2.2 LOCATION

The site is in the Town* of Carlton in the southeast corner of Kewaunee County, Wisconsin, on the west shore of Lake Michigan. The city of Green Bay is about 27 miles WNW of the site. Milwaukee is about 90 miles to the SSW. It is located at longitude 87° 32.1'W and latitude 44° 20.6'N, and is shown in Figure 2.2-1. The closest distance to the international boundary between Canada and the United States is approximately 200 miles northeast of the site.

The site as shown on Figure 2.2-2 is all owned by WPS except for the highways and one cemetery site (1.13 acres) located on the highway north of the plant. Total acreage owned as plant site is 907.57 acres.

The cemetery site is owned by and will remain in the ownership of the Town of Carlton with perpetual care provided by the Town. There are no dwellings or public buildings on the cemetery site.

* Wisconsin townships are referred to as Town of......

2.6 HYDROLOGY

2.6.1 SUMMARY

The plant's circulating water is drawn from Lake Michigan. All radioactive liquid wastes generated at the plant are collected, treated and monitored in accordance with 10 CFR 20 so that release concentrations at the circulating water discharge are maintained ALARA. The nearest potable water intake is 11½ miles north at the Rostok Plant intake near Kewaunee. Circulating water released from the plant is diluted by a factor of approximately 60 by the time the water flow reaches the Rostok intake, assuming an average lake current flow of 0.35 ft/sec. This dilution factor is calculated according to Equation (5) in Section 2.6.4.

As mentioned above, normal operation of the plant results in releases ALARA at the point of discharge, consequently, normal operation results in insignificant drinking water radioactivity content at the nearest point of such use. The Point Beach Nuclear Plant wastes, which are also discharged to the lake ALARA, produce a concentration of less than 2E-9 μCi/cc at the Rostok Plant water intake. Consequently, the normal effluent to the lake waters from both plants simultaneously is more than adequately diluted at the water intake near Kewaunee.

2.6.2 GENERAL LAKE HYDROLOGY

The normal water level in Lake Michigan is approximately 577.0 feet, based upon the International Great Lakes Datum, 1955 (IGLD, 1955). The maximum-recorded water level was 582.3 feet in 1986 and the minimum recorded level was 575.4 feet in 1964. At the time IGLD 1955 was established, it was recognized that this common datum would have to be periodically revised due to isostatic rebound, sometimes referred to as crustal movement. Isostatic rebound is the gradual rising or "bouncing back" of the earth's crust from the weight of the glaciers that covered the Great Lakes-St. Lawrence River region during the last ice age. This movement is very gradual and has been occurring since the retreat of the glaciers.

The IGLD was revised to the 1985 standard (IGLD, 1985) when the standard was issued (1992). This new standard affects the reporting of water levels in Lake Michigan. The U.S. Army Corps of Engineers has established a delta of 0.7 feet between the older standard (1955) and the newer standard (1985) due to this rebound effect. Due to the simultaneous movement of the water and landmass, there is no difference in the vertical position of the Kewaunee plant in relation to Lake Michigan. The difference exists in the currently reported water levels in relation to historic values. This is a result of the benchmark elevation changes due to adjustments for crustal movement, more accurate measurement of elevation differences, a new reference zero point location, and an expanded geodetic network. The zero point for IGLD 1985 is at Rimouski, Quebec.

Current, Tides, Waves and Littoral Drift (Reference 1)

On the west side of Lake Michigan, the surface current is largely parallel to the shore and nearly 22° to the right of the prevailing wind. The predominant current direction near the western shore during the period of greatest stratification is in the northerly direction. However, temporary reversals of the general trend may take place (Reference 2).

Current velocity was measured (Reference 3) at 20-minute intervals from August to October, 2-miles off the coast of Sheboygan. The measurements were taken from the surface of the lake down to a depth of 30 feet. The observed persistence patterns for different current velocities are shown in Table 2.6-1. It is fairly certain that this pattern does not differ greatly during the other months of the year.

Tides on Lake Michigan created by the attraction of the moon and sun are insignificant. The total range of oscillation does not exceed 2 inches. However, squalls may raise the surface of the lake by several feet. Deep-water wave heights in the general vicinity of the site due to storms or seiches, and the expected frequency, are shown in Table 2.6-2. Waves are responsible for most of the littoral drift on Lake Michigan. The predominant drift appears to be to the north.

Waves are potentially damaging to the shore structures from impact and run-up. Shore stability is well established as evidenced by the stable location of the shoreline over the long period of time that records are available. Historical publications making reference to the lake commerce at the site occupied by the Kewaunee plant, old photographs, and reports by old-time residents in the area indicate that the shoreline has not changed significantly over the last sixty years. The most recent occurrence of shore erosion was during construction of the plant in 1969. Wave erosion during a severe storm undercut the bank at the promontory protruding into the lake at the southeast end of the site. The damage was repaired and the bank was stabilized with large riprap, which also serves to protect the circulating water discharge.

The shore protection fronting the plant consists of riprap starting at the lake bottom at about Elevation 575.0 feet, a layer of riprap, consisting of face stones about 1,500 pounds to 3 tons each, is laid on the ground rock fill (a mixture of 50 pounds to 150 pounds graded rock and pit run gravel) at a slope of 2.0 horizontal to 1.0 vertical and extends up to a 5-foot-wide promenade at Elevation 586.0 feet.

From the shore side of the promenade a layer of riprap consisting of face stones about 500 pounds each is laid at a slope of 1.5 horizontal to 1.0 vertical on the pit run gravel fill and extends up to the edge of the bank.

Specific gravity of the riprap is about 2.4 with a 2.3 minimum. All riprap stones have a 2% maximum absorption, as per AASHO T-85 with a maximum abrasion loss of 45%.

In addition to the continuous riprap along the shoreline, riprap protection is also installed on both sides of sheet-pile walls of the discharge structure and in the overflow canal immediately in front of the screenhouse forebay.

At Kewaunee, the circulating water screenhouse-forebay structure is the plant structure nearest to the shoreline and is the structure most likely to be affected by waves. The screenhouse-forebay structure is located 180 feet from the normal shoreline. Waves cannot impact directly on the structure. It is possible for wave run-up to reach the screenhouse-forebay structure on occasion. Wave run-up that reaches the screenhouse-forebay structure will have negligible effect and will neither endanger the structure nor adversely affect the operation of the circulating water system. Any water that reaches the screenhouse-forebay structure will spill harmlessly into the screenhouse-forebay through the forebay overflow weir.

Computations of maximum wave run-up are based on information from the Office of the Chief of Engineers (Reference 4). Wave height data given in Table 2.6-2 were used to establish maximum expected run-up and frequencies of occurrence. The run-up at the Kewaunee site is that for a protective beach, which in this case is the submerged and unsubmerged terrain extending from the plant into the lake. The beach is characterized by a rather uniform 1% slope. For maximum run-up there is a "significant wave" height which can be related to the deepwater waves summarized in this section. In general, waves remain intact until bottom influences near shore cause them to break. A wave's energy is transmitted relatively undiminished until it breaks. Upon breaking, energy is rapidly dissipated on the unsubmerged beach.

The analysis in Reference 4 was modified to determine hurricane surge height. This resulted in a maximum surge height of 1.9 feet, produced by the combined effects of wind and pressure. Based upon the study by the Corps of Engineers (Reference 20), the result is considered satisfactory. As previously stated, the maximum recorded lake level in the vicinity of the Kewaunee site is 582.3 feet. This figure in combination with the 1.9 foot storm surge results in a probable-maximum water level resulting from probable-maximum meteorological events coincident with maximum lake level of 584.2 feet. However, since most severe storms occur during the winter months and highest lake levels usually occur during the summer months; the probability of maximum level and maximum storm surge occurring simultaneously is relatively small, and therefore, the analysis is considered to be conservative.

The Atomic Energy Commission (AEC) independently calculated the maximum lake level for Kewaunee to be 589.9 feet (see Reference 21). To accommodate this higher water level the Kewaunee screenhouse was modified during original construction. These modifications included:

1) two bulkhead type doors on exterior access doors to the screenhouse,
2) screenhouse floor covers and manholes to be bolted down,
3) low interior bulkheads,
4) gasketed traveling water screen covers to be sealed and strengthened, and
5) a 4-inch high ramp across the access tunnel to prevent seepage water from reaching the diesel generator room.

These modifications were considered adequate by the AEC to prevent damage to safety-related equipment.

At the Kewaunee site, the "significant deep water wave" is 22.5 feet high and will probably have a period of 11.4 seconds. The wave will break in 28.1 feet of water, which occurs approximately

2000 feet from the shoreline at high water. The resulting maximum run-up, for maximum size waves attendant to probable maximum lake level, is at an elevation of 585.4 feet. The top elevation of the wall nearest the lake is 582.5 feet. This is the crest of the forebay overflow (shown in Figure 10.2-10). The top of the non-overflow section of the screenhouse-forebay is at an elevation of 592.5 feet. These wave run-up computations show that on rare occasions some waves may reach the lakeward wall of the screenhouse-forebay structure. The depth of the water reaching the wall will be minimal and will not contain sufficient energy to cause any structural damage. That part of the wave reaching the lakeward wall will spill harmlessly into the circulating water forebay. No part of the wave will overtop the non-overflow part of the wall.

Investigations were made of the structures that could be possibly affected by the dynamic loads caused by high lake levels. The bottom elevation of the discharge channel is 572.0 feet. Thus, the maximum water depth in the discharge channel is 11.8 feet. Based on the breaking wave theory described in Reference 4, the maximum non-breaking wave that can enter the channel is 9.22 feet, disregarding height limitations imposed by lake bottom topography. By applying the Sainflow method for wave forces due to non-breaking waves described in the same reference, the calculated maximum wave force acting on the discharge structures such as concrete wall and sheet piling, is about 15 psi which is well within the capability of these structures.

Regarding the wave force on the screenhouse structure, the maximum waves, which can penetrate into the forebay, are much lower because of shallow water depth in the overflow channel. The maximum non-breaking wave height reaching the forebay is only 1.90 feet. The calculated dynamic force is less than 1.0 psi, which is well below the force, which this structure can absorb.

The discharge structure, intake crib and screenhouse have been designed for the dynamic forces caused by the probable maximum lake level conditions or conditions which exceed the maximum lake level conditions.

These structures are discussed in greater detail below.

Discharge Structure

The major element of the discharge structure subject to the effects of high water is the sheet pile wall forming the afterbay. The condition determining the design of the sheet piling was the construction condition, which is as follows:

1. Computed back fill (Moist Granular Sand) behind sheets to elevation 582 feet and opposite side excavated to elevation 564 feet. This produced a cantilevered sheet pile design, which was the critical condition.

Since the elevation of the top of the sheets varied from 586.5 feet to 577 feet, it was determined that dynamic forces due to wave action after completion of construction would not be as severe as the construction condition. Dynamic forces due to the maximum lake level condition was not considered during the construction condition because the entire discharge construction work was protected by a cofferdam.

The concrete work of the discharge structure was designed for the following dynamic loading.

1. Ice pressure of 10-kpf thrust due to expansion of an 18-inch thick sheet of ice. This loading was applied to the east side of the structure and is based on information in Vol. 112 ASCE Transactions 1947, Thrust Exerted by Expanding Ice Sheet by E. Rose, utilizing the following assumptions:

    a. Ice Thickness - 18 inches
    b. Solar Energy Considered
    c. Rate of Air Temperature Rise - 10°F Per Hour
    d. Complete Lateral Restraint of Ice Sheet Exists

2. Baffle pier walkway was designed for an uplift pressure of 200 psf due to surge.

3. Baffle wall was designed for a uniformly applied horizontal load of 70 psf due to surge.

Screenhouse

The relative location of the screenhouse with respect to the shoreline eliminated the necessity for applying dynamic load conditions due to probable maximum lake level conditions. Where applicable, the maximum static high water level conditions were considered throughout the design of the screenhouse.

Intake Crib

The intake crib top is about 20 feet below still water level during the probable maximum water level. Therefore, there is no possibility that wave dynamic forces will endanger this structure.

Pack ice, in the form of frozen spray and ice floes, has been reported to a height of 20 feet at the shore by local residents. No measurements of the extent or depth of the pack ice have been made, and no official observations or records have been kept by any agency to verify the reports of local residents. The extent of the pack ice was established by interviewing land owners bordering the site from which it was determined that the maximum offshore extent of pack ice ranges between approximately 800 feet to 950 feet. It is shown in Table 2.6-2 that 17-foot waves may be expected on Lake Michigan once each ten years. If such waves occurred towards the shore at a time of ice break-up on the lake (a very remote possibility), it is conceivable that there would be some ice pile-up on the shore. Experience at three plants of the Wisconsin Electric Power Company on Lake Michigan has shown that no significant problems have arisen from icing as a result of design features incorporated in these plants. The Kewaunee Plant design incorporates features to insure a continuous supply of cooling water.

Lake Temperatures and Effect of Warm Water Discharges

The temperature stratification and circulation patterns of water in Lake Michigan have very distinct characteristics, as follows:

At the beginning of March, a warming trend starts in the lake water and at the end of May all of the water in the lake has reached approximately 40°F, which is the temperature of maximum water density. Until the temperature reaches this point, the surface water is colder than the deeper water in the lake. The colder surface water, which remains at approximately 34°F, is lighter than the 40°F deeper water. This layer of colder water circulates on the surface of the warmer deep water, reaching depths of 25 to 30 feet from the surface.

When all the water in the lake reaches approximately 40°F, the thermocline layer disappears and thorough mixing of the water in the lake takes place. However, when the ambient air temperature warms up the surface water, a thermocline layer is formed again at depths of 30 to 50 feet from the surface.

This occurs from May to July and at this time parts of the water in the lake reach 65°F to 70°F. Consequently, the warmer and lighter surface water circulates above the denser and relatively stagnant 40°F water at the bottom of the lake. This condition continues until a cooling trend starts in September, reaching a peak about the last part of January, at which time the water in the lake again reaches an overall temperature of 40°F. At this time, mixing of the waters in the lake takes place until a colder and lighter layer of surface water starts to build up. Seasonal lake temperatures are given by Church (Reference 5 and 6).

The circulating water intake is a submerged crib-type intake located in approximately 15 feet of water. A thermocline does not exist in the vicinity of the intake since it is located at depths greater than the intake structure. Summertime water temperatures are generally above the thermocline. Historical data for lake water temperatures applied to the Kewaunee site were taken from the city of Green Bay's Rostok intake located near Kewaunee, at approximately 50-foot water depth. The water temperatures at the Rostok intake are generally above the thermocline.

The circulating water discharge facility is an onshore structure discharging at the shoreline and designed for minimum impact on the lake environment. The discharge at the shore edge is from a 40-foot wide channel, 5 feet deep (at normal lake level). Design outlet velocities range from a minimum 2.5-fps to 4.7-fps. The discharge structure provides the termination for the circulating water discharge pipe, a transition from the 120-inch pipe to the open discharge bay, and the outlet to the lake. The discharge bay (or afterbay) receives the discharge circulating water from the submerged pipe transition outlet. At the upstream end, the floor of the discharge bay rises as the sides widen. The downstream portion of the discharge bay is a rectangular channel, 40 feet wide. The discharge bay is normally 5 feet deep but may range from a minimum of 3.4 feet at lowest lake level to 9.9 feet at highest lake level. With two pumps in operation, the discharge is 420,000 gpm but on occasion may be 220,000 gpm with one circulating water pump operating. The discharge flows into the shallow beach area, and generally tends to stratify at the surface. Flow disperses away from the discharge point mixing with the cooler substrata, as water depths become greater. Surface water temperatures will decrease as distance from the plant increases. This apparent cooling is the combined effect of mixing and heat loss to the air. At approximately 1 mile from the plant, surface water temperature returns to within one degree of the lake temperature.

