrumentation are provided with single (or double) isolation valve(s) and pipe caps. 8) High point vents are provided with signle (or double) isolation valve(s) with caps.		
2	Low Pressure Core Spray (LPCS)	Same as Item #1	Same as Item #1 Except 5A, D	Same as Item #1 Except 1 and 3	
3	High Pressure Core Spray (HPCS)	Same as Item #1	Same as Item #1 Except 5A, D	Same as Item #1 Except 1 and 3	
4	Reactor Core Isolation Cooling (RCIC)	Same as Item #1	Same as Item #1	Same as Item #1 Except for 1.	
5	Main Steam Isolation Valve Leakage Control System (MSIV-LCS)	Same as Item #1	Same as Item #1, except for 2 and 5 level in MSIV-LCS Stem Leakage collection tanks monitored and alarmed.	Not applicable	

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TABLE III.D.1.1-1 (Cont'd)

ITEM No.	SYSTEM	SYSTEM LEAK TESTING CAPABILITY	LEAK REDUCTION/ COLLECTION MEASURES	INTERSYSTEM LEAK DETECTION/REDUCTION	REMARKS
6	RWCU Sampling System	Same as Item # 1 for safety portions of the system. In addition, the entire system is pressurized during normal operation and can be periodically observed for leakage. Note that valve seat leak tests in this system applies only to containment isolation valves; other valves exempted per IWV-1200.	1) Seamless sample tubing is used. 2) System has no pumps. 3) Same as Item 1, measure 3. 4) Zero leakage valves up to the sample panel. 5) Radiation Monitors (may include fixed or portable area monitors or stack monitors) will detect leakage. 6) Same as Item 1, measure 6. 7) Same as Item 1, measure 7. 8) System has no high point vents.	Auxiliary Closed Loop Cooling Water System (service water to the sample heat exchanger) is provided with a radiation monitor to alert the operator of tube leakages.	59
7	H_2 Monitoring/Post Accident Gaseous Sampling	Same as Item 6, except all testing is pneumatic.	1) Seamless sample tubing is used. 2) Zero leakage pumps will be used. 3) Same as Item 1, measure 3. 4) Zero leakage valves are used in the system. 5) Same as Item 1, measure 5 6) Same as Item 1, measure 6 7) Same as Item 1, measure 7 8) System has no high point vents.	No identifiable path for intersystem leakage.	
8	Post-Accident Liquid Sampling System	Same as Item 6.	Same as Item 6 except for Measure 2 2) Zero leakage pumps will be used.	Essential Service Cooling Water System (service water to the sample heat exchanger) is provided with a radiation monitor to alert the operator of tube leakages.	

TABLE III.D.1.1-1 (Cont'd)

ITEM No.	SYSTEM	SYSTEM LEAK TESTING CAPABILITY	LEAK REDUCTION/ COLLECTION MEASURES	INTERSYSTEM LEAK DETECTION/REDUCTION	REMARKS
9	Liquid Radwaste System	Same as Item # 1 for safety portions of the system (containment isolation valves and ECCS area sump isolation valves). In addition, the system is routinely in use and can be periodically observed for leakage.	Same as Item 1, except as follows: 4) Plug, ball (containment isolation valves only) and diaphragm valves are used. 5) Not Applicable.	Reboiler Steam Systems (service steam to the concentrators) is provided with radiation monitoring and conducting cells in the condensate return line to alert the operator to intersystem leakage.	
10	Gaseous Radwaste (Condenser Offgas) System	The system will be provided with connections to permit pressurization and pressure measurement for periodic leak testing.	1) Piping joints are of welded/ seamless construction. 2) System has no process pumps (or compressors) 3) Drains will be piped to an equipment drain or to the condenser. 4) Gas bufferred ball and metal diaphragm valves will be used. 5) Not applicable 6) Not applicable 7) Pressure test connections for temporary instrumentation are provided with isolation valves and pipe caps. 8) The system has no high point vents.	No identifiable path for inter-system leakage.	

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ITEM III.D.3.3 <u>IN-PLANT RADIATION MONITORING</u>

<u>NUREG 0718 REQUIREMENT</u>

Applicants shall review their designs to assure that provisions for monitoring inplant radiation and airborne radioactivity are appropriate for a broad range of routine and emergency conditions. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new 57
designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design basis and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

<u>RESPONSE</u>

Portable airborne iodine samplers and sample analysis equipment as required by 59
NUREG-0737 Item III.D.3.3 will be available onsite prior to the issuance of the operating license. This equipment will not be purchased for several years, but it is expected that it will be cart mounted and backup battery powered. Plant personnel will be trained in the use of this equipment under 57
both routine and emergency conditions. Details will be provided in the FSAR.

ITEM III.D.3.4 CONTROL ROOM HABITABILITY

NUREG 0718 REQUIREMENT

Applicants shall review the design of their facilities for conformance to requirements stated in the Action Plan. NRC will consider possible new criteria to preclude control room contamination via potential internal pathways indicated by the TMI-2 experience.

Applicants shall address prior to the issuance of the construction permits or manufacturing license, how they will implement the existing requirements set forth in this Action Plan item. Applicants shall also address the extent to which improvements have been made to prevent control room contamination via pathways not previously considered. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

RESPONSE

The present ACNGS control room habitability design is state-of-the-art. It was previously reviewed against Regulatory Guides 1.78 and 1.95 and Standard 57
Review Plans 2.2 and 6.4 by the NRC and found acceptable.

The Control Room Ventilation System design concept has not changed from that presented in the PSAR. It is a pressurized design with two widely separated emergency air intakes. Each intake is provided with redundant radiation monitors to permit the operator to select the cleaner intake post-accident and the intakes are provided with a "series valves in parallel" arrangement which is single failure proof for both isolation and opening. The normal air intake is automatically isolated on high radiation, and is single failure proof to isolate.

The Action Plan makes reference to possible new criteria on internal contamination pathways. Obviously, this new criteria cannot be addressed for ACNGS until it is issued, but it is not expected that any reasonable requirements would affect ACNGS. The Control Building is a separate structure, and the only equipment in the Control Building which is not control, instrumentation or electrical in nature is the HVAC equipment servicing the Control Building. This is unlike other plants in which the Control Room is actually a part of the Reactor Auxiliary Building, where potentially radioactive equipment may be located. Thus, it is difficult to envision any additional contamination pathways into the ACNGS Control Room other than the airborne pathway already considered.

Addressing the problem of control room contamination via potential internal pathways as indicated by the TMI-2 experience, it is observed that the causes of contamination at TMI-2 were: (a) lack of adequate control room access control, (b) access by contaminated personnel, (c) doors that were left open, and (d) the inability to accurately monitor the control room atmosphere in the recirculation mode.

ACNGS should not have the above listed difficulties as the plant will be provided with a dedicated Technical Support Center (TSC) and an onsite Operational Support Center to be used as staging areas for emergency support personnel. 59

A three stage continuous air monitor will be provided inside the control room ventilation system to check accurately on possible control room airborne contamination at all times. Portable iodine monitors (see Item III.D.3.3) will be available to control room personnel to be used in checking on that specific and important type of airborne radioactive contamination.

NUREG 0718 CATEGORY 5

"A requirement for information of the type customarily reviewed at the preliminary design stage for the following types of items:

a. Items for which the required information should be sufficient to demonstrate that the requirement has been satisified by the application. This is the kind of information and degree of detail customarily provided at the preliminary design stage with respect to site and major systems and structures to satisfy 50.34(a)(1). This will also be applicable to items relating to technical qualifications of the applicant and its management for design and construction. 57

b. Items for which the required information should be sufficient to assure that the requirement will be met at the final design stage. This is the kind of information and degree of detail customarily provided at the preliminary design stage with respect to the preliminary design of the facility to satisfy 50.34(a)(3)(4), etc."

RESPONSE

Responses to the Category 5 items are given herein.

ITEM I.C.5 PROCEDURES FOR FEEDBACK OF OPERATING, DESIGN AND CONSTRUCTION EXPERIENCE

NUREG 0718 REQUIREMENT

"Applicants shall subm t a description of their administrative procedures for the evaluation of operating, design, and construction experience and describe how they will assure that applicable importanc industry experiences originating from both within and outside the applicant's construction organization will be provided in a timely manner to those designing and constructing the plant. These procedures shall: (1) Clearly identify organization responsibilities for review and identification of these important experiences and the feedback of pertinent information to those responsible for designing and constructing the plant; (2) Identify the administrative and technical review steps necessary in implementing applicable important experie es; (3) Identify the recipients of various categories of information from these experiences or otherwise provide means through which such information can be readily related to the job functions of the recipients; (4) Assure that applicant and contractor personnel do not routinely receive extraneous and unimportant experience-related information in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency; (5) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to applicant and contractor personnel for implementation until resolution is reached; and (6) Provide practical interim audits to assure that the feedback program functions effectively at all levels. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of the construction permits or manufacturing license." 57

RESPONSE

HL&P, Ebasco and GE each have administrative procedures for the evaluation of operating, design and construction experience. The procedures of each company complement and overlap each other to assure that applicable industry experience is incorporated into ACNGS. The following is a description of those procedures. 59

1. Organizational Responsibilities

Operating, design, and construction experiences from outside HL&P are directed to the Nuclear Licensing Department. Within this department, there exists a generic licensing group that is responsible for reviewing the information received and identifying those experiences which may be of interest to ACNGS. The licensing group is also responsible for categorizing these experiences such that operating, design, and construction experiences are directed to the respective sub-organizations within HL&P for their review and use. 57

Operating, design, and construction experiences from within HL&P are directed to the ACNGS Project Manager, who is responsible for reviewing and categorizing the information, then directing the information to the ACNGS operations personnel, engineering team, or Construction Manager, as appropriate.

As the Applicant, HL&P has primary responsibility to assure that significant operating, design and construction experience is factored into AGNGS. However, both Ebasco and GE conduct activities independent of HL&P towards this end, as described in the next section. 59

2. Administrative and Review Steps and

3. Recipients of Information

a. General

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HL&P contracted for design and construction with Ebasco Services Incorporated and the General Electric Company, the principal contractors. As part of its responsibilities, the General Electric Company has, within its Nuclear Services Department, established and maintained a formal service advisory communication system that is designed to provide the BWR Owner-Operator with a broad coverage of BWR operating and maintenance information and recommendations. In addition, GE routinely reviews other available industry experience for applicability to the equipment and services it supplies to HL&P for ACNGS. Similarly, Ebasco reviews available industry experience for applicability to the design, construction and other activities it provides HL&P for ACNGS. Additionally, HL&P is responsible for advising both Ebasco and GE of operating, design, and construction experience uniquely available to HL&P, such as from South Texas Project design and construction, and from utility owners groups, in cases where that unique data is relevant to Allen's Creek. 59 57

b. Houston Lighting and Power Company

HL&P functions within the program to 1) review and approve Principal Contractors programs, 2) audit and monitor principal contractor implementation of their programs, 3) furnish data from a selected document list, including data uniquely available to HL&P, and 4) provide direction to Ebasco for incorporating and implementing design and construction experiences into the ACNGS design. 59

Operating, design, and construction experience information from external sources enters the HL&P program from two general categories: 1) Regulatory Agencies and 2) Industry Sources. Examples of documents reviewed are as follows: 57

1. Regulatory Agency Information

- License Event Reports
- Regulatory Guides
- Regulations (10CFR and 49CFR)
- IE Bulletins, Circulars, Orders and Notices
- NUREGS
- Standard Review Plans (including Branch Technical Positions)

2. Industry

- Topical Reports from General Electric and the Nuclear Safety Analysis Center
- IEEE, ANS, ANSI, and ASME Codes and Standards
- License Event Reports
- NSAC/INPO Significant Events Evaluation Information Network
- Owners Group Activities

As external information enters the HL&P system, it is directed to the Nuclear Licensing Department. There it is categorized, screened for applicability, and documented. The review at this stage is two-fold in purpose: 1) to reduce the quantity of information received to manageable amounts by culling out information clearly not relevant to ACNGS, and 2) to broadly categorize the information into operations, design, or construction categories. The Nuclear Licensing Department will then transmit the information to either the ACNGS Project Operations Team Leader, in the case of operational information, or the ACNGS Engineering Team Leader, in the case of design information or the ACNGS Construction Manager, in the case of construction information, along with a specified time by which disposition of the items must be fed back to Nuclear Licensing.

HL&P will provide continuous assessments of the efficacy of the experience feedback programs at HL&P and the principal contractors by using a commitment tracking system which will provide feedback to the Nuclear Licensing Department as to the ultimate resolution of the information that Nuclear Licensing sends out to the project. The same commitment tracking will be utilized for HL&P internally generated experience information. 57

In addition to the project organization, the information is sent to the line organization departments for information.

Operating Experience

Information on operating experience is received from the Nuclear Licensing Department by the ACNGS Operations Team Leader, who will perform a more detailed review of the information. This review will determine if the information is applicable to ACNGS and if it is of sufficient concern to pursue with the principal contractors. If warranted by the nature of the item, Operations will consult with the ACNGS Project Engineering Team and recommend a course of action. The Operation's Team Leader will obtain assistance as necessary from the ACNGS Engineering Team in his review.

In some cases, the appropriate action will be decided within HL&P, particularly if it is in plant maintenance or operations. The operational concern may then be resolved as part of the normal process of design, operator training, or procedures development.