2.6.3 GENERAL SITE HYDROLOGY

Rainfall

Lake Michigan and Lake Huron are considered a unity from the standpoint of drainage and water level since these two lakes are connected. The drainage basin for these two lakes comprises 115,700 square miles and has an average annual rainfall of about 31 inches. The average and maximum precipitation recorded at various locations on the Wisconsin Shore of Lake Michigan is given in Table 2.6-3.

Floods

There are no large rivers or streams in the vicinity of the site. The major part of the site is 20 feet or more above the normal lake level, and there is no record that it was flooded by the lake at anytime.

The small stream directly south of the plant is one of several drainage channels lying in the immediate vicinity of the plant, that drain storm water from a high ridge located some 7,000 feet west. The close proximity of these drainage channels and their associated drainage areas relieves the total maximum floodwater flow to the plant drainage channel.

The maximum probable rainfall may be determined from the one-hundred-year hourly rainfall intensity of 2.5 inches as shown in the "Rainfall Frequency Atlas of The United States", Technical Paper No. 40, U.S. Weather Bureau, which compares favorably with the greatest hourly rainfall shown in the Weather Bureau records for Green Bay, Wisconsin. (Total record available at time of license application was 10 years.)

The maximum hourly rainfall intensity falls on the area drained by the plant channel which is centered between two other channels; one lying immediately north of the plant area and one immediately south. The drainage area is pie-shaped, with its nose at the westerly high ridge, and its base at the Lake Michigan shoreline. The total area is not more than 640 acres.

The drainage channel has an effective length of 1 mile and averages 30 feet in width. The channel only flows during heavy rains. The side contours of the ditch are such that a depth of 4 feet of water can be carried through the plant area without overflowing.

In considering the maximum probable runoff, the rational method was used and was then related to the interval of time, starting from the onset of the period of precipitation for the runoff from the most remote portion of the drainage area. This time interval, when related to a maximum hourly rainfall intensity, results in a rainfall equivalent of 1.75 inches per hour. (From Rouse "Engineering Hydraulics", Chapter IV, Hydrology.)

Thus, using the rational method, the peak run to the drainage channel is 336 CFS. The peak flow that the drainage ditch can handle, without overflowing, is 466.53 CFS. It was concluded that no flooding of the plant could occur from the probable maximum flood flow.

Based on the improbability of flooding from rain and the height of the safety equipment above the maximum lake water level (585.5 feet), it was concluded that flooding is not a problem. Any safety equipment that is located below ground level is further protected by plastic sheeting associated with the concrete construction.

Flooding of the service water pumps, circulating water pump room, and plant access tunnel is not probable. These are shown in Figure 10.2-10. The maximum probable water levels that can occur in the open forebay under the most adverse weather conditions either from pump-trip upsurge (585.5 feet) or from maximum wave run-up (585.4 feet) are below the floor level (586.0 feet) of the service water pump room and access tunnel. The only flood water access to the circulating pump room is from this floor level. Hence, none of these areas are subject to flooding.

A review and re-evaluation of external flooding was performed in response to Generic Letter GL 88-20, Individual Plant External Events (IPEEE) for Severe Accident Vulnerabilities and resolution of generic issue GI-103, Design for Probable Maximum Precipitation (PMP). Using a revised PMP of 16.5 inches per hour, it was concluded that the site continued to have adequate design capability to handle the 100-year hourly rain intensity, which historical experience has not challenged (Reference 26).

Ground Water

Observations of surface drainage and water levels at the site borings indicate that the static ground water level inland from the lake ranges from 10 to 25 feet below the ground surface. The water table at the site generally slopes to the east, indicating a migration of ground water in that direction. At the base of the bluffs, ground water levels are controlled by the elevation of Lake Michigan.

The regional movement of ground water is from west to east. Therefore it is unlikely that discharge into the aquifers at the site would affect any municipal well fields. Fluctuations in the level of Lake Michigan are not of sufficient magnitude to affect the direction of ground water movement. Heavy pumpage from the glacial drift or the Niagara dolomite aquifers in the vicinity of the site would reverse the direction of ground water movement for a distance of only a few hundred yards.

Because of the clay composition of the glacial drift, it is not likely that appreciable amounts of any surface discharge from the plant would seep into the ground. Most of the effluent would flow into Lake Michigan.

The principal water-bearing formations underlying the site are the glacial drift and Niagara dolomite aquifers, which are described in detail in Appendix A.

Potable Water Sources

Lake Michigan is used as the source of potable water supplies in the vicinity of the site for the cities of Two Rivers (13 miles south) and Green Bay (intake at Rostok 11.5 miles north). No other potable water uses are recorded within 50 miles of the site along the lakeshore. All public water supplies drawn from Lake Michigan are treated in purification plants with steps consisting of chemical addition of alum, activated carbon, mechanical mixing, flocculation, sedimentation, filtration and disinfection. The nearest surface waters used for drinking, other than Lake Michigan, are the Fox River at a point 43 miles west and Lake Winnebago 40 miles west of the site.

Ground water provides the remaining population with potable supplies. Public ground water supplies within a 20-mile radius of the site are listed in Table 2.6-4. Additional wells for private use are in existence throughout the rural region.

The sole users of ground water to be found within the general area of the plant are farm residences. No public water supplies, nor any surface water users, are to be found within this area.

However, those users relative to the plant, as shown in Figure 2.6-1, are only those rural wells located in the south half of Sections 23 and 24, in the west half of Sections 26 and 35, and the south half of Section 36 (all in T22N).

No public record of these wells has been made. It is known, however, that about half of the wells within the general plant area use ground water found in a glacial drift that lies about 100 feet below ground level. This drift consists of clayey soils inter-bedded with water bearing sand and gravel out washes. These out washes are irregular and are not continuous at the plant site. The wells that draw from this glacial drift are typically 6 inches in diameter and 100 feet deep.

Each well typically produces about 17 gallons per minute. There are a total of 18 wells that relate to the plant site, of which only 17 are ground water users; therefore, water usage from ground water sources is $(18 \div 2) \times 17$, or 153 gallons per minute, and 220,320 gallons per day.

Fishing (Reference 7)

Commercial fishing in Lake Michigan has decreased in the last twenty-five years due to proliferation of the sea lamprey, causing a reduction in lake trout and an increase in less desirable rougher species of fish. Alewives, chubs and yellow perch accounted for 89% of the 1968 production from Lake Michigan. Efforts are being made by various organizations to reduce the sea lamprey population and increase the abundance of edible fish.

Fishing is practiced generally throughout the lake. Fishing depths are greater than 12 fathoms (72 feet). These depth restrictions place the fishing grounds at least 5 miles offshore. Inshore fishing is licensed occasionally when alewives (a shad-like food fish) are schooling in along the shore. This fish is used mostly for fertilizer and fishmeal manufacture.

Fishing in Lake Winnebago (40 miles west of the site) is confined primarily to rough species; most of which go to mink ranchers in the area for use as animal food.

Sport fishing is one of Wisconsin's prime tourist attractions. It may be considered as existing throughout the state and along all shoreline areas of the lake. Brown, rainbow, lake trout, chinook and coho salmon accounted for 95% of the sport fishing catch in 1980.

2.6.4 DILUTION AND DIFFUSION IN LAKE MICHIGAN

Water from Lake Michigan is used extensively for municipal and domestic water supplies. As described in Section 11, all radioactive liquid wastes generated at the plant are collected and treated for possible reuse and monitored before being discharged from the site. All liquid waste is released consistent with KNPP's ALARA commitment before it reaches the nearest water supply intake. The nearest municipal and domestic water intakes are located at Rostok and Two Rivers (approximately 11.5 miles north and 13 miles south of the site, respectively).

Radioactivity discharged to the plant circulating water can occur in two modes. The first is the normal controlled release of small amounts of activated corrosion products and fission products into the circulating water stream. The second, conceivable only as a result of an operating error or equipment failure, may be regarded as a short-term release before the waste release is shut off.

Computational models for evaluating the dilution of both types of radioactive releases are discussed below.

Short Term Release

A number of diffusion relationships have been derived to describe diffusion in large bodies of water. A widely used relationship is that derived by Okubo and Pritchard (Reference 8):

$$S(r,t)=\frac{M}{\pi D(Pt)^2}\exp-\frac{(r^2)}{(Pt^2)} \qquad (1)$$

Where:

S(r,t)= concentration as a function of time and distance,

$$\frac{\mu Ci}{cc}$$

M = total activity release, µCi

D = depth of mixing layer, cm

P = diffusion velocity, cm/sec

r = distance downstream from release point at which S is determined, cm

t = time after start of release, sec

Experimental measurements in Lake Ontario for the Ginna Nuclear Station resulted in estimates of the diffusion velocity ranging from 0.2 to 2 cm/sec.

Based on studies of Lake Michigan currents and water masses (Reference 1) it was determined that the mixing depth of the lake is 25 to 50 feet, depending on the time of the year.

For the purposes of this analysis, it was assumed that:

P = 0.5 cm/sec
D = 10^3 cm

Furthermore, since the conditions of most interest are those that will transport the radioactive material along the shore rather than into the open reaches of the lake, the equation for concentration is multiplied by a factor of 2. This factor accounts for the restricted diffusion in the direction of the shore.

The peak concentration at any given time can be assumed to exist at the center (origin) of the drifting plume and is a function of time only:

$$S_{peak} = \frac{2M}{\pi D (Pt)^2} \tag{2}$$

The velocity of the current and its persistence at various speeds has been discussed previously (Section 2.6.2). An average velocity calculated from these values is approximately 0.35 ft/sec. The peak concentration as a function of distance from the site, assuming this average current velocity, is given in Table 2.6-5.

As required by 10 CFR 20, the annual average concentrations of unknown radionuclides in unrestricted areas must not exceed 2E-9 µCi/cm^3. It may be seen that short-period release of radioactivity at the site will be diluted at the nearest municipal water intake (11.5 miles) to a peak concentration of 8.54E-14 µCi/cm^3 per µCi of activity released. Furthermore, it should be noted that the above concentration would be a transient value and not the average concentration, which would enter the water intake.

Normal Release

From the relationship used in the previous section for diffusion of an instantaneous release, it is possible to obtain an expression for the concentration from a continuous release as follows:

$$S(y,r) = \frac{2Q}{2\sqrt{\pi}\, PDr} \exp - \frac{y^2}{(Pt)^2} \qquad (3)$$

Where:

S(y,r) = Concentration as a function of cross plume and distance,

$$\frac{\mu Ci}{cm^3}$$

Q = Release rate, Ci/sec

P = Diffusion velocity, cm/sec

r = The distance downstream from release point at which S is determined, cm

D = Depth of mixing, cm

y = Cross plume point at which S is determined, cm

t = Plume travel time to reach distance r with average current velocity, sec

At a given distance r, the concentration S equals zero initially (t=0), but eventually a saturation condition is reached, corresponding to a maximum condition S_{max}, which will exist as long as the radioactive material is released at a constant rate. Under these conditions, S_{max} is a function of distance only. The maximum concentration occurs at the centerline of the plume and, thus:

$$S_{max} = \frac{Q}{\sqrt{\pi}\, PDr} \qquad (4)$$

The maximum concentrations per unit activity release for various distances are shown in Table 2.6-6.

The dilution factor DF (y,r) is given by

$$DF(y,r) = \frac{A}{S(y,r)} = \frac{\sqrt{\pi}\, PDr}{V} \qquad (5)$$

Where:

Q = AV

V = Discharge volume in cc/sec

A = Activity concentration µCi/cc

Using equation (5), it is calculated that a continuous discharge of radioactivity from the plant would be diluted by a factor of approximately 60 by the time the flow reached the nearest municipal drinking water intake, based on a 420,000 gpm circulating water flow.

The effluent from the Point Beach Nuclear Plant (4.5 miles south of the site) has not created any significant problems. Although lake flow is normally in the direction from the Point Beach site toward the Kewaunee site, the concentration of any radioactivity in the effluent from the Point Beach Plant will be diluted by a factor of 35 by the time the effluent reaches the Kewaunee Plant intake, based on a discharge flow from the Point Beach plant of 300,000 gpm.

5.2 CONTAINMENT SYSTEM STRUCTURE DESIGN

5.2.1 REACTOR CONTAINMENT VESSEL DESIGN

Design Conditions

The Reactor Containment Vessel is designed for a maximum internal pressure of 46 psig and a temperature of 268°F. The Reactor Containment Vessel design internal pressure as defined by ASME Boiler and Pressure Vessel Code is 41.4 psig.

The vessel is 105 ft. inside diameter, 206 ft. high, and contains an internal net free volume of 1,320,000 ft^3.

The vessel plate nominal thickness does not exceed 1½" at the field welded joints so the vessel, as an integral structure, did not require field stress relieving. The hemispherical dome is ¾" thick and the ellipsoidal bottom is 1½" thick. A polar crane of 230 tons capacity is mounted on a girder attached to the vessel wall. Reinforcing plates at penetration openings exceed 1½" in thickness; however, these were fabricated as penetration weldment assemblies and were stress-relieved before they were welded to adjacent vessel shell plates.

The following loadings were considered in the design of the Reactor Containment Vessel, in addition to the pressure and temperature conditions described above:

- Dead Loads
- Design Basis Accident (DBA) Loads
- Operating Loads
- External Pressure Loads
- Seismic Loads
- Foundation Deformation Loads
- Internal Test Pressure
- Thermal Stresses (in steel shell due to temperature gradients)
- Pipe Reaction and Rupture Forces

Design Leakage Rate

The Reactor Containment Vessel, including penetrations, is designed for low leakage. At the completion of erection, the Reactor Containment Vessel was tested with the penetrations capped. The measured leakage rate was 0.02% of the Reactor Containment Vessel's net free volume at a nominal internal pressure of 46 psig. The Reactor Containment Vessel was re-tested with all penetrations installed to assure that the leakage requirements as set forth in the analysis of Section 14, have been met. Testing is described in Section 5.7.

Design Loadings

Dead Loads

Dead loads consist of the dead weight of the Reactor Containment Vessel and its appurtenances, the weight of internal concrete, and the weight of structural steel and miscellaneous building items within the Reactor Containment Vessel.

Weights used for dead load calculations are as follows:

♦ Concrete:	143 PCF
♦ Steel Reinforcing:	489 PCF using nominal cross-section areas reinforcing bar sizes as defined in ASTM A 615
♦ Steel Containment Vessel:	489 PCF

Design Basis Accident (DBA) Loads

This load was determined by analysis of the transient pressure and temperature effects that could occur during the Design Basis Accident (see Section 14.3.4).

Operating Loads

Operating loads include the following:

- Gravity loads from all equipment and piping,
- Weight of water in the refueling cavity,
- Weight of crane,
- Loads resulting from the restraint of the free movement of the vessel at the line of embedment in concrete,
- Piping reactions at nozzles resulting from thermal movement.

The analysis of piping reaction forces acting on the steel shell penetration nozzles is based on the algebraic summation of the loading and movements derived from the analysis of the containment vessel, interior and exterior structures, and the attached piping systems.

This analysis included:

1. Piping systems under dead load, live load, thermal, and seismic conditions.

2. Containment vessel and interior and exterior structures under dead load, live load, thermal and seismic conditions.

The analysis considered the interactions between the exterior and interior structures and the containment vessel and included any reversal of penetration reaction forces that could occur.

Major piping systems are not anchored to the containment shell, but are allowed to move freely through the shell penetration. Indirect piping loads on the penetration bellows that are used to seal these systems at the containment boundary have been included in the containment vessel penetration analysis and design.

Piping systems that are directly attached to the containment shell (without expansion bellows) are of small pipe sizes and are of such geometry and design as to be flexible systems imposing minimal loads on the containment shell.

The moment, shear, and radial thrust structural capabilities of all the vessel shell penetrations have been analyzed by the Chicago Bridge and Iron Co. and have been provided and documented in the Containment Vessel Stress Report on file at the site.