Design Experience

For design experience, the ACNGS Engineering Team has the primary responsibility for resolving concerns once the information is received from the Nuclear Licensing Department. The Engineering Team Leader will review the information and direct it to the appropriate discipline leader for a determination of the necessary action. As with operating experience, the Discipline Leader may consult with either or both principal contractors to evaluate the concern. From this point, normal design control processes are used. 57

Construction Experience

For construction experience, the ACNGS Construction Manager has the primary responsibility for resolving concerns once the information is received from the Nuclear Licensing Department. He may use assistance from the ACNGS Engineering Team and either or both principal contractors, as appropriate. Construction concerns that affect plant design will be resolved in accordance with the project's normal design process.

c. General Electric

i. Formal Advisory Service 59

The GE-Nuclear Services Department maintains a formal service advisory communication system that is designed to provide the BWR Owner-Operator with a broad coverage of BWR operating and maintenance information and recommendations. This system, implemented by the Service Information Letter (SIL), is designed to collect, process, and disseminate information pertinent to:

1) unique operating conditions and experiences
2) improved methods, techniques and procedures for operating and maintaining BWR plant equipment
3) plant performance improvement and equipment upgrading
4) safety, licensing and other regulatory matters.

The major sources of information, including data, drawings, equipment, catalog/part numbers, problem definition, technical work recommendations, and other technical material required to prepare SILs include: 57

1) Application Information Documents (AIDs)
2) Field Engineering Memos (FEMs)
3) Product Experience Reports (PERs)
4) Safety and Licensing Reports
5) Reports and Instructions prepared by GE Engineering organizations
6) GE and Vendor Equipment Instruction Manuals
7) Equipment Failure and Reliability Reports
8) BWR Plant Owner-Operator(s) and utility management suggestions
9) Startup and Preoperational Test Reports

Occasionally, a need may arise to transmit to the utility owners with operating BWRs an urgent announcement of a potential operational hazard or other information which potentially could seriously impact plant operations. In general, such announcements will consist of a brief but adequate explanation of the situation with advice or precautionary measures to be observed.

Prior to release from GE-Nuclear Services Department, SILs will undergo 57
formal review by responsible design engineer, other cognizant engineers, and GE management representing various disciplines including engineering, startup tests, licensing, and services.

ii. NRC Information

Information received from the Nuclear Regulatory Commission falls into the following categories:

1) I&E Bulletins, Circulars and Information

2) NUREGs, Regulatory Guides and SRPs

I&E Documents are received by one individual within the GE licensing department, who reviews and routes it to the proper unit within the department. In turn, that particular unit will review and communicate with each project to which that information may be applicable.

NUREGs, Regulatory Guides and SRPs are received directly from the NRC distribution list by the following organizations within the GE licensing department:

a) Standardization 59

b) Operating Reactor Service

c) BWR Project Licensing

d) BWR System Licensing

e) Washington Liaison Office

Each organization reviews the documents received and cross communicate within to disseminate the data to the proper individual within the licensing organization. At that time all Project Managers are made aware of the information if the particular project is affected.

iii. Field Information

Within the General Electric Nuclear Division, all systems are assigned a Lead System Engineer with the prime responsibility for that particular system. If at any time, a problem is encountered in the field by the AE or GE field representatives, GE personnel will write a field deviation disposition report (FDDR) describing in detail the problems encountered. At the same time, that report may suggest a solution which is transmitted back to the GE Lead System Engineer in San Jose. That particular Lead Engineer will review the FDDR for its application. If it is a generic problem, an Engineering Change Authorization (ECA) will be written for review and approval. If the ECA is approved, then an engineering change notice (ECN) will be issued to all projects to correct the problem. If the Lead System Engineer finds that the problem is only applicable to a certain project, the same procedure described above will take place but only the specific project management will be notified. All ECN & ECA are transmitted to the ACNGS project at HL&P.

d. Ebasco Services, Incorporated

i. Feedback of internal experience into Ebasco Activities

The prime mechanism for assuring that experience from design and construction activities on Ebasco projects is factored into Ebascos' ongoing and future activities is the Corporate Feedback Program. This program may also be used as a means of factoring external experience into Ebasco activities.

Any individual within Ebasco may (and is encouraged to) submit feedback recommendations on items which can enhance ongoing and future activities. The feedback is formally submitted to the Department Head or designee, who will review the item within five working days. If the item is judged to be worthy of further action, it is referred to the appropriate Manager of Feedback Program, who will assign an engineering discipline with responsibility and followup to assure resolution. Resolution may take the form of revised Engineering Procedures, Department Technical Directives, etc. The feedback Administrator in the Standards and Procedures Department maintains a central record of all feedback items passed on for further action and their ultimate resolution. 59

ii. Feedback of operations considerations into Ebasco activities

Ebasco is not involved in the operation of power plants, but it is important that Ebascos' design and construction activities reflect operational considerations where applicable. This is accomplished informally through a number of means (the presence of former operations personnel on Ebasco design and construction staff, etc.) but is a formal responsibility of the Plant Operations and Betterment Department (PO&B).

A PO&B engineer is assigned to every project, including ACNGS, through the entire course of the project, from conceptual design to power operation. It is their responsibility to assure that plant systems are operable and maintainable. Sources of information to PO&B include participation in startup activities on Ebasco plants, troubleshooting and retrofitting on operating plants, Licensee Event Reports, industry contracts, INPO reports, etc. PO&B feeds this information back through the Corporate Feedback mechanisms where appropriate, and directly into the plant designs through the formal design review cycle on the project, topical meetings with department and project personnel, etc.

iii. Feedback of Other Experience into Ebasco Activities

Valuable design, construction and operating experience from the industry overall is available from a variety of sources, including:

1. NUREGs, Regulatory Guides, Standard Review Plans

2. INPO/NSAC Significant Operating Experience Reports

3. NRC I&E Bulletins, Circulars and Information Notices

4. Deficiency Reports (10CFR 50.55a, 10CFR21)

Ebasco receives these documents through direct distribution, NUS Corporations Licensing Information Service, the Atomic Industrial Forum, etc. The focal point for this information is the Nuclear Licensing Department (NL), which reviews it and distributes it to the appropriate Department Head(s) or designee(s), nuclear Project Engineers and Project Licensing Engineers within Ebasco. The Department Heads are responsible for reviewing the information for applicability to their departments activities, and, if necessary, for assuring that any necessary corrective actions are carried out. Further, the Department Heads distribute the information within their departments, including the discipline engineers assigned to the nuclear projects. 59

The discipline engineers on the nuclear projects review the information for applicability to their own projects activities design. After a preliminary assessment of the item, Ebasco notifies HL&P of the impact on ACNGS by letter or verbally. Significant items will be documented and upon HL&P authorization, further work is done on evaluation and/or incorporation of the item.

The Ebasco review of design, construction and operating experience is independent of that done by HL&P and GE, thus providing added assurance that the ACNGS design reflects significant industry experience.

4. The discipline engineers (engineers assigned to the ACNGS project) of General Electric, Ebasco and HL&P discuss and transmit information to each other on a day to day basis as a matter of good engineering practice. Information on Operating, Design and Construction experience is part of this informal process. This type of exchange further insures that feedback information is considered in project activities.

5. Avoidance of Extraneous and Unimportant Information

The Nuclear Licensing Department of each organization, through its normal screening process, will assure the avoidance of extraneous and unimportant information. 57

6. Avoidance of Conflicting or Contradictory Information 59

The Nuclear Licensing Department of each organization will assure that potentially conflicting or contradictory information is identified and 57 transmitted to the appropriate organization for resolution.

7. Practical Interim Audits 59

HL&P will ensure compliance with these requirements by monitoring and 57 performing technical reviewing of General Electric and Ebasco activities to verify that significant information is being properly distributed within the organizations. HL&P Nuclear Licensing will perform the reviewing of General Electric and Ebasco. HL&P Engineering Management is 59 responsible to review the performance of HL&P Nuclear Licensing.

ITEM I.F.1 EXPAND QA LIST

"Prior to issuance of the construction permits or manufacturing license, applicants shall revise their QA programs by expanding their QA lists to include all items and activities affecting safety as defined by Regulatory Guide 1.29 and Appendix A to 10 CFR Part 50, and shall provide a commitment to apply the revised QA program to all such items and activities." 57

RESPONSE

HL&P has expanded its interpretation of "important to safety" to include the three existing quality categories of "safety related," "fire protection," and "radwaste," and a new fourth category named "safety significant," which is described below. To avoid confusion and unnecessary retrofitting of existing plant documents these systems will continue to be referred to by the appropriate one of these four quality categories, but it is understood that all are considered important to safety as defined by Regulatory Guide 1.29 and 10CFR50 Appendix A.

Ebasco Project Engineering is responsible for preparation and maintenance of the "Q-list", i.e., the list of drawings and specifications which are important to safety. Each revision of the Q-list contains the issue date, approval date or revision and authorized signatures. Ebasco recommends inputs to the Q-list except for items that fall within the GE scope of work, for which inputs to the Q-list are recommended by GE. Changes to the Q-list are reviewed and approved by the HL&P Project Engineering Manager.

The criteria for designation of an item within the quality categories of important to safety and the applicable QA program are given below (the previously recognized QA categories are simply re-capped, as they are well understood and are described elsewhere in the PSAR in detail).

- Safety Related: These are the items which have traditionally been considered as safety related in accordance with Section 3.2. Generally, these are systems which: 59

  o are part of the reactor coolant pressure boundary

  o are intended to directly perform an ESF function such as reactor vessel makeup (e.g, High Pressure Core Spray) or post-accident fission product control necessary to the 10CFR100 dose criteria (e.g, Containment Isolation).

  o support those systems which directly perform an ESF function (e.g, RHR Service Water or ECCS Equipment Area Cooling).

The QA program used for safety systems is described in Chapter 17, and is in conformance with 10CFR50 Appendix B.

- Fire Protection: Fire protection systems are given their own QA category. The QA program used for fire protection systems is described in Appendix 9.5-1A and meets NRC BTP APCSB 9.5-1, Section C.

- Radwaste Systems: Radwaste systems are given their own QA category. The QA program used for radwaste systems is described in Section 11.2.3, and meets the requirements of ETSB 11-1.

- Safety Significant: Items are determined to be safety significant by systems analysis techniques as follows:

(A) Develop a List of "Unacceptable Safety Results"

This list is a matter of judgement plus regulatory requirement and represents an extension of the general intent of station hardware design criteria. A set of unacceptable safety results, similar to the examples below, will be developed for each major category of station events. Examples:

	EVENT CATEGORY	UNACCEPTABLE SAFETY RESULTS
(1)	Planned Operations	The release of radioactive material to the environs to such an extent that the limits of 10CFR20 are exceeded. 59
(2)	Transients and Anticipated Operational Occurrences	The release of radioactive material to the environs to such an extent that the limits of 10CFR20 are exceeded.

Accidents are evaluated under the category of safety related.

(B) Identify and Define the Physical States (Operating States) in Which the Plant May Exist

This step will be designed to divide the plant operating spectrum into a few major conditions to facilitate consideration of various events in each state.

(C) Types of Operation and Events Applicable to Each Operating State

(1) Examples of planned operations are refueling outage, power operation, and achieving shutdown.

(2) Accidents and transients are defined as those postulated events analyzed in Chapter 15.

(D) Identification of Safety Actions

It is at this point that sequence analysis will be started. A safety action is an ultimate action in the station which is essential to the avoidance of specified conditions considered to be of primary safety significance (Unacceptable Safety Results).

Once the above analyses are completed, designation of system as safety significant is determined by the following steps:

(A) Establish system boundaries.

(B) Determine for what abnormal transient or accident the system is required.

(C) Determine which components are required to allow system to satisfy its functions which are important to safety. 59

(D) Designate an item as Active (Auto or Manual) or Passive (pressure retention/structural considerations).

The QA program with the applicable portions of 10CFR50 Appendix B will be developed for safety significant items. After the system is designated as safety significant, the OA program will be applied to all subsequent system design, procurement, construction and operations activities.

ITEM I.F.2 DEVELOP MORE DETAILED QA CRITERIA

NUREG 0718 REQUIREMENT

Applicants shall describe the changes to their QA programs that have resulted from their review of the accident at TMI-2. In addition, applicants shall address the appropriate matters discussed in this Action Plan item and the extent to which they have been considered in their QA program. Applicants shall submit, prior to the issuance of the construction permits or manufacturing license, a revised description of their QA program that includes consideration of these matters.

PROPOSED 10CFR50.34(e)(3)(iii)

Establish a quality assurance (QA) program based on consideration of:

(A) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions;

NRC ACCEPTANCE GUIDANCE

The QA program includes:

2A1 (1B2) Verification of conformance to established requirements is accomplished by individuals or groups within the QA organization who do not have direct responsibility for performing the work being verified. Rationale and justification must be provided if performed by other than the QA organization. 57

2A2 (10B1) The QA organizational responsibilities for inspection are described. Individuals performing inspections report to the QA organization.

2A3 (7A2) Verification of suppliers' activities during fabrication, inspection, testing, and shipment of materials, equipment, and components is planned and performed with QA organization participation in accordance with written procedures to assure conformance to the purchase order requirements. These procedures, as applicable to the method of procurement, provide for:

a. Specifying the characteristics or processes to be witnessed, inspected, or verified, and accepted; the method of surveillance and the extent of documentation required; and those responsible for implementing these procedures.

b. Audits, surveillance, or inspections which assure that the supplier complies with the quality requirements.

2A4 (7B1) Receiving inspection is performed by the QA organization to assure:

a. The material, component, or equipment is properly identified and corresponds to the identification on the purchase document and the receiving documentation.

b. Material, components, equipment, and acceptance records satisfy the inspection instructions prior to installation or use.

c. Specified inspection, test and other records, (such as certificates of conformance attesting that the material, components, and equipment conform to specified requirements) are available at the nuclear power plant prior to installation or use.

2A5 (8B3) Correct identification of material, parts, and component is verified and documented by the QA organization prior to release for fabrication, assembling, shipping, and installation.

2A6 (9B2) Procedures are established for recording evidence of acceptable
accomplishment of special processes using qualified procedures, equipment, and personnel. The QA organization verifies the recorded evi-
dence and documents the result. 57

2A7 (10C3) (11C1) Inspection and test results are documented, evaluated, and their acceptability determined by a responsible individual or group. The QA organization as a minimum evaluates, verifies, and documents completeness of this activity.