The appropriate combinations of loads, moments, shears, and radial thrusts have been applied to the vessel penetrations in conjunction with the operating load parameters of the piping system analysis. The resulting stresses were found to be within suitable margins of the specified design criteria so as to preclude any damage to the steel shell or any anchorage system.

Equipment loads used were those specified on the drawings supplied by the manufacturers of the various pieces of equipment. Floor loadings for the design of internal slabs are consistent with their intended use.

External Pressure Load

During normal operation, annulus pressure will be essentially ambient barometric pressure. The Reactor Containment Vessel's shell plates are of suitable thickness to meet the specified internal pressure requirements and are capable of withstanding an external pressure differential of 0.8 psi in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code.

Automatic vacuum relief devices are used to prevent the Reactor Containment Vessel from exceeding the external design pressure in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Article 17.

During and following the Design Basis Accident, the annulus pressure will not exceed 1-inch water column positive or 5-inches water column negative. These values include results of analysis examining the effects of a single failure in the Shield Building Ventilation System.

Two valves in series are used in each of two lines to the annulus. One valve is actuated by a differential pressure signal and is independent of electrical power. The second valve is self-actuating. The vacuum breaker system is sized to provide air flow into the Reactor Containment Vessel at a rate to meet the following conditions, assumed to occur simultaneously without exceeding the maximum internal pressure differential (vacuum) of 0.8 psi below the external pressure:

a. Atmospheric pressure increasing at a rate equivalent to the maximum recorded by the nearest weather station,

b. Internal air being cooled at the maximum rate expected with all four fan-coil units and both internal spray units operating, with cooling water temperatures of 55°F for spray units and 32°F for fan-coil units;

c. No heat energy entering the containment atmosphere.

The vacuum breaker system has sufficient capability to meet single-failure design criteria, and is capable of proper functioning during a Design Basis Earthquake.

Seismic Loads

Seismic loads were computed using the following:

a. Operational Basis Earthquake seismic ground acceleration of 0.06g horizontal; (see Appendix A)

b. Design Basis Earthquake seismic ground acceleration of 0.12g horizontal.

A vertical component of 2/3 of the horizontal ground acceleration is applied simultaneously with the horizontal acceleration.

The Reactor Containment Vessel earthquake design included the seismic effects of the inertial mass of the air locks and equipment hatch, and the seismic effects of the air locks vibrating as an independent system. The independent vibration effects are considered to act in two directions:

a. Along the longitudinal axis of the air lock

b. In the rotational direction about the point of support on the vessel shell.

The seismic effects of the inertial mass of the crane is included in the Reactor Containment Vessel earthquake design.

The plots of the seismic response spectra are shown in Appendix A. The classification of plant structures and equipment and the applicable damping factors are shown in Appendix B.

Foundation Deformation Loads

During grouting, while the Vessel was supported on temporary columns, deformations of the base slab due to the weight of grout were not imposed on the Reactor Containment Vessel. Deformations of the base slab at this time were accommodated by the manner in which the grout was placed.

The bottom internal deck structure that was fitted to the inside contour of the vessel bottom, the grout base that confines the outside of the vessel bottom, and the heavily reinforced slab foundation mat formed a stiff integral structural system capable of transmitting all internal building loads directly into the supporting soil without relative deformation of the system.

Internal Test Pressure

The vessel was designed to be internally pressure tested on temporary supports. The vessel can also be tested to 46 psig at any time during its service life with the reactor shutdown. Prior to reactor operation the vessel was tested at an over-pressure of 51.8 psig.

Pipe Reaction and Rupture Forces

Pipe ruptures were postulated in the high pressure portions of all piping systems and the resulting jet forces considered in the design of the containment vessel and vessel penetrations.

In the design of the vessel shell, consideration was given to potential hazards from jet impingement resulting from ruptures of adjacent piping. The force of the jet impinging upon the vessel was computed as a function of distance from the hypothetical rupture. All high pressure piping within containment was examined to assure that the selected routings imposed no potential hazard.

The combination of loading which are to include pipe rupture forces (faulted condition), and the associated stress limits, are given in Appendix B. The load combinations to be considered are given in Table B.7-5 and the associated stress limits given for pressure vessels in Table B.7-2.

Thermal Stresses in Steel Shell Due to Temperature Gradients

The steel shell in the knuckle region was designed for the combined pressure and temperature gradients present.

Codes

Design Codes

The design, fabrication, inspection, and testing of the Reactor Containment Vessel comply with the requirements of the ASME Boiler and Pressure Vessel Code, Section II, Materials; Section III, Nuclear Vessels, Subsection B, Requirements for Class B Vessels, and applicable paragraphs of Appendix IX, Nondestructive Examination Methods; Section VIII, Unfired Pressure Vessels; and Section IX, Welding Qualifications.

The Reactor Containment Vessel design and construction meet all the requirements of state and local building codes.

Vessel Classification

The Reactor Containment Vessel is a Class B vessel as defined in the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels N-132.

Code Stamp

The Reactor Containment Vessel is code-stamped for pressures of both 46 psig and 41.4 psig in accordance with Paragraph N-1500, ASME Boiler and Pressure Vessel Code, Section III.

Materials

The Reactor Containment Vessel is fabricated of SA 516 Grade 70 steel plate meeting SA 300 requirements except that impact test requirements are as specified in the ASME Boiler and Pressure Vessel Code, Section III, N-1211 (a) for a minimum service metal temperature of 30°F.

Charpy V-Notch specimens (ASTM A 370 Type A) used for impact testing materials of all product forms were in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, N-330. All material except austenitic stainless steels or non-ferrous metal associated with the Reactor Containment Vessel have an NDT temperature of at least 0°F or less when tested in accordance with the appropriate code for the material.

The containment vessel has been specified to have Charpy V-notch temperature requirements not higher than 0°F. Charpy V-notch test data which demonstrate that the nil-ductility transition temperature is at least as low as 0°F are to be found in a voluminous record of impact test data on file at the site.

The Shield Building, as described in Section 5, protects the entire containment vessel from direct exposure to the outside atmosphere. An analysis of temperature gradients across the shell itself, the bulk air within the annulus, and the Shield Building (assuming an outside temperature of -20°F) shows that an internal containment air temperature of 38°F results in a shell temperature of 30°F. Containment air temperature of 38°F or less is not considered credible for the following reasons:

a. Containment air temperatures are periodically monitored to verify proper operation of the containment air cooling system and, if necessary, manual action can be taken to prevent overcooling.

b. The heat load from internal equipment within containment is approximately 4E+6 Btu/hr, while the heat loss to the surroundings at -20°F is 4.7E+5 Btu/hr. This suggests that the containment air could be heated at a rate of 3.8°F/min if the fan coil units are not removing heat. Thus the operating equipment provides an excellent heat source for assuring an adequate operating temperature of the containment vessel shell.

c. The containment purge and ventilation system heaters are designed to supply 75°F air at a rate of 33,000 CFM. The equivalent net heat input of this system is 1.3E+6 Btu/hr with respect to the lowest permissible containment air temperature 38°F. Since this heat input is almost a factor of 3 higher than the heat loss of 4.7E+5 Btu/hr, the containment purge system heaters are capable of maintaining containment air temperature above 38°F when the reactor is shutdown, even if the fan coil units are in operation.

d. Normal maximum and minimum daily temperatures at Kewaunee are about 20°F and 0°F in January. Therefore, the assumption of constant outside temperature at -20°F is conservative, in spite of a few hours of extreme minimum at -40°F being recorded in January.

In the event of a LOCA, the initial containment ambient air temperature will exceed 40°F, as specified under the requirements for the containment system.

For any break, which released the stored energy of the reactor coolant, the air temperatures in the containment and in the annulus will remain above NDT considerations until long after the Shield Building Ventilation System is in the full recirculation mode.

Beyond this time, the containment air temperature can readily be maintained above 40°F by deliberately not overcooling the containment atmosphere with the post-accident heat removal systems. During eventual containment purge, the purge air heaters can be used if the outside air requires warming.

Calculations based on -20°F outside air and 32°F cooling water temperatures, heat input only from the containment, 200 cfm of annulus in-leakage, and a combined convective-radiative heat transfer coefficient of 1.5 Btu/hr ft^2 °F yield the following values:

- Annulus air temperature 25°F
- Containment air temperature 50°F
- Minimum penetration temperature 32°F (Fan-coil coolant inlet)

These values assure that the containment shell NDT limit of 0°F will not be approached or breached.

The lower portion of the shell, which is embedded in concrete, is also afforded significant protection by the Shield Building.

Of the entire lower shell, the portion in the transition zone between El. 584'-0" and El. 605'-6", as shown in Figure 5.2-1, is the most critical because of its minimum distance to the outside surfaces. Analysis indicated that the shell between these two elevations would be at 30°F or higher if the outside temperature is constant at -20°F and the inside temperature is 34°F or higher. As previously discussed, the inside temperature will always be maintained above 34°F. Therefore, the containment vessel temperature in this region could never be below 30°F in normal operation.

The reactor will not be operated unless the containment vessel air temperature exceeds 40°F. This will assure a minimum 2°F margin above the lowest critical temperature of 38°F. As stated earlier, a condition where containment air temperature is 38°F or less is not considered credible.

Code Requirements

The design internal pressure for the Reactor Containment Vessel is as specified in the provisions of the "Winter 1965 Addenda" to Section III of the ASME Boiler and Pressure Vessel Code.

The design requirements for Class B vessels are contained in Paragraph N-1311 through N-1314 of the Addenda. Paragraph N-1312 states that the design internal pressure may differ from the maximum internal pressure but may not be less than 90% of the maximum containment internal pressure. A maximum internal pressure of 46 psig and a design pressure of 0.9 x 46.0 = 41.4 psig have been specified.

The Reactor Containment Vessel has been pressure-tested for acceptance of the vessel, air locks, equipment door and all vessel penetration nozzles in accordance with the rules of Section VIII, UG-100 and Section III, N-1314(d). The maximum test pressure was 1.25 times the design internal pressure i.e., 1.25 x 41.4 = 51.8 psig.

In-Service Inspection

The basis of the In-Service Inspection Program is ASME Boiler and Pressure Vessel Code, Section XI, 1992 Edition including 1992 Addenda Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants".

Design Stress Criteria

The Reactor Containment Vessel will retain the capability to restrict leakage to the acceptable specified level under all conditions of loading that might occur during its lifetime. The vessel is designed to exhibit a general elastic behavior under accident and all earthquake-loading conditions. No permanent deformations due to primary stresses are permitted in the design under any conditions of loading.

For the Operational Basis Earthquake, designated as loading Condition 1 in Table 5.2-1, the Reactor Containment Vessel is capable of continued safe operation during a DBA. For this loading condition, the structure will function within the normal design limits specified by Section III of the ASME Boiler and Pressure Vessel Code, Figure N-414 and as listed in Table 5.2-1. Loading Condition 1 provides the design basis upon which the Reactor Containment Vessel is code-stamped.

For the Design Basis Earthquake, designated as Loading Condition 2, the margin provided in the design assures the capability to maintain the vessel in a safe operating condition. For this loading condition, the basic design was reviewed to insure that the Reactor Containment Vessel and its components retain their capability to perform their containment function.

Primary stress intensities are conservatively limited to 90% of the yield strength of SA 516-GR70 carbon steel plate at the accident temperature. The application of this criterion to Table N-414, ASME Boiler and Pressure Vessel Code, Section III is shown under Loading Condition 2 of Table 5.2-1.

Earthquake stresses are added linearly and directly to stresses caused by the Design Basis Accident, dead loads, and the appropriate operating loads to obtain the total stresses. These total stresses are within the maximum stress limits allowed by the design criteria, listed in Table 5.2-1.

Prior to the special ruling of ASME Code Case 1392 (Reference 1). The interpretation of the ASME Code for the design of containment vessels was to treat all vessel configurations and loading under the design rules of Section III(b) and satisfy the basic stress intensity limits of paragraphs N-414.1, N-414.2, N-414.3 and N-414.4 of Section III.

The interpretation presently permitted by this code case is to accept the design provisions of Section VIII in the absence of substantial mechanical or thermal loads other than pressure.

The design rules of Section VIII are satisfied for all configurations and loadings explicitly treated by Section VIII, using "Sm" [See N-1314(b)] in place of "S" in the various formulas.

The Reactor Containment Vessel in this application is applicable to the requirements of ASME Code Case 1392. The minimum thickness of the bottom configuration is predicated on the design rules of Section VIII.

In consideration of the large diameter of the vessel the shell bottom was analyzed by the "Yale Shell Program" (Reference 2).

Circumferential compressive stresses resulting from internal pressure forces were calculated and held below the critical buckling stress by a margin of safety compatible with good pressure vessel design practice.

The design was reviewed to assure that any resulting deflections or distortions would not prevent the proper functioning of the structure or pieces of equipment and would not endanger adjacent structures or components. The ASME Boiler and Pressure Vessel Code provisions, for out-of-roundness tolerance was not considered appropriate for the cylindrical vessel of the magnitude of the Reactor Containment Vessel. Therefore, it was specified to have an out-of-roundness tolerance of not greater than one half of the ASME Boiler and Pressure Vessel Code permissible tolerance, i.e., one-half percent of the normal diameter.

All other applicable tolerances of fabrication and erection as specified in the ASME Boiler and Pressure Vessel Code were applied to the Reactor Containment Vessel.

The ellipsoidal bottom of the Reactor Containment Vessel is bonded to and in intimate contact with the support grout under the vessel bottom. It is noted that those surfaces internal and external to the vessel that are in contact with concrete will not be readily available for inspection.

This concept of Reactor Containment Vessel support and internal concrete construction is recognized by the ASME Boiler and Pressure code and has been approved on numerous nuclear power plants.

The successful application of this method of support is predicated on the inherent built-in quality of construction associated with pressure vessels, and the test requirements of acceptance and code certification.

All weld seams on the bottom are fully radiographed and have been leak tested at 5 psig and also at the design internal pressure.

Design Review and Analysis

The Reactor Containment Vessel was designed, fabricated, constructed, and stamped in accordance with the rules of Section III, Subsection B, of the ASME Boiler and Pressure Vessel Code.

The bottom head of the Reactor Containment Vessel was designed using the formulas of Section VIII of the ASME Code as allowed by Paragraph N-1314(a)(1), (Summer 1968 Addenda) when substantial mechanical or thermal loads other than pressure are not present.

Temporary stiffeners were added to protect the structure during the overload pressure test and prevent the possibility of any damage to the vessel during construction. Following the overload pressure test, the temporary stiffeners on the bottom head were removed by arc air gouging the metal to within ¼" of the vessel plate surface. Then grinding the remaining metal smooth to the surface of the vessel, to prevent damage to the internal concrete or the vessel that could result from movements of the vessel. After removal of the stiffeners, the area was examined using the magnetic particle method.

The Reactor Containment Vessel's concrete fill support under the ellipsoidal bottom was placed using a concrete placement and grout. The grout is a two component epoxy polysulfide material containing no solvent and does not shrink. The mixture consists of an epoxy resin, an organic amine, and a polysulfide. It is chemically inert to the steel and concrete and therefore will not promote corrosion of these materials. The technique used is as follows:

a. The concrete fill was placed in a predetermined sequence of pours. The size, location and timing of placement of the individual pours were all considered in determining the pattern and sequence of pours.

b. The placement of low-viscosity chemical grout proceeded in sequence during the above operation after the concrete grout had completed its cycle of hydration and shrinkage. The chemical grout was pressure injected through grouting ducts. Injection continued at each duct until the chemical grout appeared around the periphery of the pour. This method ensured full bearing under every part of the Reactor Containment Vessel's ellipsoidal bottom.

After embedment in concrete, the bottom head is subject to both thermal and pressure loads not considered by the formulas of Section VIII. A detailed thermal study and stress analysis was performed to show compliance with the stress intensity limits of N-414.1, N-414.2, N-414.3 and N-414.4 as required by N-1342(a)(2). This analysis is of the type normally required for a Class A vessel.