2A8 (16.3) Follow-up action is taken by the QA organization to verify proper implementation of corrective action and to close out the corrective action in a timely manner.

RESPONSE (HL&P and Ebasco) 59

2A1 Verification of conformance to established requirements is accomplished
by personnel within the QA organization. These individuals do not have
direct responsibility for performing the work being verified. Chapter 57
17.1.1 of the Allens Creek PSAR describes the Ebasco and the HL&P QA
organization. Figure 17.1.1.A-3 shows the HL&P project QA organization
and Figure 17.0.B-1 shows the Ebasco QA organization. 59

2A2 The Ebasco QC organization reporting to the Quality Program Site Manager is responsible for the performance of inspection.

2A3 The Project QA/QC organization will perform the following activities:

a. Planning, performance, and verification of suppliers' activities
during fabrication, inspection, testing, and shipment of materials,
equipment, and components in accordance with written procedures or
vendor inspection plans to assure conformance to the purchase order 57
requirements. These procedures or vendor inspection plans, as
applicable to the method of procurement provide for:

1) Specifying the characteristics or processes to be witnessed, inspected, or verified, and accepted; the method of surveillance and the extent of documentation required; and those responsible for implementing these procedures.

2) Audits, surveillance, or inspections which assure that the supplier complies with the quality requirements.

2A4 b. Receiving inspection to assure:

1) The material, component, or equipment is properly identified and corresponds to the identification on the purchase document and the receiving documentation.

2) Material, components, equipment, and acceptance records satisfy the inspection instruction prior to installation or use.

3) Specified inspection, test, or other records (such as certificates of conformance attesting that the material, components, and equipment conform to specified requirements are available at the nuclear power plant prior to installation or use.

2A5 c. Correct identification of material, parts, and components is verified and documented prior to release for fabrication, assembling, shipping, and installation.

2A6 d. Procedures are established for recording evidence of acceptable accomplishment of special processes using qualified procedures, equipment, and personnel. The QA organization verifies the recorded evidence and documents the result.

2A7 e. Inspection and test results are documented, evaluated, and their acceptability determined by a responsible individual or group. The QA organization, as a minimum, evaluates, verifies, and documents 57
completeness of this activity.

2A8 f. Follow-up action is taken by the QA organization to verify proper implementation of corrective action and to close out the corrective action in a timely manner.

PROPOSED 10CFR50.34(e)(3)(iii)

(B) performing the entire quality assurance/quality control function at construction sites;

NRC ACCEPTANCE GUIDANCE

The QA program provides provisions to assure that:

2B1 (1C3) The person at the construction site responsible for directing and managing the site QA program is identified by position. He reports to the offsite QA organization and has appropriate organizational position, responsibilities, and authority to exercise proper control over the QA program. This individual is free from non-QA duties and can thus give full attention to assuring that the QA program at the plant site is being effectively implemented.

2B2 (1B6) Designated QA individuals are involved in day-to-day plant activities important to safety (i.e., the QA organization routinely attends and participates in daily plant work schedule and status meetings to assure they are kept abreast of day-to-day work assignments throughout the plant and that there is adequate QA coverage relative to procedural and 57 inspection controls, acceptance criteria, and QA staffing and qualification of personnel to carry out QA assignments).

RESPONSE (HL&P and Ebasco) 59

2B1 The HL&P Project Quality Assurance Manager shall be located at the Construction site and is responsible for directing and managing the site QA program. He is free from non-QA duties. The HL&P Project Quality Assurance Manager is responsible for providing the programmatic direction and administering the policies, goals, objectives, and methods for the Allens Creek Project which are described in the Project Quality Assurance Plan. Programmatic direction is defined as the role of HL&P in establishing the program requirements and ensuring the adequacy of the quality assurance program for HL&P and the prime contractors. The Project Quality Assurance Manager reports to the Manager, Quality Assurance, who reports directly to the Executive Vice President and has the independent authority to identify quality related problems, to initiate or recommend solutions, to control existing nonconformances, to verify implementation of approved dispositions, and when necessary to stop work.

2B2 Project QA personnel are involved in plant activities important to safety and are kept abreast of work schedule and construction activities by periodically attending construction status meetings. Project 57 QA personnel ensure that there is adequate QA coverage relative to procedural and inspection controls, acceptance criteria, and QA staffing and qualification of personnel to carry out assignments.

PROPOSED 10CFR50.34(e)(3)(iii)

(C) including QA personnel in (the review and concurrence) of quality-related procedures (and documents) associated with design, construction, and installation;

NRC ACCEPTANCE GUIDANCE

The QA program includes:

2C1 (2B1a) Provisions are established to assure that quality-affecting procedures required to implement the QA program are consistent with QA program commitments and corporate policies and are properly documented, controlled, and made mandatory through a policy statement or equivalent document signed by the responsible official.

2C2 (2B1b) The QA organization reviews and documents concurrence with these quality-related procedures.

2C3 (4A1) Procedures are established for the review of procurement documents to determine that quality requirements are correctly stated, inspectable, and controllable; there are adequate acceptance and rejection criteria; and procurement documents have been prepared, reviewed, and approved in accordance with QA program requirements. To the extent necessary, procurement documents should require contractors and subcontractors to provide an acceptable quality assurance program. The review and documented concurrence of the adequacy of quality requirements stated in procurement documents is performed by QA personnel.

2C4 (6A2) Procedures for the review, approval, and issuance of documents and changes thereto are established and described to assure technical adequacy and inclusion of appropriate quality requirements prior to implementation. The QA organization reviews and documents concurrence with these documents with regards to QA-related aspects.

2C5 (10C1) Inspection procedures, instructions, or checklists provide for the following as reviewed and concurred with by the QA organization for QA aspects and other technical organizations, as appropriate:

a. Identification of characteristics and activities to be inspected.

b. A description of the method of inspection. 57

c. Identification of the individuals or groups responsible for performing the inspection operation in accordance with the provisions of item 10B1.

d. Acceptance and rejection criteria.

e. Identification or required procedures, drawings, and specifications and revisions.

f. Recording inspector or data recorder and the results of the inspection operation.

g. Specifying necessary measuring and test equipment including accuracy requirements:

2C6 (11B1) Test procedures or instructions provided for the following as reviewed and concurred with by the QA organization for QA aspects and by other technical organizations for technical aspects:

a. The requirements and acceptance limits contained in applicable design and procurement documents.

b. Instructions for performing the test.

c. Test prerequisites such as calibrated instrumentation, adequate test equipment and instrumentation including their accuracy requirements, completeness of item to be tested, suitable and controlled environmental conditions, and provisions for data collection and storage.

d. Mandatory inspection hold points for witness by owner, contractor, or inspector (as required).

e. Acceptance and rejection criteria.

f. Methods of documenting or recording test data and results.

g. Provisions for assuring test prerequisites have been met.

2C7
(12.3) Procedures are established and described for calibration (technique and frequency), maintenance, and control of the measuring and test equipment (instruments, tools, gages, fixtures, reference and transfer standards, and nondestructive test equipment) that is used in the measurement, inspection, and monitoring of structures, systems, and components. The review and documented concurrence of these procedures is described and the organization responsible for these functions is identified.

2C8
(13.2) Procedures are established and described to control the cleaning, handling, storage, packaging, and shipping of materials, components, and systems in accordance with design and procurement requirements to preclude damage, loss, or deterioration by environmental conditions such as temperature or humidity. The QA organization reviews and documents concurrence of these procedures.

2C9
(14.1)
(14.4) Procedures are established to indicate the inspection, test, and operating status of structures, systems, and components and throughout fabrication, installation, and test. The QA organization reviews and documents concurrence with these procedures. 57

2C10
(14.2)
(14.4) Procedures are established and described to control the application and removal of inspection and welding stamps and status indicators such as tags, markings, labels, and stamps. The QA organization reviews and documents concurrence with these procedures.

2C11
(14.3)
(14.4) Procedures are established and described to control altering the sequence of required tests, inspections, and other operations important to safety. Such actions should be subject to the same controls as the original review and approval. The QA organization reviews and documents concurrence with these procedures.

2C12
(15.1) Procedures are established and described for identification, documentation, segregation, review, disposition, and notification to affected organizations of nonconforming materials, parts, components, and as applicable to services (including computer codes) if disposition is other than to scrap. The procedures provide identification of authorized individuals for independent review of nonconformance, including disposition and closeout.

2C13 (15.2) QA and other organizational responsibilities are described for the definition and implementation of activities related to nonconformance control. This includes identifying those individuals or groups with authority for the disposition of nonconforming items and involvement of the QA organization in documenting concurrence to the disposition, satisfactory completion of the disposition, and corrective action.

2C14 (16.1) Procedures are established and described indicating an effective corrective action program has been established. The QA organization reviews and documents concurrence with the procedures. 57

RESPONSE (HL&P and Ebasco) 59

2C1 2C2 The HL&P Project Quality Assurance Program for the Allens Creek Nuclear Generating Station is described by the HL&P Project Quality Assurance Plan (PQAP). A letter signed by the Executive Vice President in the front of the PQAP makes the requirements of the PQAP mandatory. Procedures are reviewed by project QA personnel during preparation for inspections, surveillance, implementation reviews and audits to ensure consistency with project requirements. Additionally, selected procedures are reviewed and concurred with by the project QA organization prior to issuance.

2C3 Procedures are established for the review of procurement documentation by project QA personnel to determine that 1) Quality requirements are correctly stated, inspectable, and controllable; 2) there is adequate acceptance and rejection criteria and; 3) that procurement documents have been prepared, reviewed, and approved in accordance with QA program requirements. To the extent necessary, procurement documents will require contractors and subcontractors to provide an acceptable quality assurance program.

2C4 Procedures for the review, approval and issuance of documents (including procedures, instruction, specifications, and construction drawings) and changes thereto are established and described to assure technical adequacy and inclusion of appropriate quality requirements prior to implementation. Selected documents are reviewed and concurred with by the project QA organization for Quality Assurance related aspects. 57

2C5 Inspection procedures, instructions, or checklists are developed by QA, reviewed by other technical organizations as necessary, and issued by QA for the following as appropriate:

a. Identification of characteristics and activities to be inspected.

b. A description of the method of inspection.

c. Identification of the individuals or groups responsible for performing the inspection operation.

d. Acceptance and rejection criteria.

e. Identification of required procedures, drawings, and specifications and revisions.

f. Recording inspection or data recorder and the results of the inspection operation. 57

g. Specifying necessary measuring and test equipment including accuracy requirements.

2C6 Test procedures or instructions are developed by QA, reviewed by other technical organizations as necessary, and issued by QA for the following as appropriate:

a. The requirements and acceptance limits contained in applicable design and procurement documents.

b. Instructions for performing the test.

c. Test prerequisites such as calibrated instrumentation, adequate test equipment, and instrumentation including their accuracy requirements, completeness of item to be tested, suitable and controlled environmental conditions, and provisions for data collection and storage.

d. Mandatory inspection hold points for witness by owner, contractor, or inspector (as required).

e. Acceptance and rejection criteria.

f. Methods of documenting or recording test data and results.

g. Provisions for assuring test prerequisites have been met. 57

The QA organization reviews and documents concurrence with the procedures developed to control the following:

2C7 a. Calibration (technique and frequency), maintenance and control of the measuring and test equipment that is used in the measurement inspection, and monitoring of structures, systems, and components.

2C8 b. Cleaning, handling, storage, packaging, and shipping of materials, components, and systems in accordance with design and procurement requirements to preclude damage, loss, or deterioration by environmental conditions such as temperature or humidity.

2C9 c. Systems for the indication of the inspection test and operating status of structures, systems, and components throughout fabrication, installation, and test.

2C10 d. Application and removal of inspection and welding stamps and status indicators such as tags, markings, labels, and stamps.

2C11 e. Altering the sequence of required tests, inspections, and other operations important to safety. Such actions should be subject to the same control as the original review and approval.

2C12 Responsibilities for QA and other organizations are described in project procedures for the definition and implementation of activities for nonconformance control including the identification of individuals or groups with authority for the disposition of nonconforming items and involvement of Project QA in documenting concurrence to the disposition, satisfactory completion of the disposition, and corrective

2C13 action. Procedures are established and described for identification, documentation, segregation, review, dispositions, and notification to affected organizations of nonconforming materials, parts, components, and as applicable to services (including computer codes) if disposition is other than to scrap. The procedures provide identification of authorized individuals for independent review of nonconformances,
2C14 including disposition and closeout. QA develops procedures for an effective corrective action program.

Proposed 10 CFR 50.34 (e) (3) (iii)

(D) establishing criteria for determining QA requirements for specific classes of equipment;

NRC Acceptance Guidance

The QA program provides provisions to assure that:

2D1 (2B3) The QA organization and the necessary technical organizations participate early in the QA program definition stage to determine and identify the extent QA controls are to be applied to specific structures, systems, and components. This effort involves applying a defined graded approach to certain structures, systems, and components in accordance with their importance to safety and affects such disciplines as design, procurement, document control, inspection tests, special processes, records, audits and others described in 10 CFR 50 Appendix B. 57

2D2 (7B4) For commercial "off-the-self" items where specific quality assurance controls appropriate for nuclear applications cannot be imposed in a practicable manner, special quality verification requirements shall be established and described to provide the necessary assurance of an acceptable item by the purchaser.

2D3 (10A) The scope of the inspection program is described that indicates an effective inspection program has been established. Program procedures provide criteria for determining the accuracy requirements of inspection equipment and criteria for determining when inspections are required or define how and when inspections are performed. The QA organization participates in the above functions.