The top of the internal concrete and the interior surface of the steel shell above the floor has been coated with a nuclear grade coating capable of withstanding a DBA. At the intersection of the concrete floor and the steel shell, a sealant was provided to prevent moisture from penetrating the joint.

The exterior concrete and the steel shell have been chemically bonded tight to assure no penetration of moisture. Above the chemical grout at the "air gap" between the steel shell and the concrete, the steel surface has been sandblasted, primed and coated with a rust inhibitive, high temperature paint. The "air gap" was formed with two layers of styrofoam with all joints staggered and taped and concrete poured against it.

The top of the concrete in the annular space has been provided with sumps and a concrete curb all around (Figures 5.2-1 and 5.2-2) to collect any moisture.

The space between the concrete curb and the steel shell was caulked with a sealant to assure that moisture will not penetrate into the air gap.

Reactor Containment Vessel at Embedment Region

In its final configuration, the containment shell knuckle is embedded by the internal concrete to a point 2 feet-3 inches above the shell tangent line and externally to a point approximately 2 feet-9 inches below the tangent line. A transition zone exists in a region that extends from the top of the internal concrete to some point on the shell below the temporary stiffeners. This condition is shown in Figure 5.2-1.

Embedment of the Reactor Containment Vessel knuckle in concrete produces bending stresses, resulting from thermal and pressure expansion, at the interface between the encased and non-encased portions of the shell. These bending stresses were minimized by providing a smooth transition between the part of the shell, which is free to expand and the part, which is fixed in concrete.

Figures 5.2-1a and 5.2-1b show the typical arrangement of the reinforcing steel patterns for the base slab and at the lower portion of the wall. The reinforcement in the discontinuity zone of the wall with the base is shown in Figure 5.2-1a.

Figures 5.2-1c and 5.2-1d show the typical arrangement of the reinforcing steel pattern for the dome and the upper portion of the wall. The reinforcement in the discontinuity zone of the wall with the dome is shown on Figure 5.2-1c.

Figure 5.2-1e shows the typical arrangement of the reinforcing steel pattern at a typical large opening. In this illustration the opening was provided for an airlock. It is to be noted that where the general pattern of wall reinforcement has been interrupted by an opening additional perimeter beam reinforcing has been added.

Since the Shield Building is not subjected to any significant internal pressure, reinforcement bar anchorage will generally not occur in tension zones.

When the Reactor Containment Vessel is pressurized, in operational configuration, it will exert a pressure on the internal concrete in the knuckle region as the shell attempts to deform inward. Also, the vessel will exert a pressure on the concrete outside of the vessel where the elliptical head is tending to deform outward. These reactions on the concrete are due to the tendency of the elliptical head to become hemispherical in shape when pressurized.

The analysis of the concrete-steel shell interaction in the embedded zone is basically a flexibility method of analysis of the concrete-steel structural systems.

The primary objective of the analysis is to determine stresses in the steel shell and the contact pressures on the concrete from the steel in the embedment region.

Three structures are used in the analysis:

- Internal concrete
- Steel shell, and
- External concrete

The model of the ellipsoidal bottom shell structure is divided into carefully selected segments to best represent the shape of the model.

The restraint of the concrete on the steel shell is modeled using "Analogous Springs" and a distributed pressure. The "Analogous Springs" are represented by a flexibility model of both the internal and external concrete supplied by Pioneer Service & Engineering Company. This model is in the form of load per unit length of circumference per unit of concrete displacement and is given for various positions on a meridian line of the shell bottom. The pressures applied at the selected loading points of the shell bottom are equal to the algebraic sum of the internal pressure plus the reaction pressure of the concrete. The pressure distribution between loading points specified in the program is assumed to be linear.

Continuity and interaction are established by assuring that the deflections are equal at the points where concrete is in contact with the steel shell. Equations for the internal and external and contact points are constructed and solved for redundants. A trial and error procedure is used for the solution since the equations corresponding to the redundants with negative contact pressure must be eliminated.

A. Kalnins' method (Reference 2) is used to develop the equations to be used in the flexibility analysis; i.e., deflections, shears, and loads.

The compatibility of deformations of the structures is amply demonstrated when the solution of the flexibility model is carried to the point where the deflections of structures in contact are equal. This has been done and the final internal and external contact pressures of the concrete on the steel are in equilibrium with the structures as a free body for axisymmetric loads. There are no non-axisymmetric loads considered acting on the bottom in this analysis.

Concrete and Steel Interface Design

The final stress report demonstrated that the Reactor Containment Vessel, as specified, analyzed, and designed, meets all applicable requirements of the ASME Code.

As a further verification of the design, a second overpressure test (1.25 times design pressure), not required by ASME Code, was performed at a pressure of 51.8 psig with the temporary vessel stiffeners removed and the internal concrete support system in place, but before fuel was loaded. The maximum pressure was maintained only long enough to verify the pressure level.

Non-Axisymmetric Loading Due to Concrete Shielding at Fuel Transfer Tube

An area 43.5 ft high and about 24 ft wide on the inside of the containment vessel is covered by interior concrete walls. This area will be referred to as the "cold spot".

The design basis for the steel containment vessel is that after a LOCA the internal pressure will peak at 46 psi at about 10 seconds, and the internal temperature will peak at 268°F, also at about 10 seconds. After that, both pressure and internal temperature will decline (see Section 14.3.4).

During the first 10 seconds after a LOCA the temperature of the inner surface of the containment vessel shell rises from 120°F to 155°F, and the outer surface rises from 120°F to 129°F. Thereafter, the shell temperature continues to rise to 165°F at the inner surface and 155°F at the outer surface. This increase does not occur in the portion of the shell protected by the internal concrete thus producing the "cold spot" which results in additional stresses in the steel shell. These stresses are classified as Thermal Stresses.

The area behind the "cold spot" is assumed not to be subject to the 46-psi internal pressure. Thus, the pressure around the circumference of the steel shell is unbalanced giving rise to local pressure stresses.

Kalnins' Static Shell Program was used to determine the effects of the missing temperature rise and the missing pressure increase at the "cold spot". For each investigation, 21 Fourier harmonics were combined to represent the sudden changes at the edges of the "cold spot".

Additional stresses due to missing temperature at the "cold spot", after 10 seconds, were obtained and are given in Table 5.2-7.

The circumferential stresses are the maximum stresses at the three elevations listed in the table.

In determining the effect of the missing pressure, cognizance was taken of the fact that the increase in pressure and temperature of the steel shell after a LOCA would cause an increase in diameter. This would result in a theoretical finite gap between the steel shell and the exterior face of the concrete walls. However, the unbalance in forces around the circumference of the shell, due to the missing pressure against the portion of the shell protected by the concrete wall, would cause the steel shell to press against the concrete and thus prevent the gap from opening. At the maximum internal pressure of 46 psi, a pressure differential of 3 psi is required to close the gap. Thus, the steel shell will then press against the concrete with a pressure of 43 psi. The additional stresses in the steel shell were calculated on this basis and are given in Table 5.2-8.

The additional stresses given in the table must be added to the stresses, which would exist if no "cold spot" were present. These stresses are summarized in Table 5.2-9.

The total, combined stresses in the containment vessel steel shell due to the presence of the "cold spot" are summarized in Table 5.2-10. These stresses occur 10 seconds after the start of LOCA. The stress intensity for each loading condition is also shown in this table. The stress intensity is calculated in accordance with Paragraph N-413 of Section III of the ASME Boiler and Pressure Vessel Code (1968 Edition).

The allowable stresses in the containment vessel steel shell were established by the criteria listed in Table 5.2-1. For the Operating Basis Earthquake condition, this table refers to Figure N-414 of Section III of the ASME Boiler and Pressure Vessel Code (1968 Edition). The limits of stress intensities in Figure N-414 are based upon the allowable stress intensity S(m). For Class B vessels, this allowable stress intensity is obtained from Section VIII of the ASME Code, and is 17,500 psi for A-516 Grade 70 steel. This allowable stress intensity is applicable to general membrane stress intensities when local membrane and bending stresses are not present. When the latter stresses are present, the allowable stress intensity is increased to 1.5S(m) or 26,250 psi. When secondary stresses due to differential thermal expansion are included, the allowable stress intensity is increased to 3.0S(m) or 52,500 psi.

Allowable stress criteria for the Design Basis Earthquake are given in Table 5.2-1 under loading condition 2. The allowable stress intensities computed in accordance with these criteria are:

- General membrane = $1.16(S_m)$ = 20,300 psi
- General membrane plus bending = $1.6 \times (1.5S_m)$ = 30,450 psi
- All stresses including = $3S_m$ = 52,500 psi secondary stresses

The OBE and DBE dynamic vertical and horizontal seismic loads and overturning moments of the containment vessel are transmitted into the concrete foundation and sub-foundation through the combined action of friction and bearing on the shell bottom.

The lateral and vertical loads caused by earthquake are of such a minimal nature that the friction is neglected to simplify the analysis and a simplified but conservative approximation of the reaction bearing load is considered.

The seismic overturning moment of the containment vessel will produce vertical reaction pressures on the shell bottom. The horizontal seismic forces will produce lateral pressure reactions on the shell bottom. Since the forces are minimal, a conservative, simplified mode of lateral load transfer and analysis of steel and concrete is made.

First, the load analysis consists of using John A. Blume's modal analysis seismic acceleration curves for the containment vessel. The calculation of the vessel shear, membrane and surface stresses is by the application of the Kalnins' program to shells. The most critical area of stress in the steel shell under these conditions occurs in the discontinuity zone at the line of the external embedment in the concrete. The calculated stresses and allowable stresses at this line are given in Table 5.2-9a. The allowable stresses are based on the criteria of Table 5.2-1.

Next, the total lateral seismic load of the containment vessel is conservatively assumed to be carried by tangential shear above the embedment line into the vessel plates whose planes are most nearly oriented in the direction of the seismic loads. These loads will then be carried by membrane action into a hoop band of limited width at or near the mezzanine floor line as an integral internal member. This lateral load is thereby transferred to the diametrically opposite surface of the internal concrete member. The load is then transferred through the shell plate onto an expanded area of the external concrete foundation.

To further simplify the analysis, this band is conservatively assumed to be 20 in. wide and the resulting stresses and allowable stress criteria are shown in Table 5.2-9b.

The allowable stress intensities are also shown in Table 5.2-10.

It can be determined from an examination of Table 5.2-10 that the presence of the "cold spot" does not result in any case in which actual stress intensity exceeds the allowable stress intensity.

After the first ten seconds following a LOCA, the internal pressure drops and the temperature differential between the two surfaces of the steel shell also declines, but the average temperature of the shell increases another 18°F to 160°F. The change in stresses due to the first two effects is greater than that caused by the third effect. The net result is a decline in total stresses after about ten seconds.

It has previously been pointed out that at ten seconds after a LOCA, the steel shell exerts a 43-psi pressure against the outer surface of the concrete wall. At the same time there is an internal pressure of 46-psi against the inner surface of the concrete wall. The 3-psi pressure differential must be carried by the concrete wall.

During the first ten seconds following a LOCA the pressures do not increase linearly, so the pressure differential could, briefly, exceed 3-psi during this interval. For instance, at one second the interior pressure is 12 psi. This 12-psi pressure was used to calculate stresses in the concrete wall on the conservative assumption that the steel shell had not had time to react and press against the concrete.

STRESS computer program was used to calculate moments and shears, and Pioneer Service & Engineering Co. Program S-020 was used to determine the required reinforcing steel. The amount of reinforcing steel thus determined did not exceed 35% of the steel required by other loading conditions. Thus, the internal concrete structure will not be overstressed by the loads caused by the presence of the "cold spot".

In conclusion, it has been demonstrated that all stress combinations, which include those computed on the basis of non-axisymmetric loading at the "cold spot" meet the criteria for allowable stress for the steel containment vessel and the interior concrete construction.

Testing of the Reactor Containment Vessel on Temporary Supports

The vessel was erected on a temporary support system. To fulfill the requirements of the ASME Code and provide a basis for acceptance of the vessel, the acceptance overpressure test was made while the vessel was supported on the temporary support system. To provide additional safety margin for those engaged in the erection and testing of the vessel, internal stiffeners were provided as a precautionary measure.

The temporary internal stiffeners consisted of four flat bars that were attached by fillet welds to the inside surface of the shell in the knuckle area. These bars are circumferential rings spaced about two feet apart. The stiffener bars are shown in Figure 5.2-1. These stiffener bars were installed to provide lateral stability of the knuckle plate during pressure testing.

This was necessary because the knuckle plate is subject to circumferential compression stress when it is pressurized for testing. After the vessel was acceptance-tested, the temporary stiffeners were removed and the required lateral support was provided by internal concrete structures.

The vessel was temporarily supported on fifteen laterally-braced pipe columns shaped to the contour of the vessel. The columns were welded to the shell plate immediately below the tangent line at the knuckle and also to the horizontal web of an external horizontal tee member. This horizontal tee ring-girder developed the lateral stiffness of the temporary bracing. These temporary supports are shown in Figure 5.2-2.

The temporary structure supported the weight of the vessel (2300 tons) and specified construction loads. In addition, it resisted lateral loads, such as wind, which were imposed during construction. The vessel loads were then transferred to the grout support. The temporary legs, bracing, and external tee member have been removed by cutting to within ¼" of the vessel face and then ground smooth.

Wind Analysis

The Reactor Containment Vessel and penetrations associated with Primary Containment are completely enclosed by the Shield Building and are therefore not directly subjected to the forces and effects of wind and tornadoes.

Seismic Analysis

The seismic analysis of the Containment System, critical appurtenances, structures, and equipment were based on the ground response acceleration spectra, the building response spectra (Reference 3) and the floor response acceleration spectra (Reference 4). For details see Appendix B.

Building Response Spectra

Building response spectral analyses (Reference 3) were performed, based on Operational Basis Earthquake (0.06g) and Design Basis Earthquake (0.12g). The analyses developed values for maximum translational accelerations, displacements, shears and moments and maximum torsional accelerations, moments and rotations of the building structures.

The mathematical model considered the three major structures (Reactor, Auxiliary and Turbine Building) in a combined idealized three-dimensional model with 63° of freedom-translation for symmetrical elements, and both translation and torsional rotation for unsymmetrical and irregular elements.

Floor Response Acceleration

The floor response acceleration spectral analyses (Reference 4) are based on the Operational Basis Earthquake (0.06g) and the Design Basis Earthquake (0.12g). The analyses developed the generated acceleration time-history response spectra at mass points designated for the seismic analysis of critical equipment and piping located throughout the structural complex. The mathematical model utilized for this analysis is the same one used for the building response spectra analysis.

Airlock Seismic Analysis

The specification for the Reactor Containment Vessel requires that the Earthquake design include the effects of the airlocks vibrating as an independent system. The Reactor Containment Vessel contractor, Chicago Bridge & Iron Co., (CB&I) developed an analysis of the airlock system under their seismic consultant, John A. Blume & Associates. The calculations were performed according to this method, and the results show that the stresses in the airlocks do not exceed the allowable stresses given in Table 5.2-1.

In this method of analysis, the horizontal earthquake is assumed to act in one of two directions as shown below. Case I will result in seismic forces and moments from the airlock being applied to the Containment Vessel in the circumferential direction. Case II will result in a radial thrust being applied to the Containment Vessel.

Dynamic Constants for Airlocks

Since the airlocks are completely separated from the concrete Shield Building and adjacent concrete structures by an air space, there will be no interaction between airlock structures and concrete structures.

Case I - Vertical and Circumferential Directions

The spring constant for the seismic analysis of airlocks in the circumferential direction of the vessel shell is evaluated by applying a unit moment (1,000 in-lbs) at the shell and determining the rotation (θ) at the shell-to-nozzle junction (Reference 5):

$$K = \frac{M}{q}$$

Where: K = Spring constant shell

$$W = \left(\frac{K}{I}\right)^{1/2}$$

W = Angular frequency of lock

$$T = \frac{2p}{W}$$

I = Moment of inertia of lock about point of support on shell

T = Fundamental period of lock
M = Unit moment at shell
θ = Rotation at shell to nozzle junction

Case II - Radial Direction

In a similar manner the spring constant in the radial direction of the vessel shell is determined by applying a unit thrust (1,000 lbs) at the shell and determining the deflections (w) at the shell-to-nozzle junction (Reference 6):

$K = \frac{P}{w}$ Where: W = Weight of lock plus insert

g = 386.4 (in/sec^2)

$T = 2\pi \left(\frac{W}{Kg}\right)^{1/2}$ P = Unit load

w = Deflection at shell to nozzle junction

Seismic Forces on Airlocks

The vibrational driving force on the airlocks was determined from accelerations derived from the response acceleration spectra prepared by the seismic consultants, John A. Blume & Associates (Reference 4). Using the fundamental period of the airlock (T) and 1% damping (as recommended by John A. Blume & Associates), the acceleration in percent of gravity is obtained. The airlock dead loads were then multiplied by this acceleration to obtain the seismic forces acting on the airlock.

The applicable ASME Section III allowable stress intensities for stresses due to pressure and applied loads at the containment vessel shell to airlock nozzle junction were calculated as follows:

<u>For Operating Basis Earthquake (OBE)</u>

- Membrane Stress: PL = (1.5) (17500) = 26250 psi
- Surface Stress: PL + Q = (3) (17500) = 52500 psi

<u>For Design Basis Earthquake (DBE)</u>

- Membrane Stress: PL = (1.16) (1.5) (17500) = 30450 psi
  PL + Q = (3) (17500) = 52500 psi

Seismic Stresses

Stresses in the shell due to the airlocks vibrating as an independent system from a horizontal and vertical earthquake were determined by the use of Welding Research Council Bulletin 107 and Chicago Bridge and Iron Computer Program 6-20N.

Stresses were limited to the allowable stress criteria as set forth in Table 5.2-1.

A horizontal earthquake acting perpendicular to the airlock (Case 1) will result in a circumferential shear and moment being applied to the shell. An earthquake acting parallel to the airlock (Case II) will subject the shell to a radial thrust.

Stresses were checked at three locations: at the neck-to-insert-plate junction, at the insert-to-shell junction, and at a distance of $0.5(Rt)^{1/2}$ from any local stress area. (R = radius of curvature and t = thickness).

Stresses in the reinforcing insert-plate and in the shell due to the applied earthquake loads were calculated. Pressure stresses were added directly to the earthquake stresses, and an equivalent stress intensity was calculated (per maximum shear theory) compared to allowable values in Table 5.2-1.

The designs are based on the Operational Basis Earthquake and are reviewed for the case of Design Basis Earthquake to assure that the stress limits for this loading are not exceeded.

A summary of the membrane and surface stresses at the connecting weld of the containment vessel shell plate to the airlock barrel nozzle plate for the two earthquake loading conditions is given in Table 5.2-6.

Vertical Earthquake

A vertical earthquake acceleration applied to the Vessel and airlocks was assumed to act simultaneously with the horizontal earthquake.

Soil - Structure Interaction

The problem of soil-structure interaction was accounted for by introducing rotational and translational springs in the model used for seismic analysis. The rotational springs account for the deformation of the soil under the building, i.e., rocking, and the translational springs account for frictional resistance of the soil on the bottom and sides of the building, as well as soil bearing against the sides of the building. The stiffness of these springs was determined by using elastic finite element method of analysis and checked using equations developed for the case of rigid plate on a semi-infinite elastic half space.

Foundation Damping Factor

The following recommendation on damping factor to be employed for the foundation materials was made by Dr. Ralph B. Peck:

"Damping values for the rocking mode have not been well established for clay soils. Although some laboratory information is available, it seems preferable to use values (expressed as percent of critical damping) that have been back calculated from prototype installations, such as machine bases. Values from 5.1 to 7.1% were determined for a soft clay on this basis by D. D. Barkan (Reference 7) "Dynamics of Bases and Foundations". McGraw-Hill Book Company, New York, p. 126 (1962). The data quoted in Whitman (Reference 8) and Richart "Design Procedures for Dynamically Loaded Foundations", ASCE, Journ. Soil Mechanics and Foundations Division, Vol. 93, No. SM6, Nov. 1967 pg.178, which refer to Barkan's text, p. 67, are based on laboratory tests and may be in error."

"The damping factor can be expected to be smaller for a stiff clay than for a soft clay. On the other hand, the presence of granular material with an average thickness on the order of 10 ft between the clay and the bedrock at the Kewaunee site will introduce considerably more damping than would the presence of an equivalent amount of stiff clay. For this reason, the damping factor would be greater than for clay alone. I would recommend, therefore, the use of a factor of 4% for the design earthquake and 6% for the maximum earthquake. The increase is permissible because the deformations associated with the greater earthquake would involve greater strains and loss of energy".

Dr. Peck's recommendation was the subject of discussion between John A. Blume & Associates (Seismic Design Consultants), Pioneer Service & Engineering Co., and Dr. Peck. As a result of these discussions it was agreed that 5% damping factor would be used for both Operational Basis Earthquake and Design Basis Earthquake. For further details refer to Appendix E.

Penetrations

Design Basis

To maintain designed containment integrity, containment penetrations have the following design characteristics:

a. They are capable of withstanding the maximum internal pressure, which could occur due to the postulated rupture of any pipe inside the Reactor Containment Vessel.

b. They are capable of withstanding the jet forces associated with the flow from a postulated rupture of the pipe in the penetration or adjacent to it, while still maintaining the integrity of containment.

c. They are capable of accommodating the thermal and mechanical stresses, which may be encountered during all modes of operation and test.

d. Materials of piping penetrations furnished as a part of the Reactor Containment Vessel are either ASME SA-333 GR. 6 or ASME SA-312 TP. 304. All hot penetrations are fabricated of ASME SA-333 GR. 6. The cold penetrations are fabricated of either of the foregoing materials, but in all cases are compatible with the material of the process line, which is to be welded directly to the penetration nozzle.

e. The materials for penetrations, including the personnel access airlocks and the equipment access hatch, conform to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Code Case 1392, Rev 0. The process and guard pipes and flued heads were designed, specified and fabricated in accordance with the Code for Pressure Piping, USASI B31.1.0-1967, N-7, and ASME Boiler and Pressure Vessel Code Section III, 1968, Class B, Code cases 1177, 1330, and 1425.

Charpy V-notch tests were performed on the material used for construction of the carbon steel flued heads and penetrations. The test temperature was 0°F with break values based on three material samples from each penetration or flued head. All test data showed that the samples had greater than 20 ft-lbs energy capability. The description of the tests and data are filed at the Kewaunee site.

Electrical Penetrations

The electrical penetration assemblies are designed for field installation by welding to the inside end of the nozzle type penetration passing through the Reactor Containment Vessel wall. Each penetration assembly is provided with a single connection to allow pressure testing for leaks. All components of the penetration assemblies are designed to withstand, without damage or interruption of operations, the forces resulting from an earthquake, in addition to the normal and accident design requirements.

All materials used in the design are selected for their resistance to the environment existing under normal operation and the DBA.

Figure 5.2-3 shows the D.G. O'Brien penetration configuration for the following types of electrical penetrations provided in the design:

- Type I Medium Voltage Power (MVP) - 5000 Volt Insulation for use on 4160 volt Resistance Grounded System - 4 provided
- Type II Low Voltage Power (LVP) - 600 volt insulation for use on 480V grounded system - 18 provided
- Type III Instrument & Control (I&C) - 600 volt insulation - 22 provided
- Type IV Control Rod Drive Power (CRDP) - 1000 volt insulation for use on 140V DC-5 provided
- Type V Nuclear Instrumentation Systems (NIS) Triax Cables - 4 provided.
- Type VI Radiation Monitor Cables (RM) - 1 provided

The electrical penetrations entering the Reactor Building are subdivided into six basic groups having twelve penetration nozzles in each group. Electrical penetration assemblies are installed in the penetration nozzles. Spare penetration nozzles in each group are sealed with a welded cap. Instrument and control leads are segregated from power leads by placement into different electrical penetration assemblies. The circuit arrangements are as follows:

- Group A Normal power and Train "B" power
- Group B Normal power and Train "B" power, control and instrumentation, and "White" Reactor Protection Channel
- Group C Normal instrumentation, Train "B" instrumentation, and "Blue" Reactor Protection Channel.
- Group D Normal instrumentation, Train "A" instrumentation, and "Red" Reactor Protection channel.

- Group E Normal control, Train "A" control and instrumentation, and "Yellow" Reactor Protection channel.
- Group F Normal power and Train "A" power.

In addition to the above, there are penetrations into the two airlocks which are used for door position signal circuits, internal airlock lighting and communication circuits.

The approximate sequence of testing and tests performed on both prototype and production DG O'Brien penetrations are listed in Table 5.2-1a. Detailed descriptions of the manufacturer's test procedures are contained in D.G. O'Brien, Inc. Test Procedure Manual. The test data for all the tests performed are filed at the Kewaunee Reactor site. Since the specifications are different for each application, it is not possible to summarize the data into a concise form.

Leak Tests were performed on the D.G. O'Brien penetrations using helium leak-detection procedures. The tests were performed at a temperature of 270°F with a helium differential pressure of 52 psig. Maximum allowable leak rate was E-6 cc per second. Leak tests were performed at least twice during the sequence of tests on prototype models and once on each production unit. The leak rates measured were all less than the allowable leak rate (E-6 cc/sec).

Two Conax electrical penetration assemblies were installed to provide additional instrumentation and low voltage power circuits required for the RVLIS and Appendix R modifications. The Conax penetration assemblies, designed and tested in accordance with IEEE Standard 317-1976 as modified by Reg. Guide 1.63-1978; will withstand, without damage or interruption of operations, the forces resulting from an earthquake, in addition to the normal and accident design requirements. The Conax penetration assemblies are designed for field installation by welding to the inside end of a spare nozzle type penetration passing through the reactor containment vessel wall. Each Conax penetration assembly is provided with a single connection to allow pressure testing for leaks. The Conax low voltage instrumentation assembly is installed in Group C. The Conax low voltage power and control assembly is installed in Group F.

Piping Penetrations

All penetrations listed in Table 5.2-2, except the vacuum breakers, penetrate the Shield Building as well as the Reactor Containment Vessel. Both the Reactor Containment Vessel and Shield Building are provided with capped spare penetrations for possible future requirements.

General Design Description

All process lines traverse the boundary between the inside of the Reactor Containment Vessel and the outside of the Shield Building by means of piping penetration assemblies made up of several elements. Two general types of piping penetration assemblies are provided; i.e., those that are not required to accommodate thermal movement (designated as cold penetrations in Figure 5.2-4) and those which accommodate thermal movement (hot penetrations depicted in Figures 5.2-5 and 5.2-5a). All piping penetration assemblies are listed in Table 5.2-2.

Both hot and cold piping penetration assemblies consist of a containment penetration nozzle, a process pipe, a Shield Building penetration sleeve and a Shield Building flexible seal. In the case of a cold penetration, the Containment Vessel penetration nozzle is an integral part of the process pipe. For hot penetrations, a multiple-flued head becomes an integral part of the process pipe, and is used to attach a guard pipe and an expansion joint bellows. The expansion joint bellows is welded to the Reactor Containment Vessel penetration nozzle. The flued head fitting is the only part of the penetration assembly, which comes into contact with the Shield Building at any time.

At the termination of a piping penetration assembly near the Shield Building, a low-pressure leakage barrier, is provided in the form of a Shield Building flexible seal as shown in Figures 5.2-4, 5.2-5 and 5.2-7. These devices provide a flexible membrane type closure between the Shield Building penetration sleeve, which is embedded in the Shield Building, and the process pipe. In the case of hot penetrations 8 and 11, a circular plate is being used rather than a flexible seal as shown in Figure 5.2-5a. This plate will serve as both an anchor and a Shield Building seal.

Design Basis Common to Hot and Cold Piping Penetrations

All Containment Vessel penetration nozzles are designed to meet the requirements for Class B vessels under Section III of ASME Boiler and Pressure Vessel Code. In compliance with the code, the operating stresses in a containment vessel penetration nozzle caused by the attached penetration assembly are limited to the allowable values given in the code. For earthquake analysis, Section III of the ASME code permits the use of 1.5 times the allowable stress value for the material being used.

The double-bellows expansion joints in the hot-pipe penetration assemblies and the Shield Building flexible seals for all pipes are designed to accommodate the maximum combination of vertical, radial, and horizontal differential movements of the Reactor Containment Vessel, the Shield Building, and the piping. The design considers the calculated displacements resulting from earthquake, pressure, and temperature, as presented in Figure 5.2-6, and also accounts for the actual measured displacement of representative penetration nozzles made during the initial pressure testing of the Containment Vessel.

Design Evaluation Common to Hot and Cold Piping Penetrations

Seismic loads on Containment Vessel penetration nozzles were determined by performing a dynamic model analysis of the piping systems. Response spectra at the piping system anchor points were used in this analysis. These response spectra were developed from the results of a dynamic time-history seismic analysis of the plant structures. Differential movement between points in the various structures have been included in the analysis of the piping.

The validity of the computer program used to perform the piping system seismic analysis was proven by comparison with an independent analysis of selected systems performed by recognized consultants.

Loads on Containment Vessel penetration nozzles due to thermal expansion of the pipe, thermal and pressure movements of the Reactor Containment Vessel, and piping system weight were determined by a flexibility analysis of the piping system. This analysis was performed with the aid of a computer program using established methods documented in current technical literature. The piping configuration and supports, restraints, or anchors on each side of the penetrations were designed to limit the stresses in the Containment Vessel at the penetration nozzle to the criteria defined in Appendix B.

Design Description - Hot Penetrations

A hot piping penetration assembly is used when the differential between the normal operating temperature of the fluid carried by a process line and the Reactor Containment Vessel wall temperature would create unacceptable thermal or cyclic stress at the attachment of the vessel penetration nozzle.

In addition to the elements contained in a cold piping penetrations assembly, as shown in Figure 5.2-4, a hot assembly has a multiple-flued head, a guard pipe, an expansion bellows and an impingement ring. The multiple-flued head is machined from a solid forging. It is welded into, and becomes an integral part of, the process line. The inner flue provides support for the guard pipe and the outer flue provides support for the expansion joint bellows. The guard pipe is located concentric to the process pipe, and is cantilever-supported by a weld attachment to the inner flue of the flued head. The length of the guard pipe is set so that it extends past the Reactor Containment Vessel penetration nozzle into the vessel.

The only interaction between the hot penetration assemblies and the Shield Building is through the multiple-flued head. This interaction takes place in one of two ways and is described as follows:

1. For the main steam, feedwater and residual heat removal penetrations, the multiple-flued head passes through a sleeve in the Shield Building as shown in Figures 5.2-5 and 5.2-7. The sleeve acts as a horizontal and vertical guide, which allows rotational and axial movements. The piping system and hence the flued head is allowed to rotate or move axially within the Shield Building sleeve but is restrained by the sleeve from moving in any direction perpendicular to the axis of the process line for all seismic, temperature, weight and jet loads. There are no pressure loads that have any effect on the flued head - Shield Building interaction for these assemblies other than the vertical and transverse movements of the containment vessel due to internal vessel pressure. The loads due to this movement are small, being a function of the transverse spring constant of the penetration expansion bellows, and have been considered in the design of the process line, the multiple flued head and the Shield Building sleeve.

2. For the steam generator blowdown and letdown line penetrations the multiple-flued head is anchored to a sleeve in the Shield Building as shown in Figure 5.2-5A. All movement of the flued head and consequently the process line is restrained at this point. The design of this anchor and the process line considered all loads due to seismic, weight, temperature, pressure and pipe rupture jet effects of both the piping system and the structures.