2D4 (10C2) Procedures are established and described with the involvement of the QA organization to identify, in pertinent documents, mandatory inspection hold points beyond which work may not proceed until inspected by a designated inspector.

2D5 (11A1) The description of the scope of the test control program indicates an effective test program has been established for tests including proof tests prior to installation and preoperational test. Program procedures provide criteria for determining the accuracy requirements of test equipment and criteria for determining when a test is required or how and when testing activities are performed.

2D6 (1881) Audit data is analyzed by the QA organization and the resulting reports indicating any quality problems and the effectiveness of the QA program, including the need for reaudit of deficient areas, are reported to management for review and assessment. 57

RESPONSE (HL&P and Ebasco) 59

2D1 The project QA organization and the necessary technical organization participate early in the QA program definition stage to determine and identify the extent QA controls are to be applied to specific structures, systems, and components. For items determined to be important to safety where specific QA Controls cannot be imposed in a practical manner, an evaluation will be made to determine special quality verification requirements to be applied during installation or testing to provide the necessary assurance that the item(s) meet project requirements.

2D3 The project QA organization participates in the definition of the scope of the inspection program. Procedures provide criteria for determining the accuracy requirements of inspection equipment and criteria for determining when inspections are required or define how and when inspections are performed.

2D4 Procedures are established to identify in pertinent documents, mandatory inspection hold points beyond which work may not proceeed until inspected by a designated inspector.

2D5 A test control program will be established to include proof tests prior to installation and preoperational tests. Procedures provide criteria for determining accuracy requirements of test equipment and criteria or determining when a test is required and how and when testing activities are performed. 57

2D6 Audits are conducted and the results analyzed by QA. Audit reports indicate any quality problems and the effectiveness of the audited QA Program. Reaudits of deficient areas are conducted as necessary to assume implementation of corrective action and recurrence control. Audit results are reported to management for review and assessment.

Proposed 10 CFR 50.34(e)(3)(iii)

(E) establishing minimum qualification requirements for QA and QC personnel;

NRC Acceptance Guidance

The QA program provides provisions to assure that:

2E1 (2D) Indoctrination, training, and qualification programs are established such that:

a. Personnel responsible for performing quality-affecting activities are instructed as to the purpose, scope, and implementation of the quality-related manuals, instructions, and procedures.

b. Personnel verifying activities affecting quality are trained and qualified in the principles, techniques, and requirements of the activity being performed.

c. For formal training and qualification programs, documentation includes the objective, content of the program, attendees, and date of attendance.

d. Proficiency tests are given to those personnel performing and verifying activities affecting quality, and acceptance criteria are developed to determine if individuals are properly trained and qualified.

e. Certificate of qualifications clearly delineates (a) the specific functions personnel are qualified to perform and (b) the criteria 57
used to qualify personnel in each function.

f. Proficiency of personnel performing and verifying activities affecting quality is maintained by retraining, re-examining, and/or recertifying as determined by management or program commitment.

g. The description of the training program provisions listed above satisfies the regulatory position in Regulatory Guide 1.58, Rev. 1.

2E2 (10B2) A qualification program for inspectors (including NDT personnel) is established under direction of the QA organization and documented, and the qualifications and certifications of inspectors are kept current.

RESPONSE (HL&P and Ebasco) 59

2E1 The training qualification and certification programs are established so that:

a. Personnel responsible for performing quality affecting activities are instructed as to the purpose, scope, and implementation of the quality related manuals, instructions, and procedures. 57

b. Personnel verifying activities affecting quality are trained and qualified in the principles, techniques, and requirements of the activity being performed.

c. For formal training and qualification programs, documentation includes the objective, content of the program, attendees, and date of attendance.

d. Proficiency tests are given to those personnel performing and verifying activities affecting quality, and acceptance criteria are developed to determine if individuals are properly trained and qualified.

e. Certificate of qualifications clearly delineates (1) the specific functions personnel are qualified to perform, and (2) the criteria used to qualify personnel in each function.

f. Proficiency of personnel performing and verifying activities affecting quality is maintained by retraining, re-examining, and/or recertifying as determined by management or program commitment.

g. The description of the training program provisions listed above satisfies the regulatory position in Regulatroy Guide 1.58, Rev. 1.

2E2 Personnel performing quality control functions at the site and at vendor facilities are qualified in accordance with ANSI-N45.2.6.

Proposed 10 CFR 50.34(e)(3)(iii) 57

(F) sizing the QA staff commensurate with its duties, responsibilities, and importance to safety.

NRC Acceptance Guidance

The QA program provides provisions to assure that:

2F1 (1A5) Organization charts identify the "onsite" and "offsite" organizational elements which function under the cognizance of the QA program (such as design engineering, procurement, manufacturing, construction, inspection, test, instrumentation and control, nuclear engineering, etc.), the lines of responsibility, and a description of the criteria for determining the size of the QA organization including the inspection staff.

2F2 (-) The QA organization is involved in establishing long range projected work schedules and staffing of QA and QC personnel and evaluates these periodically (i.e., monthly) to assure they are valid or if necessary modify staffing level.

RESPONSE (HL&P and Ebasco) 59

2F1 Figures 17.1.1A-3 and 17.0.B-1 show the project QA organization and indicate which personnel are "onsite" and "offsite". The PSAR Section 13.0, shows project personnel from other organizations. The criteria for determining staffing for the QA organization includes:

a. Establishing the number of QA/QC personnel based upon the project schedule to ensure that personnel are available, qualified, and certified to perform quality related inspections and evaluations.

b. Establishing the need for specially qualified QA/QC personnel based upon the schedule for activities requiring special or unusual expertise as far in advance of the activity as possible. 57

c. Establishing the number of QA personnel based upon the number and criticality of problems identified during routine activities in order to perform additional or supplemental inspections, reviews, or evaluations as required to ensure implementation of project requirements.

2F2 Staffing projections are periodically reviewed based upon the project schedule and are re-reviewed and revised, as necessary, as the project schedule changes. QA management personnel participate in short and long range scheduling activities. Staffing levels for QA/QC are a prime consideration in determining the level of effort for quality related activities. Prior to allowing quality related activities to be conducted, adequate numbers of qualified QA/AC personnel must be available. Adequate QA/QC staffing must be available to prevent QA/QC personnel from being required to perform inspections or evaluations without adquate preparation time or under pressure to complete inspections within a scheduled time period. Adequate QA/QC staff must be available to allow for prompt closeout of opern nonconformances and proper followup to ensure corrective action has been taken. 57

Proposed 10 CFR 50.34(e)(3)(iii)

(G) establishing procedures for maintenance of "as-built" documentation;

NRC Acceptance Guidance

The QA program provides provisions to assure that:

2G1 (6A1) The scope of the document control program is described, and the types of controlled documents are identified. As a minimum, controlled documents include: As built documents.

2G2 (6C1) Procedures are established and described to provide for the preparation of as-built drawings and related documentation in a timely manner to accurately reflect the actual plant design.

RESPONSE (HL&P and Ebasco) 59

2G1
2G2 The PSAR, Section 17.1.6A, includes "As-Built" drawings in the document control system. Project procedures will be developed to ensure that drawings are provided to indicate the as-built configuration. The as-built drawings will stand alone and delineate actual location - elevation, azimuth, etc.; actual component identification or numbering; dimensions and other relevant information. When changes occur subsequent to issuance of as-built drawings, procedures will require a re-review and re-issue of the drawings.

Proposed 10 CFR 50.34(e)(3)(iii)

(H) providing a QA role in design and analysis activities. 57

NRC Acceptance Guidance

The QA program provides provisions to assure that:

2H1 (3E1) Procedures are established and described requiring a documented check to verify the dimensional accuracy and completeness of design drawings and specifications.

2H2 (3E2) Procedures are established and described requiring that design drawings and specifications be reviewed by the QA organization or other individuals knowledgeable and qualified in QA/QC techniques to assure that the documents are prepared, reveiwed, and approved in accordance with company procedures and that the documents contain the necessary quality assurance requirements such as inspection and test requirements, acceptance requirements, and the extent of documenting inspection and test results. 57

RESPONSE (HL&P and Ebasco) 59

2H1 Procedures require a documented check to ensure the dimensional accuracy (including tolerance for accept/reject criteria and inspectability) and the completeness of the drawings and specifications. QC inspections of quality related activities will be conducted using procedures or inspection checklists developed from the engineering specifications and drawings for the system, component, or structure. 57

2H2 Procedures are established to require that design drawings and specifications be reviewed by individuals knowledgeable and qualified in QA/QC techniques to assure that the documents are prepared, reviewed, and approved in accordance with procedures and that documents contain the necessary QA requirements such as inspection and test requirements, acceptance requirements, and the extent of documenting inspection and test results.

ITEM II.B.8 RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

NUREG 0718 REQUIREMENT

"Applicant shall:

(2) include provisions in the containment design for one or more dedicated penetrations, equivalent in size to a single 3-foot diameter opening. This shall be done in order not to preclude the installation of systems to prevent containment failure, such as filtered vented containment systems."

RESPONSE

A dedicated three (3) foot diameter containment penetration assembly below the spring line will be incorporated into the plant design. The space provision to reflect the penetration assembly is available in the present containment design. The dedicated 3-foot diameter containment penetration assembly will consist of a capped penetration in the Steel Containment and the Shield Building. The Elevator Access Room is dedicated for the containment penetration assembly, and allows space for a future outboard isolation valve, (space provision only). Space inside the Containment is dedicated for the containment penetration assembly, and a future inboard isolation valve, if required (space provision only). 57

The penetration assembly and the welded caps will be designed in accordance with the requirements of the ASME Section III, Subsection NE, and seismic Category I. The penetration assembly and the welded caps shall be protected from natural phenomena in the same manner as the Containment Steel Shell.

Periodic tests and inspection will be performed in accordance with the normal plant operation procedure. Test connections as per 10CFR50, Appendix J, shall be provided.

A detailed description of this provision is located in Section 3.8.2.1.2.

ITEM II.B.8 <u>RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS</u>

<u>NUREG 0718 REQUIREMENT</u>

"Applicant shall:

(4) provide preliminary design information at a level consistent with that normally required at the construction permit stage of review sufficient to demonstrate that:

(a) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Division 1, Subsubarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required,
considering pressure and dead load alone. For concrete containments 57
by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Division 2, Subsubarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent, depending upon which option is chosen for control of hydrogen. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions."

<u>RESPONSE</u>

A preliminary evaluation has been done using the current (including modifications described before ACRS on 2/6/81) configuration of the containment vessel and its anchorage into the foundation mat. The evaluation results show that containment integrity will be maintained by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Division 1, Subsubarticle NE-3220, Service Level C Limits for containment vessel Division 2,
Subsubarticle CC-3420, Factored Load Category for concrete anchorage and 59
Division 2, Subsubarticle CC-3720, Factored Load Category for bottom liner plate (considering pressure (45 psig) and dead load alone) during an accident
that releases hydrogen generated from 100% active fuel clad metal-water 57
reaction and the pressure from post-accident inerting assuming carbon dioxide
is the inerting agent. 59

In order to assure containment integrity for the required conditions, the applicant has revised the commitments found in Section 3.8 as follows:

(i) Additional loads have been defined in Sections 3.8.2.3.1 and 3.8.5.3.1. 57
These loads were established in the following manner: for the accident case the static pressure was taken as the greater of the maximum pressure determined in the accident analysis described later in this response and the minimum 45 psig of this requirement. The temperature for this case was taken from the accident analysis.

(ii) An additional load combination has been specified in Sections 3.8.2.3.2 and 3.8.5.3.2. This load combination includes the consideration of dead weight, pressure, and temperature as given in (a) above.

A complete description of the containment vessel design and analysis procedures is provided in Section 3.8.2.4. In summary, the containment vessel will be designed in accordance with the rules in ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE for Class MC Components and will consider the loading conditions listed in Section 3.8.2.3.2. The vessel shell will be analyzed using the basic membrane equations for thin shells. The anchorage transition region will be analyzed for the same loading 57
conditions indicated above. The temperature stresses will be analyzed in critical areas, with the rules of ASME, Section III governing the treatment of these stresses as either secondary or local.

A complete description of the design and analysis procedures for the concrete foundation of the containment vessel is provided in Section 3.8.5.4. In summary, the design and analysis of the anchorage region concrete will be in accordance with the appropriate sections of CC-3100 to CC-3500 of the ACI-ASME Code, Section III, Division 2 using the load conditions listed in Section 3.8.5.3.2. The analysis will be performed by conventional stiffness/flexibility computerized methods using proven industry accepted computer programs to determine the internal stresses and deformations in the anchorage region. Adequate reinforcing will be provided to resist the forces and moments resulting from the different loading conditions. The thermal stresses will be determined by the use of temperature gradients through the thickness of the 59
anchorage region for the different loading conditions. The rules in ACI-ASME Code governing the treatment of thermal stresses as secondary stresses will be used.

For the additional accident loading condition shown in Section 3.8.2.3.2, the acceptance criteria for the containment vessel will be based on the allowable stresses defined in Table 3.8-1 for abnormal extreme load combinations 57
(integral and continuous). These allowable stresses are based on the requirements of the ASME Code, Section III, Division 1, Subsection NE and are as follows:

- General membrane stresses: the greater of 1.2 S_m or S_y
- Local membrane stresses: the greater of 1.8 S_m or 1.5 S_y
- Bending + local membrane stresses: the greater of 1.8 S_m or 1.5 S_y

59

The anchorage concrete and the bottom liner will be designed for the effects of the additional accident loading condition shown in Section 3.8.5.3.2 utilizing the allowable stresses and strains for factored loads, based on the requirements of the ACI-ASME Code. For the additional accident loading condition the stress and strain limits using Factored Load Category will be as 57
follows:

- Concrete compression, shear, torsion, bearing; use allowable stresses for factored loads as specified in the ACI-ASME Code, Paragraph CC-3421.