The expansion joint bellows is attached at one end to the outer flue of the flued head and at the other end to the Reactor Containment Vessel nozzle. The expansion joint is provided with a double-layered bellows that has a connection between bellows for integrity testing. An impingement ring is mounted on the guard pipe to protect the expansion joint bellows from jet forces that might result from a pipe rupture inside containment.

The multiple-flued head with its associated guard pipe and expansion joint bellows provides a leak-tight seal for the extension of the containment boundary where the hot penetration assembly traverses the Shield Building annulus.

Design Basis - Hot Penetrations

Hot piping penetration assemblies are provided to:

a. Prevent unacceptable thermal and cyclic stress on Reactor Containment Vessel penetration nozzles,

b. Accommodate thermal movement, and

c. Protect containment from the effects of a hot process pipe rupture in the annulus between the Shield Building and the Reactor Containment Vessel.

Where hot penetration assemblies traverse the Shield Building annulus, they are designed to provide considerable margin between code allowable stress values and maximum calculated stresses in the pipe. This was accomplished by using 1.5 times the system design pressure to calculate the pipe wall thickness for the process and guard pipe, using the formula and allowable stresses given in USASI-B 31.1.0-1967. Under the normal B31.1.0 code practice, the system design pressure alone is adequate for calculating the pipe wall thickness. The same procedure was used to set the thickness of the guard pipe and the multiple-flued head.

Design Evaluation - Hot Penetrations

Stresses resulting from the combination of loads defined in Appendix B, Paragraph 7.2 were calculated for a typical process pipe in a hot piping penetration assembly using the cross sectional area of the pipe wall thickness as required to meet 1.5 times system design pressure. Comparison of calculated stress values with code allowable stresses shows:

- Thermal stresses are less than 50% of allowable;
- Combined longitudinal stresses are less than 50% of allowable;

♦ Hoop stresses are less than 60% of allowable.

The design criterion applied to the hot penetration guard pipes is defined in Table B.7-3 of APPENDIX B and is the same as that applied to all Class I piping. The allowable stress values and stress analysis results for normal, upset and faulted loading conditions for the main steam and feedwater guard pipes are shown in Table 5.2.12. The values listed represent the peak values that will occur at any given location in the guard pipe. This table shows that calculated guard pipe stress levels are well within those allowed by the criteria given in Table B.7-3 for all loading conditions. The main steam and feedwater penetrations were selected for this study because their failure would have the most severe consequences in the Shield Building annulus. A review of the design of the other hot penetration guard pipes indicates that calculated stress levels for these guard pipes would be of the same order of magnitude as for the main steam and feedwater guard pipes and well within allowable values.

Main Steam Piping Penetrations

The main steam piping penetration assembly, shown in Figure 5.2-7, uses the same elements as a hot piping penetration assembly. In addition, the main steam piping is anchored to the interior concrete of the Reactor Containment Vessel.

A limit stop designed to control lateral movement but permit axial movement is provided around the main steam piping inside containment. This limit stop serves to limit pipe movement in the event of a longitudinal pipe break, thus serving to control pipe whip inside containment. The multiple-flued head is also designed to transfer lateral loads that could result in the event of a main steam piping rupture exterior to the Shield Building, to a specially designed structural arrangement in the Shield Building. A lateral axial limit stop similar to the one provided inside containment is also provided exterior to the Shield Building.

Design Basis - Main Steam Piping Penetrations

The main steam piping between the anchor inside containment and the first isolation valve outside of the containment has a wall thickness selected by using 1.5 times the system pressure and normal code allowable stress values. The main steam piping anchor inside containment is designed to sustain the full force resulting from a 360° circumferential break of the main steam piping. The other requirements previously discussed for a hot piping penetration assembly are also met.

Equipment and Personnel Access

The equipment hatch and air locks are supported entirely by the Reactor Containment Vessel and will not be connected either directly or indirectly to any other structure.

The equipment hatch and air lock was fabricated from welded steel and furnished with a double-gasketed flange and bolted dished door. Provision is made to pressure-test the space between the double gaskets of its flange.

Two personnel air locks are provided. Each personnel air lock is a double-door welded steel assembly. Quick-acting type equalizing valves are provided to equalize pressure in the air lock when entering or leaving the Reactor Containment Vessel. Provision is made to pressure test the air locks for periodic leak-rate tests.

The two doors in each personnel air lock are interlocked to prevent both from being opened simultaneously and to ensure that one door is completely closed before the other door can be opened. Remote indicating lights and annunciators in the control room indicate the door operational status. Provision is made to permit by-passing the door interlocking system with a special tool, to allow doors to be left open during plant cold shutdown. Each door lock hinge can be adjusted to assist proper seating. A lighting and communication system, which can be operated from an external emergency supply is provided within each air lock.

Fuel Transfer Penetration

The fuel transfer penetration provided is for fuel movement between the reactor refueling cavity in the Reactor Containment Vessel and the spent fuel pool. This penetration consists of a 20-inch stainless steel pipe installed inside a 24-inch pipe. The inner pipe acts as the transfer tube and is fitted with a testable double-gasketed blind flange in the reactor refueling cavity, which provides containment integrity. A standard, normally closed gate valve in the spent fuel pool canal is also provided to isolate the refueling cavity from the spent fuel pool. This arrangement prevents leakage through the transfer tube in the event of an accident. The outer pipe is welded to the Reactor Containment Vessel. Bellows expansion joints are provided between the two pipes to compensate for any differential movement.

Containment Supply and Exhaust Purge Duct Penetrations

The ventilation system purge duct and make-up duct penetrations are welded directly to the penetration nozzles in a manner similar to the cold piping penetration. The ducts are circular in section and designed to withstand the Reactor Containment Vessel maximum internal pressure. They are provided with isolation valves as described in Section 5.3.

Missile Protection Features

High-pressure Reactor Coolant System equipment, which could be the source of missiles is suitably protected either by the concrete wall enclosing the reactor coolant loops or by the concrete operating floor to block any passage of missiles to the containment walls. The steam drum, which forms an integral part of the steam generator, represents a mass of steel which provides protection from missiles originating in the section of the containment within the shield wall and below the operating floor. A concrete structure is provided over the RCCA drive mechanisms to block any missiles generated from a hypothetical fracture of a RCCA housing.

The missile shield structures inside Containment were analyzed using the conservation of momentum method. The penetration depth was calculated by the Ballistic Research Laboratories Formula for steel and concrete.

Missile protection is designed to the following criteria:

a. The Reactor Containment Vessel is protected from loss of function due to damage by such missiles as might be generated in a loss-of-coolant accident for break sizes up to and including the double-ended severance of a main coolant pipe (DBA);

b. The Engineered Safety Features Systems and components required to maintain containment integrity are protected against loss of function due to damage by missiles.

During the detailed plant design, the missile protection concept necessary to meet these criteria was developed. These concepts are:

a. Components of the Reactor Coolant System are examined to identify and to classify missiles according to size, shape and kinetic energy for purposes of analyzing their effects.

b. Missile velocities are calculated considering both fluid and mechanical driving forces, which could act during missile generation.

c. The structural design of the missile shielding takes into account both static and impact loads.

d. The Reactor Coolant System is surrounded by reinforced concrete and steel structures designed to withstand the forces associated with double-ended rupture of a main coolant pipe, and designed to stop these missiles.

The types of missiles for which protection is provided are:

- Valve Stems
- Valve bonnets
- Instrument thimbles
- Various types and sizes of nuts and bolts
- Complete RCCA drive mechanisms, or parts thereof
- Piece of pipe
- Pressurizer valves, instrumentation thimbles and heaters

Removable slabs, blocks and partitions were evaluated to determine whether or not they could become possible missiles. These items were reviewed to assure that the functions of safety related structures and equipment will not be affected during an OBE or tornado.

Removable slabs or plates, which were found to be potential missiles are anchored to Class I structures by means of bolts and anchor bolts. Concrete inserts and bolts provided for removing the slabs, are designed with lifting capacities equal to two times their normal design stress capabilities.

Concrete block walls and partitions which are adjacent to Class I (seismic) equipment or form the boundary of zone "SV" are designed under Class I seismic requirements, reinforced and anchored to Class I structures. Recommended construction practices of the Uniform Building Code were used as a minimum requirement.

The Reactor Building crane is also regarded as a potential earthquake-produced missile. The Reactor Building crane was analyzed for seismic forces under maximum expected crane load conditions during refueling. The procedure for seismic analysis is treated as described herein before in this Section. The design provides for no loss of crane support or structural integrity during a DBE, and that no parts will shake loose.

To assure the stability of the crane during an earthquake, the bridge and trolley are equipped with fitted rail yokes that will allow free-rolling movement but will prevent the wheels from being lifted or derailed. The structural steel and Reactor Containment Vessel wall supporting the crane are designed to withstand these earthquake-induced forces.

The bridge and trolley wheels are equipped with electrically activated, spring set brakes. Upon loss of power or when the crane or trolley is not under operator control, the springs activate the brakes locking the wheels firmly in place to prevent rolling. The positive wheel stops provided and designed to prevent overtravel of the trolley also prevent the trolley from leaving the rails even in the unlikely event of brake failure. For details about missile protection of the Reactor Containment Vessel and Shielding Building, refer to Appendix B, Paragraph B.9.

Insulation

Insulation is not required for the Reactor Containment Vessel or the Shield Building.

Shielding

The primary performance objective of radiation shielding within the Reactor Containment Vessel is to minimize the exposure of plant personnel to radiation emanating from the reactor and auxiliary systems. The radiation levels prevalent during plant operation, as well as those experienced upon shutdown, are recognized in the determination of the shielding requirements as described in Section 11.2. The secondary performance objective of the radiation shielding is to minimize radiation effects upon operating equipment as described in Section 11.2.

5.2.2 SHIELD BUILDING DESIGN

Design Conditions

The Shield Building is a reinforced concrete structure of vertical cylindrical configuration with a shallow dome roof. An annular space is provided between the Reactor Containment Vessel shell and the wall of the Shield Building to permit construction operations and periodic visual inspection of the Reactor Containment Vessel. The volume contained within this annulus is approximately 374,000 cubic feet.

The Shield Building concrete wall is 2'-6" thick and the dome is 2'-0" thick for biological shielding requirements. The design bases for shielding requirements for operational radiation protection are discussed in Section 11.2. The results of analysis with respect to assumed post-accident conditions using these design parameters are discussed in Section 14.3.5.

The normal ambient temperature in the annular space is set by heat loss through the Reactor Containment Vessel and Shield Building. The design assures that the Reactor Containment Vessel metal temperature can be maintained above 30°F.

Following the Design Basis Accident (DBA), displacement of the containment shell due to the post-LOCA pressure/temperature increase inside the containment and heat transferred to the air in the annular space could cause a slight pressure rise. This annulus pressure transient is limited to less than 4.0" H_2O by venting the annular space. Conservative assumptions for temperature transmission to the space, and pressure drop in the Shield Building Ventilation (SBV) system were used in sizing the ventilation system. Following this initial pressure transient, the Shield Building is maintained at a slight negative pressure with respect to the Auxiliary Building-between ½" and 2½" H_2O. The Shield Building seals are designed to accommodate these pressures.

The structure was analyzed to assure adequate strength to accommodate thermal stresses resulting from the above temperature-induced thermal gradients.

The following loadings are considered in the design of the Shield Building:

- Structure dead load
- DBA load
- Live loads
- Wind load
- Tornado load
- Uplift due to buoyant forces
- Earthquake loads
- External missiles

Design Leakage Rate

The Shield Building is designed so that its inleakage rate is not greater than the amounts indicated in Figure 5.2-8. The inleakage rates shown in Figure 5.2-8 are total leak rates including the leakage through the personnel doors. The originally calculated contribution to total leak rate from the various sources of inleakage, with a differential pressure of ¼" of water is shown in Table 5.2-3. These leak rates will in most cases vary linearly with pressure, and extrapolations made on this basis are shown in Figure 5.2-8.

Figure 5.2-8 also shows the measured as-built leakage rate and a modified design basis leakage rate. For discussion of these leakage rates see Section 5.5.1.

Design Loadings

Dead Load

Dead load consists of the dead weight of the Shield Building, the Reactor Containment Vessel, grout under the Containment Vessel, the foundation slab, and the weight of concrete, structural steel, equipment, and miscellaneous building items within the Reactor Containment Vessel.

The basis for calculation of weights is given in Section 5.2.1.

Design Basis Accident (DBA) Load

The DBA load is determined by analysis of the pressure and temperature transients in the annulus during a Design Basis Accident (See Section 14.3.4).

Live Load

Live load consists of snow loads on the dome applied uniformly to the top surface of the dome at 40 pounds per horizontal square foot (PSF).

Wind Load

Wind loading for the Shield Building is based on Figure 1(b) ASCE Paper 3269, "Wind Forces on Structures," using the fastest wind speed for a 100-year period of recurrence. This results in a 100-mph basic wind at 30 feet above grade. In addition, this paper was used to determine shape factors, gust factors and variation of wind velocity with height.

Tornado Load

The structure was analyzed for tornado loading (not coincident with accident or earthquake) on the following basis:

a. The Shield Building is designed for an internal pressure of 3 psi acting outward and uniformly over the entire inside surface of the building, and

b. Lateral force on the Shield Building assumed as the force caused by a "funnel" of wind having a peripheral tangential velocity of 300 mph and a forward progression of 60 mph.

The design tornado that is assumed corresponds to a large funnel of arbitrary radius with a band of maximum velocity wind (300 mph) which is at least 150 feet wide and which extends from the ground surface to at least 200 feet.

The forces induced in the Shield Building by the external pressure of the above hypothetical tornado are considered in two cases. In the first case the structure is considered as an annular cantilever beam loaded with maximum wind pressure over the entire horizontally projected area of the structure. This case gives the maximum overturning moments, and membrane, shear and axial stresses. In the second case, it is assumed that the exterior pressure at any horizontal section varies from a maximum at the sub-wind centerline of the building to zero at each end of the diameter, which is perpendicular to this centerline. This pressure pattern is assumed to be the same at all elevations. This case gives the maximum ring moments and shears.

Uplift Due to Buoyant Forces

Uplift forces, which are created by the displacement of ground water by the structure, are accounted for in the design of the structure.

Earthquake Loads

Seismic loads were computed using the following:

- Operational Basis Earthquake seismic ground acceleration of 0.06g. (See Appendix A).
- Design Basis Earthquake seismic ground acceleration of 0.12g.

A vertical component of β of the horizontal ground acceleration is applied simultaneously with the horizontal acceleration. The plots of the seismic response spectra are shown in Appendix A. The classification of plant structures and equipment and the applicable damping factor are shown in Appendix B. The general procedure for seismic analysis is treated in Section 5.2.1.

External Missiles

The design tornado missile was chosen to be equivalent to an airborne 4" x 12" x 12' - 0" long wood plank traveling 300 mph. Other possible sources of tornado missiles were evaluated, as reported in Appendix B, but none were as destructive as the wood plank. The turbine missile is discussed in Appendix B.

Codes

Design Codes

Concrete structures are designed in accordance with the ACI Code 318-63 "Building Code Requirements for Reinforced Concrete". Structural steel is designed in accordance with the "AISC Specification for Design, Fabrication and Erection of Structural Steel for Buildings".

Welding is in accordance with the ASME Boiler and Pressure Vessel Code and AWS "Standard Code for Arc and Gas Welding in Building Construction".

In addition to the above, the Shield Building is designed and constructed in accordance with applicable state and local building code requirements.

Material

Specifications and working drawings for materials and their installation are of such scope and detail that the desired integrity of the Shield Building was assured.

Basic specifications for these materials include the following:

a. Concrete

All concrete work is in accordance with ACI 318-63 "Building Code Requirements for Reinforced Concrete".

The concrete was tested to assure a minimum compressive strength of 4000 psi at 28 days. Where the heat of hydration was not a factor, Portland cement Type I, conforming to Specification ASTM C150, was used. In other applications, Type II cement was used for its lower heat of hydration and slower rate of heat generation.

Testing of aggregates, cement, concrete mixes and sampling were undertaken by a qualified independent testing laboratory. Concrete samples were taken at a point nearest to placement as was possible.

Standards and specification for concrete materials, testing and construction methods are as follows:

- ACI 306 Recommended Practice for Winter Concreting
- ACI 347 Recommended Practice for Concrete Formwork
- ACI 605 Recommended Practice for Hot Weather Concreting
- ACI 614 Recommended Practice for Measuring, Mixing and Placing Concrete
- ASTM C31 Making and Curing Concrete Compression and Flexure Test Specimens in the Field
- ASTM C33 Specifications for Concrete Aggregates
- ASTM C39 Test for Compressive Strength of Molded Concrete Cylinders

- ASTM C87 Test for Effect of Organic Impurities in Fine Aggregate on Strength of Mortar
- ASTM C98 Specification for Ready-Mixed Concrete
- ASTM C143 Test for Slump of Concrete
- ASTM C150 Specification for Portland Cement
- ASTM C172 Method of Sampling Fresh Concrete
- ASTM C175 Specification for Air-Entraining Portland Cement
- ASTM C227 Test for Potential Alkali Reactivity of Cement Aggregate Combinations
- ASTM C231 Test for Air Content of Freshly Mixed Concrete by the Pressure Method
- ASTM C260 Specification for Air-Entraining Admixtures for Concrete
- ASTM C289 Test for Potential Reactivity of Aggregates
- ASTM C295 Petrographic Examination
- ASTM C350 Fly Ash for use as a Concrete Admixture

b. Reinforcement

All reinforcing is new billet steel and specified as follows:

Type & ASTM Spec No.	Grade Designation	Minimum Tensile-PSI	Minimum Yield-PSI	Minimum Elongation in 8" Spec
A-15	Intermediate	70,000	40,000	7-12%*
A-408	Intermediate	70,000	40,000	10%
A-432	--	90,000	60,000	7%
ASTM A305	Specification for Deformations of Deformed Steel bars for Concrete Reinforcement			
AWS D12.1	Recommended Practices for Welding Reinforcing Steel, Metal Inserts and Connections in Reinforced Concrete Construction			

* Varies with size of bar.

All reinforcing steel was shipped to the job site in bundles bearing a tag identifying its size, grade and code number keyed to heat numbers. This information was verified by certified mill test reports, which accompany each shipment of reinforcing steel.

All reinforcing steel bars were clearly identified with markings legibly rolled into the surface of each bar at the producer's plant showing the point of origin, size, designation and type of steel. High-strength steel bars were further identified with the minimum yield strength rolled into the surface of each bar.

All reinforcing steel was tested in accordance with ASTM Specifications. Tests included one tension and bend test per heat number or mill shipment; whichever is less, for each full size diameter bar.

The Quality Control Engineer followed construction to assure that steel bars as specified on the drawings were placed in their proper location and were of the designated size and strength.

The fabricator was required to furnish assurance that the steel as detailed, fabricated and shipped to the job site, bearing identification tags, was the same as that received from the mill.

Reinforcing steel bars were generally lap-spliced except where the design indicated that welded splices were structurally required, because the lap splice could not adequately meet the joint requirements within practical limits, or where welded splices were economically advantageous. Welded splices were made by full penetration fusion butt-welding. Only A-15 reinforcing bars were used.

Whenever the integrity of structure was dependent upon welding, the weld splicing of A-15 bars was as follows:

a. A fully pre-qualified, written welding procedure was used

b. All welders were pre-qualified by tests

c. The chemistry of the bars to be welded was determined, and no bar with a carbon content greater than 0.35% was welded;

d. Welded splices in adjacent bars were staggered at least 24 inches

e. An extra ring of bars was added and was in addition to those required by the design; and

f. Not less than one in each twenty-five welds were tested radiographically. Each weld was inspected visually.

Shield Building wall reinforcement in each direction was governed by the structural requirements of tornado pressure loadings, wall temperature gradients, and missile loading. The structural requirements were in excess of the minimum percentages of reinforcement required by ACI-318. The design indicates that the minimum percentage of reinforcement in any case is not less than 0.4%.

Design Basis

The Shield Building completely encloses the Reactor Containment Vessel, the access openings, the equipment door, and that portion of all penetrations that are associated with Primary Containment. The design of the Shield Building provides for:

- Biological shielding,
- Controlled releases of the annulus atmosphere under accident conditions,
- Environmental protection of the Reactor Containment Vessel.

The Shield Building is primarily a shielding structure and as such it is not subjected to the internal pressure loads of a pressure-containment vessel. The structure therefore will not be subject to bi-axial tension and cracking due to pressure loads.

Since the Shield Building need not be designed for internal pressures, the reinforcement arrangements are based primarily on the need to withstand the more conventional structural loads from environmental effects.

The design criteria for the openings are:

a. To provide reinforcement around the openings to carry all loads by frame action. Because the Shield Building wall thickness is set to meet radiation shielding requirements, the thickness is generally in excess of that necessary for structural requirements; therefore, it was necessary to add additional bars around the perimeter of the opening to provide a reinforced concrete frame.

b. To provide for horizontal and vertical shearing forces acting in the plane of the opening, diagonal bars are provided forming an octagonal pattern of reinforcement around the perimeter of the opening.

The reinforced concrete and structural steel of the Shield Building and foundation were designed by the working stress method and are based on allowable stresses as set forth in Table 5.2-4.

Earthquake stresses were added linearly and directly to stresses caused by the Design Basis Accident, snow, wind, dead loads, and the appropriate operating loads, to obtain the total stresses. These total stresses are within the maximum stress limits allowed by the applicable design criteria.

The summation of loads as stated in Conditions 1 and 2 of Table 5.2-4 provided the design basis for the Shield Building.

The summation of loads as stated in Condition 3 of Table 5.2-4 provided the design basis to maintain the integrity of the Shield Building so that a proper shutdown can be made during the ground motion having twice the intensity of the Operational Basis Earthquake. The design was reviewed to assure that resulting deflections or distortions did not prevent the proper functioning of the structure or piece of equipment and would not endanger adjacent structures or components.

Adequate reinforcing was placed in the concrete walls and dome to control cracking due to concrete shrinkage and temperature gradients.

The summation of loads as stated in Condition 4 of Table 5.2-4, provided the basis for a design review to assure that the Shield Building will suffer no loss of shielding or containment function due to a 300-mph design tornado.

The allowable stress values shown in Table 5.2-4 assure an elastic behavior of steel reinforcement during a Design Basis Earthquake, thus minimizing the cracking of concrete and the impairment of leaktight integrity.

It is demonstrated by the stress criteria of Table 5.2-5 that during the maximum conditions of earthquake loading, the allowable stress values of Table 5.2-4 provide adequate margins within the elastic range of the material to assure the elastic behavior of the structure.

Concrete Cracks in Walls of Shield Building During Post-Earthquake Conditions

It is expected that only negligible amounts of inleakage could occur through cracks during post-earthquake conditions and accordingly, leakage due to an earthquake has been indicated as "negligible" in Table 5.2-3.

This is primarily a consequence of the combination of low shear stresses and the predominant effect of the dead load of the structure on the moment and shear stresses produced by earthquakes.

The maximum expected tangential shear stress on a horizontal cross-section of the Shield Building wall at the base of the structure during a Design Basis Earthquake will be in the order of 90 psi. This shear stress will vary from zero to a maximum and will be distributed over the cross-section as outlined in Section 5.2.1.

The ratio of the structure deadload to the Design Basis Earthquake moment uplift forces is in the order of 1:0.95 with a deadload stress of about 220-psi and a maximum moment uplift stress of 210-psi at the outer fibers of the cross-section where the shear stresses are zero.

The structure dead load acts as a pre-stress on the structural system and modifies the trajectories of tensile and compressive principal stresses from their characteristic curved lines to near straight orthogonal lines. For example, where the horizontal tangential shear stress is zero, the principal stress trajectory will be vertical and there will be no normal horizontal stress and associated cracking of concrete. This condition occurs in the walls at the extremities of a diameter normal to the neutral axis. A second limiting condition occurs where the horizontal tangential stress is a maximum (90-psi) at the extremities of a diameter coincident with the neutral axis. The vertical dead load stress (220-psi) will combine with the shear stress to generate a principal stress trajectory that will be oriented near vertical (approximately 20° to the vertical) with a normal principal tensile stress of 30-psi.

It is evident that earthquake conditions cannot produce sufficient horizontal forces to initiate and open vertical shear cracks to any appreciable degree. It is further evident that whatever cracks might form would only open and be exposed to inleakage for an average time length of only one-half of the duration of the earthquake. Since the elastic action of the structural system will alternately open and close stress cracks during earthquake stress cycling and finally remain closed after the earthquake has subsided. In addition, the stresses due to an earthquake will vary from a maximum at the base of the structure to zero at the very top, thereby further minimizing any leakage.

Finally, it can be concluded that earthquake forces will not provide any stresses of appreciable magnitudes that might contribute to any significant cracking and associated inleakage to the Shield Building Ventilation System.

Concrete Cracks and Leakage at the Springline of the Shield Building

The construction of the Shield Building roof consisted of two stages. First, a thin reinforced concrete dome roof was placed (about 5" thick). This dome was supported by the Reactor Containment Vessel shell dome by means of temporary construction shores. When this concrete dome had sufficient strength, the temporary shores were removed and the balance of roof concrete was placed.

The design and construction of the dome roof and walls at the springline are monolithic. The dome roof and walls are reinforced in a meridional direction for structural discontinuity moments and shears. The dome roof and walls are also reinforced in a circumferential direction to resist the hoop tensile stresses caused by springing of the dome roof at its point of support. The hoop reinforcing is proportioned in accordance with the discontinuity strains of the concrete shell near the springline to achieve an efficient use of the steel for crack control. The concrete dome roof is under continual compression due to dead load; therefore, cracking in the dome will be negligible.

It is estimated that the thrust of the dome roof will result in a maximum tensile stress of 120-psi in the hoop concrete section at the springline.

This stress is well within the limiting stress at which cracking in concrete can be expected to occur. However, in keeping with a conservative approach, the concrete is assumed to crack. The resulting inleakage through these cracks is shown in Table 5.2-3.

The wall and dome surfaces are provided with a minimum of 2" concrete cover over the reinforcing bars as required by ACI-318, to provide an adequate protection of reinforcement against freezing and thawing.

Foundation Slab and Support and Adjacent Building Construction

This foundation slab of the Containment System will have structural continuity with the foundation slab of the adjacent building.

The design provides for the relative static settlements that could occur between structures as dead load is placed on the slab during construction. The design assumes a zero relative settlement between structures during an earthquake. The design takes into account local foundation deformations during an earthquake due to the dynamic action of a rocking structure.

Structures adjacent and exterior to the Shield Building walls are designed with provision for the movements of the Shield Building during an earthquake. Walls originating in the Auxiliary Building that abut the Shield Building wall are isolated from the Shield Building by an adequate physical separation to prevent damage to either structure due to hammering during an earthquake.

Floors in the building adjacent to the Shield Building wall are not supported by the Shield Building wall but are cantilevered from columns. These floor slabs and beams are separated from the Shield Building wall by a flexible expansion joint that will permit independent building movements that might occur during an earthquake. Adequate physical separation between these floor slab terminal edges and the Shield Building walls are provided to prevent hammering.

The Shield Building and Auxiliary Building were dynamically analyzed for seismicity by John A. Blume & Associates (References 3 and 4).

The results of these two analyses were correlated to determine the separation required between the two structures to assure that there would be no physical contact between the structures under earthquake conditions.

The separation space in walls and floors, which are a part of the leak barrier, is sealed by means of a flexible membrane attached to the concrete surfaces at the separation joint. This seal is of a type that will provide for movements that expand or contract the separation joint and for movements in the two other coordinate axes.

The Shield Building is monolithic with and integrally connected to the Auxiliary Building up to and including the mezzanine floor.

Flexible expansion joints have been provided between the Shield Building and adjacent Auxiliary Building structures above the mezzanine floor to provide for the relative lateral movements of the buildings during an earthquake.

A polyvinylchloride foam plastic joint filler is used to form the joint and a two-component polysulfide base sealant provides a watertight seal at the top of the joint. At the periphery of the roof, the "SV" zone and the Shield Building airlock concrete enclosures, an additional flexible non-plasticized chlorinated polyethyline reinforced with polyester mesh flashing was provided to assure continued leaktight integrity.

Prior to placement of the foundation slabs and exterior walls below grade, a 40-mil thick polyvinylchloride sheet was placed to act as a vapor barrier. This sheet acts in a dual role, preventing out-leakage of radioactive fluids from the plant, and at the same time preventing infiltration of ground water. The waterproofing membrane extends up to an elevation one foot below grade, which is approximately fifteen feet above the subsurface drainage system circumscribing the plant and twenty-two feet above high water lake level.

Analysis and Design for Missiles

The Shield Building is designed to withstand the tornado missile without loss of function. It is also designed to intercept the turbine missile and prevent it from damaging the Reactor Containment Vessel. Details of design analysis methods and criteria are given in Appendix B.

Design Analysis

The Shield Building was analyzed for individual cases of dead load, wind load, tornado, earthquake, and internal pressure loads.

The basis of the wind analysis is identified herein before in this Section.

The analysis for tornado winds considered the combined frontal pressure effects of both the 300 mph tangential and the 60 mph forward velocities acting on one-half of the Shield Building.

The seismic analysis of the Shield Building is given in Section 5.2.1.

The resulting stresses from individual loadings are combined as indicated in Table 5.2-4. Under loading condition "4" in Tables 5.2-4 and 5.9-1, the maximum allowable concrete stress is indicated to be 0.85 f'_c.

In the actual design of the structures the maximum concrete stresses for condition "4" in the Shield Building never exceeded 0.53 f'_c and the actual stresses in the internal structures (Table 5.9-1) never exceeded 0.75 f'_c.

Therefore, with a maximum allowable concrete stress of 0.75 f'_c there is no lack of consistency in the safety margins for concrete and reinforcement steel. In the final design, a balanced design of concrete and steel reinforcement has been achieved.

The Shield Building is subjected to two distinct types of shear action. These are:

a. A radial shear normal to the building walls developed by discontinuity conditions and tornado internal pressure, and

b. A tangential and accompanying longitudinal shear taken on a radial plane through the walls induced by the flexural action of the structure under the effects of wind or earthquake loads (Reference 9).

The Shield Building was analyzed for effects of thermal stress.

The containment atmosphere during normal operation varies depending on the time of year. The average temperature at any one time can be as high as 115°F.

The steady-state temperature distribution in the Shield Building was determined, using the conservative assumption that the containment vessel and the annulus air are both at 120°F and the outside temperature is at -35°F. The calculated inside and outside wall temperatures are 99°F and -7°F, respectively.

The transient temperature distributions following a LOCA were calculated by superimposing on the steady-state temperature profile the effects of radiative and convective heat transfer to the wall from the annulus air and the containment vessel due to the energy release inside the containment vessel. The maximum wall temperature is 117°F on the inside surface at 45.2 minutes following a LOCA and temperature changes at a depth beyond 10 inches are then negligible. The physical properties as well as the assumptions used to calculate the steady and transient temperature profiles are listed in Table 5.2-11.

The total concrete and steel stresses, including thermal stress due to the transient condition are also presented in the table, and these are below the allowable limits.

The calculation described is conservative in most respects, rather than realistic. The LOCA conditions are those associated with a double-ended break and calculations of outer containment wall temperatures following smaller breaks indicates very little difference in heat input to the annulus for these cases. The dominant thermal effect is the gradient resulting from initial temperatures on opposite sides of the wall and these temperatures have been chosen conservatively.