- Reinforc'ng steel tension and compression: use allowable stresses and strains for factored loads as specified in the ACI-ASME Code, Paragraph CC-3422. 57

- Bottom Liner: Use allowable strains for factored loads as specified in the ACI-ASME Code, Subsubarticle CC-3720. 59

ACCIDENT ANALYSIS

The assumptions below were used to calculate containment temperature and pressure following an accident which releases an amount of hydrogen equivalent to 100% of the active fuel clad metal-water reaction. It should be recognized that like the assumed 100% reaction, these assumptions are not intended to have a mechanistic basis but are prepared as a means to define a reasonable method of determining the containment pressure and temperature response. They should be considered in the context of the overall objective of providing assurance that an extensive metal water reaction can be accommodated in the absence of definitive accident specifications. In the future, pending rulemaking on degraded core may provide those specifications.

Initiating Event: a transient-event (such as loss of feedwater) followed by reactor scram and containment isolation. 57

Assumptions

- Inadequate core cooling results in decreasing Reactor Pressure Vessel (RPV) water level

- Operator initiates post-inerting system timer on reactor low level 1

- CO_2 discharge is complete 45 minutes after event initiation.

- core heatup results in 100% MWR of active fuel clad

- the energy equivalent to decay heat and 100% MWR is transported to the suppression pool

- temperature differential between pool and containment airspace at 42 psig is 50F. The 50°F ΔT between the pool and airspace is based on the results from an analytical model of the containment. This model calculates long term containment pressure and temperature transients based on mass and energy balances.

  temporary loss of core cooling accident was modeled assuming [illegible]ervative heat and mass transfer between airspace, suppression pool and containment structures. A heat transfer coefficient from the containment walls to the airspace was obtained assuming free convection across a vertical plate. This resulted in a heat transfer coefficient of .71 BTU/hr-F-ft^2. 59

  A heat transfer coefficient from the suppression pool to the airspace was obtained assuming a heated plate facing upward. This resulted in a heat transfer coefficient of $h_c = .22\ (\Delta T)^{1/3}$.

Finally a mass transfer coefficient from the suppression pool to the airspace was obtained using the above heat transfer coefficient and a diffusion coefficient of .99 ft^2/hr (ref. 1 page 593.) This resulted in a mass transfer coefficient of 53.7 ft/hr.

Initial Conditions:

The following initial conditions were used in the evaluation of this temperature difference. 59

$V = 1,217,707\ ft^3$ (containment only)
Rel. Hum. = 100%
$T = 95^oF$

The results of this analysis justified a minimum ΔT of 50^oF.

Reference (1) Krieth, F. "Principles of Heat Transfer," 3rd Ed., 1973.

- the RPV sensible heat released upon recovery of a low pressure ECCS system is transported to the pool 57

- The mass of hydrogen equivalent to 100% MWR is transported to the containment air space

The peak containment pressure was determined using the following assumptions:

1) The energy equivalent to 100% metal-water reaction of the active fuel cladding, 2) the RPV sensible heat release upon recovery of a low pressure ECCS system and 3) decay heat is assumed to be transported to the suppression pool. The mass of hydrogen equivalent to 100% MWR of the active fuel clad is transported to the containment airspace. 59

- 5421 lb. moles CO_2 is discharged to the containment airspace.

- The analysis was performed using the ANS + 20% decay heat for the first 1000 seconds and ANS + 10% to the end of the time period evaluated.

- No credit for suppression pool cooling or containment sprays was assumed from 0 to 85 minutes.

<u>Results</u> 57

- Maximum Suppression pool temperature = 183^oF 59

- Maximum Containment airspace temperature = 133^oF

- Containment pressure = 42.5 psig

The calculated containment temperature and pressure responses using these assumptions are reported in Section 6.2.1.3.4. These analyses provide assurance that the use of a 45 psig post-accident pressure for the containment vessel and anchorage design (see Sections 3.8.2.3.2 and 3.8.5.3.2) will be adequate to maintain containment integrity. 57

The post inerting system will be designed such that its use in combination with the more probable break events (including steam bypass) will not result 59
in containment pressure in excess of service level C. Degraded core analyses verifying this condition will be provided two years after issuance of the construction permit.

A preliminary system level review of the functions necessary to maintain containment integrity during the postulated event described above and the systems which perform these functions has been made. The approach taken, in the absence of a definitive accident specifications, was to identify, in general, those functions and the related systems which support and maintain containment integrity. Assuring that those functions are maintained provides a conservative approach to maintaining containment integrity. The results of this review are shown in Table II.B.8.4-1. 57

Preliminary review of those portions of the identified systems which must function and are environmentally exposed to this accident indicates that it is feasible to demonstrate necessary functions will be performed. It is HL&P's intent to finalize this review and then perform a detailed analysis of individual subsystems and components and demonstrate that required functions will be assured considering environmental conditions and required duration of operation. It should be recognized in this process that modifications to Table II.B.8.4-1 may occur.

Qualification of equipment will involve three basic steps: 1) definition of environmental condition; 2) determination of equipment which is needed to function; and 3) demonstration that equipment will function as required under these conditions.

The expected containment environmental conditions will be determined. Containment pressure and temperature curves will be developed for both inadvertent actuation and for the transients postulated to result in the degraded core. The inadvertent actuation analyses is expected to result in the minimum negative pressure and the minimum temperature. The hypothetical non-break accident with postulated multiple system failures which are required to reach a degraded core condition is expected to produce the maximum 59
containment temperature and pressure. Accident and inadvertent inerting thermal and pressure envelopes will be provided two years from issuance of the Construction Permit. Further definitions of this event is expected to be developed as a part of degraded core rulemaking proceedings. The accident scenarios which are identified in this process will be evaluated in preparing operator guidelines. These guidelines will assure that PAIS actuation in conjunction with pressure and/or temperature peaks resulting in a combination of accident and PAIS effects exceeding plant design will not occur.

In addition to the bulk containment effects, localized effects will be identified so that equipment located in close proximity to nozzles can be evaluated. The composition of the environment will be based on the postulated transient (i.e. hydrogen, radiation, steam, air, CO_2).

The second phase is the identification of equipment which must function. An evaluation is currently underway to finalize the list of equipment required to function so that the plant can be safely shut down following inadvertent actuation during normal operation. The specifications for this equipment will reflect these requirements. An evaluation is also underway to finalize the list of equipment necessary to ensure containment integrity and safe shutdown following the postulated accident.

The final step will demonstrate that the equipment meets these requirements. Qualifications will be accomplished by a combination of testing and analysis. It is expected that qualification to the plant design bases accidents will envelop most environmental effects. The notable exceptions are increased pressure and the short negative temperature transients. These will affect certain classes of equipment more than others. The corrosion effects of CO_2 are not expected to be a problem and literature search, analysis, and, if necessary, a test program will be conducted to demonstrate qualifications. 59

These steps will be performed as a normal engineering process on ACNGS.

Following issuance of the Construction Permit for ACNGS HL&P will incorporate the results of industry that are applicable to the investigation of Post-Accident Inerting. Within two years after receipt of the construction permit, design details, describing the hydrogen control systems, will be provided to the NRC for review; these design details, including test data and analyses, will illustrate that the hydrogen control systems will perform in the manner required by the above NRC position. The level of detail of the hydrogen control system's function and layout will be the same as that required for other systems at the CP stage of review.

The final design for hydrogen control will be described in the Final Safety Analysis Report.

NUREG 0718 REQUIREMENT

(4) (b) "The containment and associated systems will provide reasonable assurance that uniformly-distributed hydrogen concentrations do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion."

57

RESPONSE

(4) (b) The containment and associated systems are chosen to ensure that the post-accident atmosphere will not support hydrogen combustion. The concentration of CO_2 necessary to ensure non-flammability of a hydrogen-air-CO_2 mixture is based upon Bureau of Mines test data. The chosen value of 61% CO_2 by volume is considered conservative for the following reasons:

1) Ignition Source

The flammability limit is dependent on the type of ignition used. For example, ignition tests of lean mixtures have shown that spark ignition, the most likely ignition source in the containment is much less efficient than a flame. The Bureau of Mines tests utilized a flame ignition source. Coupled with the fact that the amount of inerting agent required is the greatest for lean mixtures, this indicates that a lower concentration of CO_2 would be required to preclude ignition by a spark source.

2) Effects of Water Vapor and Initial Containment Temperature

The required quantity of CO_2 is based upon dry air at 70F in the containment. Higher initial containment temperature and any water vapor content reduces the quantity of air present in the containment, and thus increases the resulting volume percent of CO_2. In addition, water vapor has the same inerting effect as CO_2, requiring approximately 58% to inert any mixture of air and hydrogen. It can be expected that the effects will be additive.

3) Allowances for CO_2 Solubility in Water

Approximately 9% additional CO_2 mass, beyond that required to reach 61% by volume in the air space, is injected to account for eventual absorption of CO_2 into the reactor and containment water masses. Initially, this quantity would be available as additional margin. As the CO_2 dissolves into the water, the water vapor content of the air would be expected to increase as the suppression pool temperature increases during the course of the accident. 57

4) Maximum CO_2 Required at Lower Hydrogen Concentration

The quantity of CO_2 is determined by the amount required to inert a hydrogen-air-CO_2 mixture containing 10% hydrogen. With higher or lower volume percents of hydrogen, less CO_2 is required. Approximately 52% CO_2 is required to inert the mixture containing hydrogen from 100% MWR. It is likely that a 10% hydrogen concentration would occur early in the event while the CO_2 allowance for water absorption was still present in the air space.

NUREG 0718 REQUIREMENT

(4) (c) "The facility design will provide reasonable assurance that, based on a 100% fuel clad metal-water reaction, combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features."

RESPONSE

(4) (c) The design objective adopted is to mix sufficient CO_2 with all of the atmosphere in the containment and drywell so that burning cannot occur regardless of the location and concentration of the hydrogen. To meet this objective it is required that the CO_2 be delivered in a controlled manner to selected locations throughout the containment and then well mixed with the surroundings.

Delivery will be achieved by a distribution system consisting of piping and manifolds extending from the storage tanks, throughout the containment and terminating at nozzles at the selected locations. The carbon dioxide stored in tanks is a liquified gas at saturation pressure and temperature. As the liquid is discharged, the tank pressure begins to drop and the remaining liquid boils as some of the liquid is evaporated to maintain saturation equilibrium. The vapor thus generated maintains a high pressure in the storage container as the bulk of the liquid is discharged into the pipeline.

The liquid in the pipeline continues to boil in response to any drop in pressure. This results in a mixture of liquid and vapor which, in a properly sized piping system, flows at a velocity that assures high turbulence for continuous mixing of the liquid and vapor portions, i.e., two-phase flow.

Because the volume ratio of vapor to liquid increases as the pressure drops, the average density decreases and the velocity throughout the pipe increases. The rate of pressure drop per foot of pipe thus becomes greater as the mixture proceeds throughout the piping at reducing pressures. This is due, in part, to greater frictional loss at higher velocities and, in part, to the energy required to increase the velocity head of the mixture.

The equations which make it possible to accurately calculate the pressure drop for two-phase flow also take into account the effect of changing densities, based on thermodynamic data for liquified carbon dioxide in question. They also account for the changing velocity head in the piping. This method of calculation has been adopted in NFPA 12, "Carbon Dioxide Extinguishing Systems." 57

Finally, the discharge rate from individual nozzles is based on test data relating discharge rate to terminal pressure conditions.

As might be expected, the calculations are too complex for practical manual solutions. The calculations are, therefore, performed by execution of computer programs previously developed for this purpose.

Using these techniques, the distribution system is optimized with respect to piping sizes and nozzle orifices to achieve the desired delivery at each of the selected locations.

Mixing with the surrounding air to assure even distribution is achieved by using nozzles which release the CO_2 at high velocity directly into the air resulting in an energetic expanding jet in which rapid momentum exchange entrains and mixes into the jet large amounts of the surrounding air. In addition to the mixing within the jet, the momentum of the jet is used to force a general circulation pattern within the containment. The two nozzle groups which discharge downward into two of the four open quadrants from the upper level to the bottom of the containment force the general circulation down these paths. This flow then moves circumferentially around the containment and returns upward through the opposite quadrants, thus closing the flow paths into recirculation loops.

Upward flow is aided by the upward momentum of the discharge from the nozzle groups in the upper portion of each of the upward flow paths. In addition, these nozzle groups assure that air in the top of the containment receives CO_2 and is forced to participate in the general circulation.

An examination of both the containment drawings and a detailed model of the containment was performed to identify locations which might not participate effectively in the general circulation described above. A nozzle will be placed in each suspect location.

A similar process of examination and nozzle location selection was performed for the drywell. However, for the drywell it was not found necessary to force a general circulation. The turbulence caused by the jets will assure mixing in this relatively small volume.

The adequacy of mixing and nozzle placement is based on experience
in the design and testing of many industrial CO_2 fire
extinguishing systems. However, it may be useful to consider the
following highly simplified illustrations of the energy and momentum
available to force the mixing process. For example, the discharge 57
velocity at the nozzle will be in excess of 500 feet per second. At
this velocity, the energy that must be dissipated in the form of
turbulence and air movement is nearly 4000 foot-pounds per pound of
CO_2. At the design discharge rate of over 300 pounds per second,
this is equivalent to more than 2000 horsepower. Also, based on the
principle of conservation of momentum, each pound of carbon dioxide
discharged is capable of accelerating 15 pounds of air to 30 feet
per second. Velocities of this order would recirculate the
containment air several times per minute.

The calculational methods and design skills used in this design have been applied in the design of fire suppression systems. Full demonstration tests have been performed on some of these systems. Such tests consist of actuation of the completed system and measuring, as a function of time, the concentration of CO_2 at points within the volume being flooded. These test results consistently show rapid and thorough mixing.

Further, test experience has shown that lengthening the discharge period markedly promotes mixing. Although tests have shown effective mixing can be achieved with a discharge period as brief as 10 seconds, discharge periods of from one to four minutes are more typical. The discharge period used in this design approaches 15 minutes and allows more than adequate time for mixing.

Finally, the mixing of gases is permanent. That is, once the carbon dioxide is mixed with the air, it will not stratify, settle, or otherwise separate from the air.

NUREG 0718 REQUIREMENT

(4) (d) "If the option chosen for hydrogen control is post-accident inerting: (a) Containment structure loadings produced by an inadvertent full inerting (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Division 1, Subsubarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Division 2, Subsubarticle CC-3720, Service Load Category), (b) A pressure test, which is required, of the containments at 1.10 and 1.15 times (for steel and concrete containments, respectively), the pressure calculated to result from carbon dioxide inerting can be safely conducted, (c) Inadvertent full inerting of the containment can be safely accommodated during plant operation." 57

RESPONSE

(4) (d) A preliminary evaluation has been done using the current (including modifications described before ACRS on 2/6/81) configuration of the containment vessel and its anchorage into the foundation mat. The evaluation results show that containment structure loadings produced by an inadvertent full inerting (assuming carbon dioxide), but not including seismic or design basis accident loadings, will not produce stresses in the containment in excess of the limits set forth in the ASME Boiler and Pressure Vessel code, Division 1, Subsubarticle NE-3220, Service Level A Limits for the containment vessel Division 2, Subsubarticle CC-3430, Service Load Category for the concrete anchorage and Division 2, Subsubarticle CC-3720, service load category for bottom liner plate. 59

A pressure test of the containment at 1.10 times the pressure resulting from carbon dioxide inerting can be safely conducted.

In order to assure containment integrity for the required conditions, the applicant has revised the commitments found in Section 3.8 as follows:

a. Additional loads have been defined in Sections 3.8.2.3.1 and 3.8.5.3.1. These loads were established as the maximum temperature and pressure as calculated in the inadvertent actuation analysis which is described later in this response. 57

b. Additional load combinations have been specified in Sections 3.8.2.3.2 and 3.8.5.3.2. These load combinations include the consideration of dead weight, pressure and temperature for the inadvertent actuation event.

A complete description of the containment vessel and its concrete anchorage design and analysis procedures is provided in Sections 3.8.2.4 and 3.8.5.4 and is summarized in 4 (a) above.

For the additional inadvertent actuation and testing loading conditions shown in Section 3.8.2.3.2, the containment vessel acceptance criteria will be based on the allowable stresses defined in Table 3.8-1 for normal load combinations and for test load combination, respectively. These allowable stresses are based on the requirements of the ASME Code, Section III, Division 1, Subsection NE and are as follows:

Inadvertent Inerting (service level A):

- General membrane stresses: S_m
- Local membrane stresses: $1.5\ S_m$ 57
- Bending + local membrane stresses: $1.5\ S_m$
- Primary + secondary stresses: $3\ S_m$

Test Condition (Test allowables):

- General membrane stresses: $0.9\ S_y$
- Local membrane stresses: $1.25\ S_y$
- Bending + local membrane stresses: $1.25\ S_y$
- Primary + secondary stresses: $3\ S_m$

The anchorage concrete and bottom liner will be designed for the effects of the inadvertent actuation and testing loading conditions shown in Section 3.8.5.3.2 utilizing the allowable stresses and strains for service loads and test loads, based on the requirements of the ACI-ASME Code as follows: 59

Inadvertent Inerting Condition (using Service Load Category)

- Concrete compression, shear, torsion, bearing: use allowable stresses for service loads as specified in the ACI-ASME Code, Paragraph CC-3431. 57
- Reinforcing steel tension and compression: use allowable stresses and strains for service loads as specified in the ACI-ASME Code, Paragraph CC-3432.
- Bottom Liner: Use allowable strains for service loads as specified in the ACI ASME Code, Subsubarticle CC-3720. 59

Test Condition (using Test Allowables):

- Concrete compression and bearing: use allowable stresses for service loads as specified in the ACI-ASME Code, Paragraph CC-3431. 57
- Concrete shear and torsion: use allowable stresses for service loads, increased by 33-1/3%, as specified in the ACI-ASME Code, Paragraph CC-3431.

- Reinforcing steel tension and compression: use allowable stresses and strains for service loads increased by 33-1/3% as specified in the ACI-ASME Code, Paragraph CC-3432. 57

- Bottom Liner: Use allowable strains for service loads as specified in the ACI-ASME Code, Subsubarticle CC-3720. 59

INADVERTENT ACTUATION ANALYSIS

The event sequence given below was developed to provide a reasonable basis upon which to calculate the containment pressure and temperature response so that containment loads can be evaluated for inadvertent operation of the CO_2 system.

Initiating Event - CO_2 system is inadvertently actuated and begins injection of CO_2 into the containment. 57

Assumptions

- A reactor scram and containment isolation (but not MSIV closure) occurs when drywell pressure exceeds 2 psig.

- A full charge of inerting agent (CO_2) is injected into the containment.

- Reactor heat removal and depressurization to approximately 150 psig is accomplished by steam flow to the main condenser. Safety relief valves do not open. HPCS and the feedwater system supply water to the reactor until both are tripped by high water level or level increase is terminated by operator action. HPCS is then secured and feedwater is restarted if necessary and used to maintain reactor water level.

- The operator promptly initiates a 100F/hour shutdown of the reactor using the turbine bypass valves.

- Containment Sprays activate and operate until containment pressure is greater than 9 psig (10 minutes after the high drywell pressure signal).

- Upper Pool dump occurs 30 minutes after the high drywell pressure signal.

A detailed analysis was performed to determine the minimum pressure and temperature transients for the inadvertent case assuming heat transfer to the walls and structures.

Minumum Pressure/Temperature Assumptions:

1) A conservative overall heat transfer coefficient of 3 BTU/hr-ft^2-F was used between the airspace and the containment walls and structures. 59

2) Available heat source and sink areas and volumes obtained from GESSAR II 238 Nuclear Island 22A7000 Rev. 0 033180 pg. 6.2-171, table 6.2-9.

3) All heat sources and sinks were modeled as flat plates with a thickness of (volume/area) so that Heisler and Grober charts (ref. 1, Figs. 4-8, 4-10) could be used to evaluate the transient heat transfer.

4) Heat and mass transfer to or from the suppression pool was neglected. This is a conservative assumption.

5) Friction heating of CO_2 in the piping is negligible.

6) The liquid CO_2 is injected at a constant 0 F.

7) The effects of the initial vapor plug on pressure were neglected, a conservative assumption. 59

8) CO_2 flow rate (135 tons in 15 minutes - linear rate).

9) Calculations were done using a time step of 1 minute.

10) The containment ventilation system was not considered. This assumption is a major conservatism since in reality the containment would be open to the atmosphere at the time of lowest pressure and thus would minimize the negative pressure.

The following information has been used in evaluating the maximum containment pressure due to inadvertent full inerting.

Initial Conditions:

$V = 1{,}217{,}707\ ft^3$ (Containment only - excludes drywell)
Rel. Hum. = 50%
$T = 95^oF$
Mair = 2922 lb mole
CO_2 addition to containment = 5421 lb mole

Assumptions:

1) No mass or heat transfer between drywell and containment.
2) Vessel cooldown is through the main condensers (i.e., no energy addition to the suppression pool).
3) The final temperature is 95^oF.
4) The final relative humidity is 50%.
5) No air or CO_2 escapes through the containment ventilation system prior to isolation.
6) All the CO_2 added for solubility is injected into the containment.
7) No CO_2 dissolves in the suppression pool.

AVAILABLE CONTAINMENT HEAT SINKS

Item	Volume (ft^3)	Surface Area (ft^2)	Material
A. Drywell Structures	11,230	2,390	Concrete
B. Containment Shell	8,227	65,815	Steel
C. Miscellaneous Steel Structures and Equipment	20,718	144,456	Steel
D. Miscellaneous Concrete Structures	68,640	17,719	Concrete

59

Results: 57

1) Pool Temperature = 95°F
2a) Maximum Containment Pressure = 26.5 psig 59
2b) Minimum Containment Pressure = -.1 (see Figure II.B.8.4-1)
3) Minimum Temperature = -80°F (see Figure II.B.8.4-1)

Certain systems will be required to achieve cold shutdown following an inadvertent actuation event. A preliminary review indicates that the following systems are required and may be affected by the resulting containment conditions: Reactor Protection System, Control Rod Drive, Rod Control and Information System (Rod Position), RHR and containment purge. The review will be completed in detail; however, the preliminary results show that it is feasible to demonstrate a safe shutdown.

NUREG 0718 REQUIREMENT 57

(4) (e) "If the option chosen for hydrogen control is a distributed ignition system, equipment necessary for achieving and maintaining safe shutdown of the plant shall be designed to perform its function during and after being exposed to the environmental conditions created by activation of the distributed ignition system."

RESPONSE

(4) (e) Since the distributed ignition system was not chosen for hydrogen control, this requirement is not applicable. If this system is subsequently decided on as the final design for hydrogen control, the required analyses will be performed.

TABLE II.B.8.4-1

SYSTEMS NECESSARY TO ENSURE CONTAINMENT INTEGRITY

DURING A POSTULATED ACCIDENT EVENT

Function	System
Reactivity Control	Reactor Protection (Trip System) Control Rod Drive (Scram function)
Reactor Depressurization	Automatic Depressurization System ADS Instrument Air (Accumulators)
Residual Heat Removal	RHR System (Containment Spray, Suppression Pool Cooling Modes)
Reactor Water Makeups	
- High Press	High Pressure Core Spray System
- Low Press	RHR System (LPCI Mode) Low Pressure Core Spray System
Containment Isolation	Containment and Reactor Isolation Systems Main Steam Leakage Control System
Special Containment Functions	Containment Vacuum Relief Drywell Vacuum Relief Suppression Pool Makeup System
Combustible Gas Control	Post-Accident Inerting System
Post-Accident Monitoring	Containment Atmosphere Monitor Post-Accident Sampling System
Electrical Power Supply and Distribution	Standby Power 4.16 Kv - Safety 480V ac Safety 120/208V ac Safety 277/480V ac Safety 120V ac RPS 125V dc Safety
Equipment and Space Cooling	Essential Services Cooling Water System ECCS Area fan cooler

57

ITEM II.E.4.1 DEDICATED PENETRATION

NUREG 0718 REQUIREMENT

"Applicants for plant designs with external hydrogen recombiners shall modify their applications as necessary to include redundant dedicated containment penetrations so that the recombiner systems can be connected to the containment atmosphere without violating single-failure criteria, such as having to open large containment purging ducts or otherwise jeopardizing the containment function. Applicants shall submit, prior to the issuance of construction permits or the manufacturing license, a detailed explanation of how the requirements will be met in order to provide reasonable assurance that the requirements will be implemented properly." 57

RESPONSE

The ACNGS design utilizes thermal recombiners which are internal to containment and have no associated mechanical penetrations. As such, this item is not applicable.

ITEM II.J.3.1 ORGANIZATION AND STAFFING TO OVERSEE DESIGN AND CONSTRUCTION

NUREG 0718 REQUIREMENT

Applicants shall describe their program for the management oversight of design and construction activities. Specific items to be addressed include: (1) the organizational and management structure which is singularly responsible for the direction of the design and construction of the proposed plant, (2) technical resources which are directed by the utility organization, (3) details of the interaction of design and construction within the utility organization and the manner by which the utility will assure close integration of the architect engineer and nuclear steam supply vendor, (4) proposed procedures for handling the transition to operation, and (5) the degree of top level management oversight and technical control to be exercised by the utility during design and construction, including the preparation and implementation of procedures necessary to guide the effort.

Draft NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources" is the keystone for similar development of guidelines for this task. Therefore, the principal applicable elements of NUREG-0731 shall be used by CP and ML applicants in addressing this task.

Applicants shall submit detailed information in order to provide reasonable assurance that the requirements will be implemented properly prior to issuance of the construction permits or manufacturing license.

RESPONSE 57

(1) Organization and Management Structure

(a) General

HL&P has always been aware that utility involvement in the design and construction phase of a Project enhances safety, reliability, and operability. Shortly after HL&P became involved in Nuclear and Coal programs, both of which were new to the Company, HL&P began utilizing the Project Management Organization (PMO) concept in our staffing structure. The PMO ensures that the decision making process is integrated during design, construction and start-up. The Manager, Allens Creek, has full authority to implement Project goals, and all contractors and HL&P team members, except Quality Assurance, are under his direction.

The major organizations involved in the ACNGS are:

a) Houston Lighting & Power Company (HL&P)

As the Applicant, HL&P has and retains the overall responsibility for the engineering, design, procurement, fabrication, construction, preoperational testing, operation and QA activities for the ACNGS.

HL&P will audit the activities of Ebasco, GE, consultants and other contractors to assure that their QA programs are implemented and have sufficient authority and organizational freedom to be effectively implemented.

HL&P will perform surveillance of the activities of Ebasco, GE, consultants and other contractors during the manufacturing, fabrication and construction of the ACNGS.

b) Ebasco Services Incorporated (Ebasco)

As the Architect - Engineer, Ebasco is delegated the responsibility to provide HL&P with engineering, design, procurements and QA services. As the constructor, Ebasco is delegated the responsibility to provide HL&P with construction and QA services at the site.

Ebasco has the responsibility to provide an acceptable QA program to HL&P for the activities that have been delegated to Ebasco. These delegated activities include the following: 57

1) design and engineering
2) procurement activities
3) home office QA activities
4) vendor surveillance activities
5) construction activities
6) site QA/QC activities

Figure 17.1.1B-1(a) is an organization chart showing the Ebasco QA organization for the Allens Creek Project and the functions or activities that they will perform.

c) General Electric Company (GE)

As the Nuclear Steam Supply System (NSSS) and Nuclear Fuel Supplier, GE is delegated the responsibility to provide HL&P with the engineering, design, procurement, fabrication and QA services for the NSSS and Nuclear Fuel. GE has the responsibility to provide an acceptable QA program to HL&P for the activities that have been delegated to GE.

These delegated activities include the following:

1) design and engineering activities
2) procurement activities
3) fabrication activities
4) vendor surveillance activities
5) QA activities

The functional interfaces between the organizations are shown on Figure II.J.3.1-1.

The HL&P organization and management structure and details of the scope of work and division of responsibilities can be found in the PSAR - Chapter 13. Quality Assurance responsibilities and scope of work are fully described in the PSAR - Chapter 17.

Following the TMI incident, HL&P began to reassess our Management structure and personnel qualifications as did most of the nuclear industry. As a result, the following changes have or will be implemented:

- corporate reorganization such that the Executive Vice-President reporting to the President has only nuclear responsibilities.

- brought into executive management a highly-experienced individual in the nuclear field to manage the design and construction of our nuclear plants. 57

- establish an active HL&P engineering organization onsite during construction to assure problems are resolved thoroughly and correctly and to provide continuity of design through start-up and operation.

Each of these points are discussed below.

(a) Executive Vice President

To insure availability of HL&P Corporate Management to be aware and involved in our nuclear projects, a major change was made in mid-1980. Nuclear and Fossil fueled operations were separated at the highest level in the company and an Executive Vice President was named having total, but only nuclear responsibilities. These nuclear responsibilities include Engineering, Construction, Operations, Fuel Management, and Quality Assurance. The change will ensure that the nuclear programs will receive thorough and timely attention and that adequate priority can be placed on Company resources to resolve any problems.

(b) Vice President - Nuclear Engineering and Construction

Another organizational change designed to bring nuclear design and construction experience into Corporate Management was the establishment of a new position - Vice President, Nuclear Engineering and Construction. This position, reporting to the Executive Vice President, was filled in October 1980.

He has 25 years of nuclear experience and has been in responsible charge of many aspects of nuclear design and construction including Project Manager, Construction Manager, and Chief Engineer. Prior to joining HL&P, he was Vice President, Construction, for Stone & Webster Engineering Corporation.

(c) Site Support

HL&P will establish an organization on site that will be involved in resolving problems and will ensure continuity and expertise after the Contractor has completed his obligations. This will be done by relocating the present HL&P Project organization physically to the site. The organization will maintain current design-related and procurement responsibilities but will also grow and be directly involved in construction changes, contractor deficiency corrections and start-up. Upon project completion, these individuals will be familiar with the plant design and be available for direct site or support activities. The relocation of the engineering organization to the site will occur approximately 18 months after start of construction. 57

(2) Technical Resources Directed by the Utility

(a) Staffing Levels

Prior to the start of Allens Creek construction, HL&P has maintained an in-house staff of approximately 31 full-time engineers and managers to oversee the design and verify conformance with the applicable regulations, codes and design criteria. This manpower has proved sufficient to meet the responsibilities of the project except in specific cases. In these cases, temporary engineering support is assigned from line departments or consultants contracted to work under the sole direction of HL&P personnel. Table 13.1-3 identifies corporate technical resources and experience. Table 59
13.1-2 identifies individual experience of personnel assigned full time to the Allens Creek Project. To support the construction and operations of Allens Creek, HL&P has scheduled staffing levels as 57
shown in Table II.J.3.1-1. 59

(b) Level of Education and Experience

HL&P has and will continue to retain a highly trained and capable staff to meet the responsibilities of overseeing the design of Allens Creek.

Also, there is a wide range of technical expertise that exists within the corporate organization covering all the major engineering disciplines plus some of the more highly-specialized fields. Should 57 a technical issue arise that is outside the scope of HL&P's engineering capabilities, HL&P has the option to obtain the outside services of experts to assist in resolving the issue.

We are aware of the technical support skills required in NUREG-0731 for operations. We have already on staff some individuals that meet or will meet the qualifications outlined in that document. We plan to acquire or train individuals so that all requirements are satisfied. For instance, this year we are actively recruiting for specialized skills in the areas of welding, engineering, metallurgy, ASME pipe stress analysis, and transient analysis. We recognize the existence of a personnel shortage in many of these skilled areas, and have been using outside recruiting agencies, open houses, and nationwide advertising to attract the appropriate personnel. The company has a formal staffing program where annually, needed job functions and skills are identified for each department, the future staffing plans to fill these positions are approved by executive management, and then a coordinated recruiting effort commences.

Once Allens Creek becomes operational, HL&P will assume design responsibility. Before that occurs, we will have in place a program that meets ANSI N45.2.11, "Quality Assurance Requirements for the Design of Nuclear Power Plants". The technical support staff will produce a manual of formal procedures governing preparation of design documents, document control (including review and approval), design verification requirements, control of design changes including design change requests, and training, among others. 57

(c) Training Programs

In addition to hiring experienced individuals into the Company, HL&P has an active technical training program. All professionals have the opportunity and are expected to attend an outside developmental course or seminar each year. Also, line departments, who are responsible for training, hold technical work shops directed by in-house experts or AE and NSSS training personnel. Typical work-shops included basic studies of codes, pumps, valves, BWR design, etc.

Further, the Health Physics Division, under the Nuclear Services Department, has established a nuclear-wide radiation training group.

The training group is developing over 13 courses to teach HL&P personnel and contractors in radiation protection, including the full range of technician training, general employee training, and operator training.

As a matter of interest, the training group will conduct a pilot training program for Radiation Protection Technicians at the request of INPO this spring. INPO has audited this group recently relative to the presentation of this pilot program and found everything highly satisfactory.

(d) Experience Feedback

An important input to the technical staff is operating experience. This experience is distributed through information contained in documents such as I&E Bulletins and LER's. At present, NRC generated input, including I&E Bulletins, Notices, New Regulations, and Regulatory Guides are screened by our Nuclear Licensing Department for applicability and importance and then sent to the appropriate technical personnel for action. See the response to Item I.C.5. 57

The publication, Nuclear Power Experience Reports, is used as another source of input to the technical support and operations staffs. The reports are reviewed by the cognizant discipline and factored into the plant design, construction and/or planned operation as appropriate along with other inputs.

In addition, both the South Texas Project Plant Superintendent and the Manager - Nuclear Services, are members of the EEI Nuclear Operations Subcommittee. This group, which meets triannually, is composed of the chief technical support and operations personnel for each utility in the U.S. They meet and exchange information concerning operational experiences. This group has been functioning for many years.

Through the efforts of NSAC and INPO, the many hundreds of LER's are now being screened and distributed to interested parties, through a service known as NOTEPAD. We are a user of that service.

Also, as a result of TMI, we are actively participating in industry efforts including the W and GE Owner's Groups. 59

(e) Analytical Capability

HL&P has recognized the need for in-house analytical capability in certain areas which will be needed. The Nuclear Services Department is developing capability to perform transient analysis and has purchased and is benchmarking such computer codes as RETRAN, CONTEMPT, and COBRA. 57

The Nuclear Fuels Department has purchased and is benchmarking such computer codes as PDQ-VII and ARMP for limited in-core fuel management capability. This group is studying the extent of additional in-house capability needed to adequately meet the Company needs.

(3) Details of the interaction of design and construction activities

(a) General

The following supplements the material in PSAR Section 13.1.2 describing in more detail, the interaction of design and construction activities by HL&P and its principal contractors, Ebasco Services, Inc. and General Electric Corporation (GE) for the nuclear steam supply system. Establishment of the divisions of responsibility and the means of assuring close integration of the work is manifested in contractual documents, project (inter-company) procedures, the balance of plant System Design Descriptions, and the GE Customer Interface Data Document.

Ebasco is responsible to HL&P project management, planning, cost control, engineering, procurement, construction, sub-contract administration, quality control, and quality assurance. Ebasco is also responsible for design interface control among Ebasco, GE, and other vendors and contractors. Ebasco is accountable to perform its services in accordance with all applicable Federal, State and Local codes and regulations including the Quality Assurance requirements of 10CFR50 Appendix B. HL&P monitors and evaluates Ebasco performance of these responsibilities by requiring Ebasco to obtain HL&P approval of the basic design criteria and selected design documents. Further, HL&P places purchase orders for all engineered equipment based on Ebasco generated and HL&P approved specifications. 57

GE is responsible to HL&P design and fabrication of the Nuclear Steam Supply System including preparation of design documents and procurement of related hardware. GE prepares system descriptions and other selected design documents for both HL&P and Ebasco. HL&P monitors and evaluates GE performance by review of these documents. Ebasco reviews these documents to ensure interface coordination between the NSSS and balance of plant. Otherwise GE has authority to determine the NSSS design, subject to HL&P QA surveillance. GE prepares: interface criteria; safety analyses; other design information; test procedures, maintenance and operating procedures; and technical support for NSSS installation. GE is accountable to HL&P to perform its services and provides NSSS designs and equipment in accordance with all applicable Federal, State and Local codes and regulations, including the Quality Assurance requirements of 10CFR50 Appendix B.

HL&P is ultimately responsible for the overall design, construction, and operation of Allens Creek in accordance with NRC regulatory requirements, including the Quality Assurance requirements of 10CFR50, Appendix B. HL&P's Project Management Organization, described in PSAR Section 13.1.1.5 is responsible for providing management oversight of principal contractor activities, obtaining Federal licenses and permits, approving basic design criteria, releasing selected design documents, and authorizing expenditures of funds. HL&P also retains stop work authority of contractor design and construction activities. A chart showing Project Responsibility Interface is attached.

The Allens Creek Project Organization is made up of individuals assigned to the Project from the several line or functional departments. The Organization consists of a Manager and a support team of engineers and professionals whose only function is to manage the design, licensing, procurement, construction, and start-up of Allens Creek. The Manager reports to the Vice-President - Nuclear Engineering & Construction, and is accountable to him for the cost, schedule and quality of the Project. The Organization in turn manages the contracts for outside support, principally Ebasco and GE. All technical and administrative direction to the contractors is provided through the Allens Creek Project Organization. 57

The individuals assigned to the project bring with them the necessary authorities and responsibilities to act on behalf of the department represented within the framework of the Allens Creek Project Organization. This allows the Project to function as a coherent unit with decisions being made at the appropriate level. Further, the Organization except Accounting and Environmental Protection is grouped physically together ensuring good communication and interface between departments and disciplines. The Project Organization is staffed so that all normal activities can be accomplished within the team.

Line Department Overview

Line departments establish technical policy and technical procedure through published policies and guidelines. These items serve to document required policy and procedures on a corporate basis and ensure that company nuclear activities are coordinated and controlled. These publications are being prepared so as to be in place upon the beginning of construction of ACNGS. Detailed project procedures reference and implement any appropriate department requirements. Project procedures may expand upon and provide more detailed quidance relative to department publications and documents (written approval) and deviations therefrom.

The line departments serve as the primary technical resource for complex or specialized tasks and for peak work periods. The line departments also plan for analytical systems, methods development, and additions to staff. The line departments and divisions assigning full-time project staff to Allens Creek are:

Nuclear Services

Nuclear Engineering
Health Physics

Nuclear Licensing

Nuclear Operations

Power Plant Engineering

Mechanical Engineering
Electrical Engineering
Civil Engineering
I&C Engineering 57

Power Plant Purchasing

Allens Creek Purchasing
Material Control

Environmental Protection

Power Plant Accounting

Quality Assurance

QA Engineering
Vendor Surveillance

Allens Creek Administration

Project Administration
Project Controls
Project Construction

Nuclear Fuels

Records Management

(b) Overview of Design

Overview of design is implemented and controlled by the Allens Creek Project Engineering Manager and those individuals of the Project Organization assigned to him. The inter-company drawing and specification development, review, and approval procedures outline step by step the activities done by Ebasco, HL&P, and equipment suppliers. Intra-company Allens Creek procedures describe Project internal activities. These procedures specify how the review is initiated, assigned, and documented.

HL&P provides management overview of design changes through the Project (inter-company) Procedures which specifically define all mechanisms that the Principal Contractors can use for design changes. In addition to the specific HL&P control aspects over design and procurement activities, HL&P Monitors the quality, cost, and timeliness of other activities performed by the Principal Contractors. Management oversight of contractor design activities is facilitated by the issuance of several status and performance reports which are directed to various levels of management. Also, copies of correspondence among contractors are sent to HL&P for information. 57

(c) Overview of Construction

The HL&P internal organization described in Section 13.1.1.1.1.7 shows the Allens Creek Project Construction staff including the Site Construction QA Group and its relation to the Allens Creek Project Organization.

The Project Construction Manager (PCM) and his staff are responsible for construction overview of contractor performance. The contractors and sub-contractors under Ebasco construction management are responsible for construction in a manner that conforms to design quality requirements. The PCM and his staff: monitor construction activities; approve schedules, field procurements, selected invoices, and other financial controls; monitor compliance with permit and license requirements; monitor procedure compliance; monitor coordination of Ebasco field engineering with Ebasco home office engineering staff; and coordinate Contractor turnover of plant systems to Nuclear Operations.

In addition, QA provides construction overview through the Site Construction QA Group which is responsible for monitoring the QA aspects of site construction, including: review of contractor site procedures; audits and surveillance of construction; identification of quality problems and monitoring of their resolution; and acceptance reviews of components, constructed structures, and completed systems. The Site Construction QA Group interacts directly with the Ebasco site organization and with the HL&P home office QA organization. HL&P will have approved procedures for construction activity. These procedures will reflect the organization and will conform to applicable regulatory requirements, contractural arrangements, and the Allens Creek Quality Assurance Plan. Procedures will exist for each organizational element involved in construction overview activities.

(4) Transition to Operation

(a) Technical Continuity

HL&P has a single Executive Vice President responsible for nuclear plant engineering, construction, fuel, QA, and operation. This will greatly facilitate the transition from construction of Allens Creek to operation. The Nuclear Services organization responsible for review and approval of plant design will continue as the technically cognizant expert resource when Allens Creek operates, performing the same functions of engineering support as they do now for the South Texas Project.

Once Allens Creek becomes operational, HL&P will provide the 57
required technical support necessary to assure safe and reliable plant operation. This support will be consistent with the guidelines suggested in NUREG-0731. Technical specialty support from outside sources will be employed when necessary. We have been studying various organization alternatives to provide this support and the transition steps from a design and construction team to an operations support team. While our studies are not yet complete, we have made note of several key factors. Since the HL&P Project Engineering Organization will be physically located at the site during construction and start-up, the individuals will have excellent familiarity with the equipment. These individuals will be a basic resource for actual transfer to the operations or engineering support groups. We think keeping this group on site will improve its performance by giving the technical support staff maximum access to systems that they will be working on and by developing a close relationship with the operating staff which should serve to improve communications. Although there will be formal procedures by which the plant staff can request design changes, this close relationship should improve the mutual understanding and performance of both groups.

Our goal for technical skill level is to have on staff individuals who are technically capable of performing design verification for all technical areas, especially those that are uniquely nuclear. For very specificalized and complex areas, such as seismic analysis, we will most likely continue to employ outside consulting assistance. We believe that a utility must have an in-depth expert knowledge and involvement in technical matters affecting plant operation, and we will direct our resources to developing and maintaining that knowledge and involvement.

(b) Operational Continuity

Nuclear Operations has had full time individuals assigned to the Project Engineering team since design began to ensure operational aspects are factored into the plant. HL&P intends to employ the operating staff with ample lead time for them to learn the plant design and operation. Furthermore, it is HL&P personnel policy to 57
open all new technical staff positions to internal staff, and to encourage transfers within the organization. Thus, engineering and management personnel involved in Allens Creek design and construction phases will be encouraged to transfer to the operational positions as they are available, which will facilitate the transfer of expertise to operation. The Nuclear Operations group will be deeply involved in the preoperational testing, hot functionals and start-up. At the South Texas project, the Assistant Plant Manager is also the Project Start-up Manager.

(c) Contractor Continuity

GE, the NSSS supplier, will provide instruction manuals for various pieces of NSSS equipment. These manuals will include operation and maintenance instructions which will be used as references during formation of the ACNGS Startup, Maintenance, and Operation 59
procedures. HL&P may request additional procedure guidance from GE during all phases of plant construction or operation. This will help ensure that plant operations reflect the engineering expertise in plant design.

In summary, HL&P's internal organization and policies are such that a smooth transition to operation will be facilitated.

(5) Management Overview

The Houston Industries Board of Directors, HL&P's parent Company, exercises top level management overview by authorizing the capital required for the project. The Board of Directors regularly reviews the summary status, progress, and prudence of the project activities.

HL&P's Executive Vice President exercises top level management overview 57
by approving funds to implement project decisions, by approving staffing complements, and by executing contracts for architect engineering services, the nuclear steam supply system, the turbine-generator, and nuclear fuel. The executive officer regularly reviews the project status, progress, and current activities and sets policy for future activities. He has no responsibilities other than nuclear and reports

directly to the President and CEO. The Manager, Quality Assurance, reports directly to the Executive Vice President. HL&P's Vice President - Nuclear Engineering and Construction is the corporate officer responsible for design, and construction of all nuclear generating stations, including Allens Creek. He has no other responsibilities. For Allens Creek, the VP - Nuclear Engineering and Construction authorizes all NRC licensing submittals, establishes the Nuclear Organizational structure and division of responsibilities, and approves the filling of each staff position within the approved staffing complement. He delegates responsibilities within the Nuclear Organization as described in PSAR Section 13.1.1.2.3. He regularly reviews status and progress information, is informed of significant project decisions, issues, problems, and project plans for resolution of issues and problems through reports prepared by the Allens Creek Project and the Principal Contractors.

The Vice-President - Nuclear Engineering and Construction regularly holds a Quarterly Management Review Meeting on Allens Creek. Ebasco executives and GE management are also present at these meetings, thus enabling the management of all three companies to be regularly informed of the project status, management and technical issues and plans for the future. 57

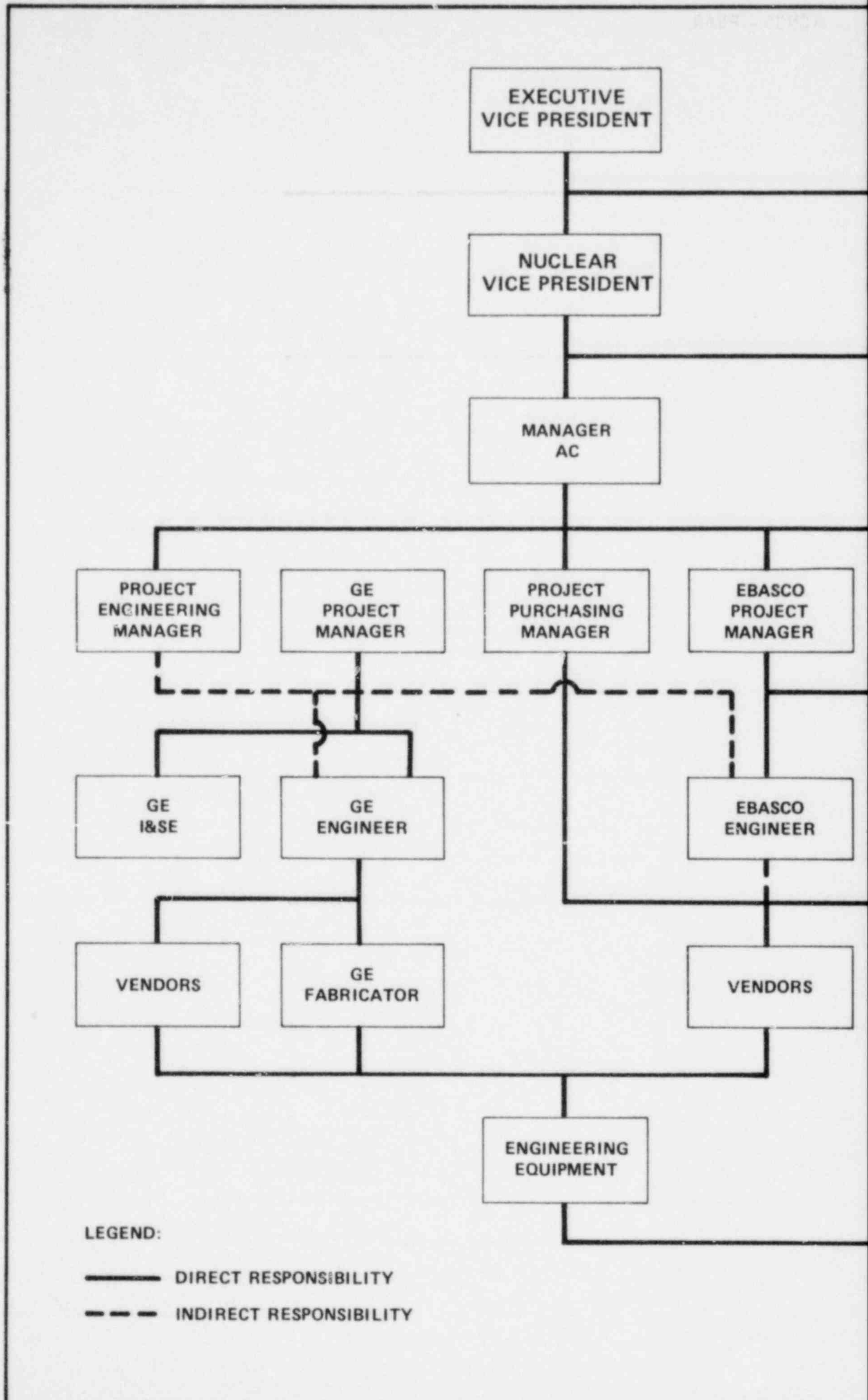
The Allens Creek Project Organization provides Monthly Status Reports to the Vice-President - Nuclear Engineering and Construction, to other HL&P executives, and to the Principal Contractor project managers. These reports identify recent progress, current difficulties, and planned activity over the next reporting period. These reports ensure that top-level executives are aware of Allens Creek Project activities.

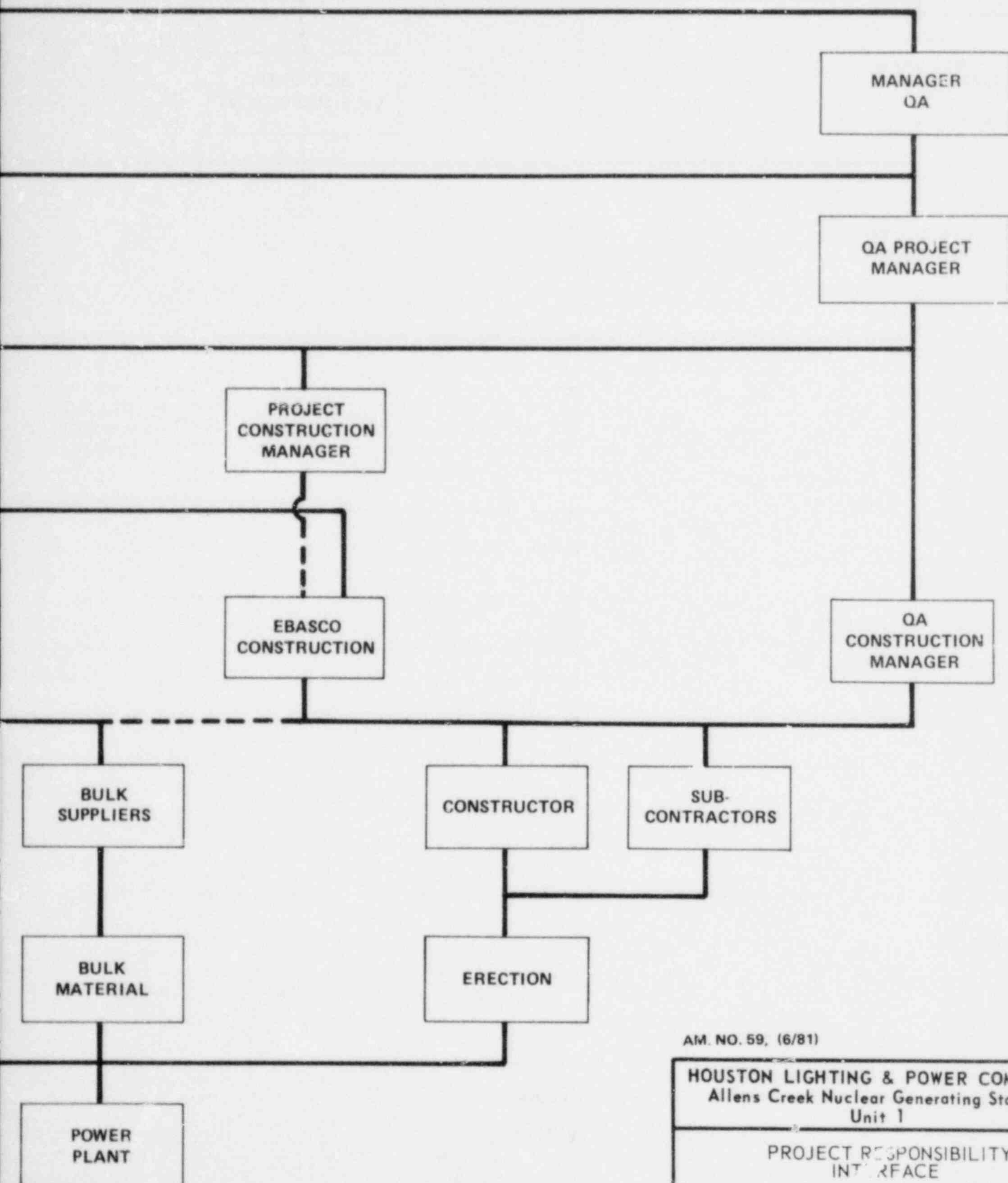
TABLE II.J.3.1-1

ALLENS CREEK ESTIMATED
STAFFING LEVELS

	Jan. 1981	Jan. 1982	Jan. 1983	Jan. 1984	Jan. 1985	Jan. 1986	Jan. 1987	Jan. 1988	Jan. 1989	Jan. 1990	Jan. 1991
Milestone		▲ CP		▲ RPV				▲ Start Test		▲ FL	▲ CO
Staff											
Engineering	29	36	45	45	45	45	45	45	45	40	25
QA	1	9	20	28	34	34	34	33	28	14	9
Construction	0	7	33	33	33	33	33	33	33	33	20
Administration	14	31	41	41	41	41	41	41	41	30	25
Operations	2	4	10	23	41	70	109	168	206	227	227
Total	46	87	149	170	194	223	262	320	353	344	306

59


EXECUTIVE VICE PRESIDENT
NUCLEAR VICE PRESIDENT
MANAGER AC
PROJECT ENGINEERING MANAGER
GE PROJECT MANAGER
PROJECT PURCHASING MANAGER
EBASCO PROJECT MANAGER
GE I&SE
GE ENGINEER
EBASCO ENGINEER
VENDORS
GE FABRICATOR
VENDORS
ENGINEERING EQUIPMENT
LEGEND:
DIRECT RESPONSIBILITY
INDIRECT RESPONSIBILITY



AM. NO. 59, (6/81)

HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Unit 1

PROJECT RESPONSIBILITY
INTERFACE

FIGURE II.J.3.1-1