The Shield Building is a structure of conventional reinforced concrete design and construction. The Shield Building is designed to resist the internal pressure of 3-psi due to tornado loading, however this does not subject the structure to significant membrane forces that are ordinarily associated with pressure vessels. This low internal pressure minimizes the cracking of concrete and assures that concrete will function as a shear-carrying member. In addition, the structural discontinuity stresses are small since no significant pressure forces are present. Therefore, radial shears will be very low.

The shear provisions of the ACI Code are applicable to concrete members under the combined action of moments and applied axial compressive or tensile loads. These provisions are applicable to the transverse and tangential shear actions noted above.

The tangential and longitudinal shear distribution over the annular cross-section and at any point will be given by:

$$V_c = \frac{V}{hrt} \cos q$$

The tangential and longitudinal shear will vary from zero over a thickness of wall located at the extremes of a diameter normal to the neutral axis to a maximum on a wall thickness located on both extremes of a diameter coincident with the neutral axis, and is given by:

$$V_{max} = \frac{V}{hrt}$$

V = Total shear on the annular cross-section;
v_c = Unit tangential or longitudinal shear on the annual cross-section;
r = Mean radius of Shield Building wall;
t = Thickness of Shield Building wall;
θ = Polar angle measured on either side of the neutral axis locating the point at which the unit shear is to be determined;
v_{max} = Maximum tangential and longitudinal shear.

The maximum unit shear is limited in the design by the allowable values of the ACI 318-63 Code, Chapter 12.

Construction

The reinforced concrete vertical cylindrical wall of the Shield Building was constructed using the slip-form method as described in Chapter 5 of ACI Code 347-68.

Several minor changes were made to the fixed-form designs to facilitate slip forming. The changes effected the following:

a. Length of forms for block-outs, plate inserts and embedded items were shortened for easy placement in forms,

b. Spacing of form-supporting yokes required shorter lengths of reinforcement,

c. Concrete slump increased to 6 inches and maximum aggregate size was limited to ¾", and

d. Form to be tapered to avoid adhesion as form is raised.

Penetrations

The Shield Building penetrations for piping ducts and electrical cable are designed to withstand the normal environmental conditions which may prevail during plant operation and also to retain their integrity during and following postulated accidents.

The openings into the Shield Building, including personnel access openings, equipment access openings and penetrations for piping, duct, and electrical cable, are designed to provide containment which is as effective as the Shield Building and consistent with the Shield Building leak rate.

The Shield Building is provided with two access openings, one located adjacent to the maintenance air lock and the other adjacent to the personnel air lock.

Each access opening is provided with double-interlocked doors. A bolted, sealed door is provided at the equipment opening.

Pipe penetrations through the Shield Building are sealed with low-pressure flexible closures. The seals are of a rubber-impregnated canvas material or equal and seal the process line to the embedded sleeve in the Shield Building.

Flexibility of all cables is provided between the Shield Building and the Reactor Containment Vessel so that no damage can occur to the cables or structures due to differential movements between the two structures.

All electrical cables in the annulus are provided with support systems of various methods.

Generally the cables will be supported by tiers of cable tray, which are supported by structural members bearing on the external support concrete and tied back to the Shield Building. The cables will not be clamped to the cable tray system and will have slack allowed in them at both ends.

The large 5000-volt cables used for the reactor coolant pump are supported somewhat differently. A supporting framework is clamped to the penetration nozzle (part of the containment vessel) to provide a rigid support for the cables at the outboard end of the porcelain bushings. The cables will have approximately 18" of length available for slack prior to entering the embedded conduit in the Shield Building.

Annulus lighting system cables are all supported by clamps fastened only to the Shield Building.

All of the above systems are designed to accommodate differential movements between the two buildings without interaction.

Further discussion of penetrations is presented in Section 5.2.1.

8.2 ELECTRICAL SYSTEM

8.2.1 NETWORK INTERCONNECTIONS

Electrical energy generated at 20-kV is transformed to 345-kV by the Main Transformer, a bank of three single-phase units, which in total are rated at 580-mVA. It is delivered through a 345-kV, 25,000-mVA (interrupting rating), 1600-ampere (552-mVA) circuit breaker to the 345/138-kV switching station located at the plant site, as shown in Figure 8.2-1. The electrical output is integrated into the American Transmission Company's 345-kV and 138-kV transmission systems. These systems are interconnected at the plant site substation by a transformer rated at 300 mVA. The 345-kV transmission system has interconnections with Northern States Power Company. Two 345-kV transmission lines are connected to the plant switching station and are on separate line structures in order to minimize the possibility of losing more than one circuit at a time. Either line is capable of carrying the full output of the generator. In addition, two 138-kV lines are connected to the plant switching station from the 138-kV grid system. These two lines together are capable of carrying the full output of the generator, but would be limited by the 300-mVA Autotransformer on a loss of both 345-kV lines.

An analysis of the integrated 345/138-kV power system has been made and shows that a fault on any one of the transmission lines, any bus section at the Kewaunee Substation, or the loss of the Kewaunee generator will not cause a cascading failure on the transmission system, thereby insuring an off-site power supply to the plant for any of the aforementioned failures. (See Stability Study in Reference 6).

The centerline of the two 138-kV lines is 265 feet south of the southern-most two 345-kV lines on a separate right-of-way as they leave the plant property. About one-half mile west of the substation the two 345-kV lines turn: one, line R-304, goes north and the other, line Q-303, turns south at a dead-end tower. Line Q-303 crosses over both 138-kV lines, one before the dead-end tower and one after (see Figure 8.2-1a). Since dead-end tower failure is not considered a credible accident, failure of a 345-kV line structure at some point could only cause a failure of one 138-kV line. There is no area where a failure of a 138-kV line structure can cause a failure of either 345-kV line. Thus, there is no single failure that can disable more than two lines. This results in three pairs of physically independent sources of offsite power (see NRC SER in Reference 1).

The 138-kV overhead transmission line from the substation into the plant on-site distribution system through the Reserve (Startup) Auxiliary Transformer is bifurcated at the substation to connect to either the East or West 138-kV buses. Each leg of the bifurcation is separated from the respective bus with a 138-kV oil circuit breaker. The controls for the 138-kV breakers are separated into two distinct 125-V d-c branch circuits; one serving the breaker closing circuit and one of the two trip circuits with the primary relaying; the other serving the second trip coil and the backup relaying. These breakers are manually controlled from the plant Electrical Vertical Panel A.

Similar connections to the East and West 138-kV buses are made to the 345-138-kV autotransformer, which can be used to energize the autotransformer if the need arises. These breakers are manually controlled from the plant Electrical Vertical Panel A, or local control in the substation control house and by operator choice can be transferred to system operating office supervisory control.

The West 138-kV bus is connected to a capacitor bank with a 138-kV oil circuit breaker (see Figure 8.2-1). Each of the four-capacitor banks have switching devices. The breaker and switching devices are normally under system operating office supervisory control and by operator choice, can be transferred to control from Electrical Vertical Panel A.

The 345-kV system can be and normally will be used to energize the 345-138-kV Autotransformer. The 345-KV oil circuit breakers are controlled similar to the East and West 138-kV bus oil circuit breakers described above.

The tertiary winding of the autotransformer is used to furnish power to the 13.8-kV Tertiary Auxiliary Transformer via an underground insulated power cable. This cable becomes the second of the two physically independent circuits to provide off-site power to the on-site distribution systems.

Both of the above circuits will normally be energized at all times and will be connected to one or the other of the engineered safeguards buses at all times. Thus, loss of the reactor, turbine generator, Main Station Auxiliary source of power does not even require a transfer for the safeguards buses. In the case of an engineered safeguards bus energized by one of the on-site power sources, i.e., the diesel-generator, and subsequent loss of the diesel-generator, the automatic transfer system will search for a transformer source and automatically close in its breaker. The above postulated condition could only occur after a loss of all off-site power and subsequent restoration of at least one source, prior to manual operator actions per emergency instructions.

Thus, loss of power from the nuclear unit should not affect the availability of power from the two off-site transmission circuits or the two standby power sources. Loss of either transmission circuit should not affect the other transmission circuit, the two standby power sources or the nuclear unit as a source. Finally, loss of one or both of the standby power sources should not affect the availability of the transmission sources or the nuclear unit.

Substation D-C System

The substation d-c system consists of two separate systems; the 48-V d-c distribution system furnishing power to the solid-state relay systems and the 125-V d-c distribution system furnishing control power and additional electro-mechanical and microprocessor based relay systems.

The 48-V d-c distribution system consists of redundant battery chargers, one battery and two distribution cabinets.

The 125-V d-c distribution system consists of redundant battery chargers, one battery and four distribution cabinets.

The battery chargers are furnished with a-c and d-c (output) failure relays and redundant a-c sources. High and low voltage alarms are provided for the d-c circuit.

The 125-V d-c distribution system is arranged so that the two circuits used to control each high-voltage circuit breaker emanate from different distribution cabinets. Each branch circuit in the three breaker distribution cabinets is individually monitored and alarmed on loss of d-c voltage. These alarms are displayed in the substation control house and are transmitted to the plant control room annunciator as a substation alarm. The branch circuits in the fuse distribution cabinet are not individually monitored and alarmed on loss of d-c voltage. A loss of power to any of the microprocessor based relays fed from the fuse cabinet would be indicated locally on the relay and transmit an alarm to the transmission system operator. The overall effect is that of a dual supply to the high-voltage breaker control trip elements with alarm should any portion of the supply system become abnormal.

The oil circuit breakers can close on 90-V d-c and trip on 70-V d-c. Each battery charger is equipped with a low voltage alarm set at 122-V d-c. The battery is considered to be fully discharged when the voltage reaches 105-V d-c. The actual closing energy is supplied from air compressors also integral with each circuit breaker. The air compressors are a-c powered. The storage cylinders have sufficient capacity when fully charged for five operations if a-c were lost. These breakers can be manually tripped at the breaker.

Loss of the substation battery and a concurrent fault on one of the transmission lines wherein the plant would continue to supply power to the fault is a set of postulated conditions, which would lead to an undetected failure. Under these postulated conditions, the fault would clear when the transmission-line remote terminal breakers opened. The Kewaunee breakers can be manually tripped and, thus, isolate the fault, thereby restoring off-site power to the plant via the remaining transmission lines.

Thus, given an undetected failure, the capability to clear and restore off-site power to the Kewaunee plant will not be lost, and the restoration of off-site power (assuming the grid is available) will be made within the time period (seven days) in which the plant can be maintained in a safe condition without off-site a-c power.

8.2.2 PLANT DISTRIBUTION SYSTEM

The Auxiliary Electrical System is designed to provide a simple arrangement of buses requiring the minimum of switching to restore power to a bus in the event that the normal supply to that bus is lost.

Single Line Diagrams

The basic components of the plant electrical system are shown on the Single Line or Circuit Diagrams, Figures 8.2-2 and 8.2-3. These figures show the 20-kV, 4160-V, 480-V and instrument bus a-c systems and the 125-V and 250-V d-c systems. In addition, Figures 8.2-1, 8.2-2 and 8.2-3 show the basic elements of the 13.8-kV, 138-kV and 345-kV substation systems.

Main Auxiliary, Reserve Auxiliary and Tertiary Auxiliary Transformers

The plant turbine-generator serves as the primary source of auxiliary electrical power during "on-the-line" operation. Power is supplied via a 20-4.16-kV, three-winding, Main Auxiliary Transformer, which is connected to the main leads from the turbine generator.

The primary sources of electrical power for the auxiliaries associated with engineered safety features during "on-the-line" operation of the plant are the Reserve Auxiliary Transformer and the Tertiary Auxiliary Transformer. Power is normally supplied to one bus (Bus 1-6) through the 138-4.16-kV, three-winding Reserve Auxiliary Transformer which is connected to the 138-kV portion of the Kewaunee Substation. Power is normally supplied to the second bus (Bus 1-5) through the 13.8-4.16-kV, two-winding Tertiary Auxiliary Transformer which is connected, by an underground line, to the 13.8-kV tertiary winding of the 345/138/13.8-kV auto transformer in the Kewaunee Substation. Either source can supply both buses.

Auxiliary power required during plant startup, shutdown and after reactor trip is supplied from the 13.8/138/345-kV Kewaunee Substation via the Reserve Auxiliary and Tertiary Auxiliary transformers. After turbine-generator trip, the auxiliaries on the 4160-V buses being fed by the Main Auxiliary Transformer are transferred by a fast bus transfer scheme using stored energy breakers to the Reserve Auxiliary Transformer. Control power for the plant auxiliary breakers is supplied by the plant batteries. The high-side (substation) breakers use the substation battery for control power.

4160-V System

The 4160-V system is divided into six buses, as shown in Figure 8.2-2. Buses 1-1 and 1-2 are connected via bus main breakers to the Main Auxiliary and Reserve Auxiliary Transformers. These buses supply power to the Reactor Coolant Pumps and the Feedwater Pumps.

Buses 1-3 and 1-4 are also connected via bus main breakers to the Main Auxiliary and Reserve Auxiliary Transformers. These buses supply power to the normal balance-of-plant auxiliaries, and each bus supplies power to three 4160 - 480-V station service transformers. A fourth transformer connected to bus 1-4 supplies power to the Technical Support Center. In addition, the Circulating Water Pumps, Condensate Pumps and the Heater Drain Pumps are directly connected to buses 1-3 and 1-4.

Buses 1-5 and 1-6 are connected via bus main breakers to the Main Auxiliary, Reserve Auxiliary, and Tertiary Auxiliary Transformers. In addition, each bus is directly fed via a main breaker by a diesel generator. The two buses are tied together via two bus tiebreakers in series, one on each bus. Each bus supplies two of the four 4160 - 480-V station service transformers for the plant's 480-V engineered safety features equipment. In addition, the Service Water Pumps, Auxiliary Feedwater Pumps, Safety Injection Pumps and the Residual Heat Removal Pumps are directly connected to buses 1-5 and 1-6.

Bus 1-5 is normally supplied from the Tertiary Auxiliary Transformer and Bus 1-6 is normally supplied from the Reserve Auxiliary Transformer. Thus, no transfer is required for the engineered safety features in the event of an incident.

The bus tie breakers between Bus 1-5 and Bus 1-6 can only be manually closed, but are interlocked so that the diesel generators cannot be operated in parallel.

480-V System

The 480-V system is divided into 11 load center or switchgear buses, as shown in Figure 8.2-2. Those fed from 4160-V buses 1-3 and 1-4 serve balance-of-plant loads, those fed from 4160-V buses 1-5 and 1-6 serve the loads associated with the engineered safety features equipment.

Transformers 1-32 and 1-42 are connected to 4160-V buses 1-3 and 1-4, respectively. Transformer 1-32 feeds 480-V bus 1-32; transformer 1-42 feeds 480-V bus 1-42. These components including the 480-V bus tie are assembled as a conventional, double-ended switchgear unit. In a similar manner buses 1-33/1-43 and 1-35/1-45 are connected to 4160-V buses 1-3 and 1-4. Bus 1-46, supplying the TSC, is connected to 4160-V bus 1-4.

The various motor control centers throughout the plant are then connected to these switchgear buses.

The power required for the 480-V engineered safety features and other vital plant loads is supplied from four 480-V buses fed from 4160-V buses 1-5 and 1-6. Transformer 1-51 is fed from 4160-V bus 1-5 through breaker 1-505 and supplies bus 1-51. This transformer, bus and breakers, including one bus tie, are assembled as a switchgear unit. In a similar manner, bus 1-52 is also connected via breaker 1-505 to 4160-V bus 1-5.

The large 480-V engineered safety features motors are connected to bus 1-51. Motor Control Centers supplying the smaller loads are fed from bus 1-52.

A redundant 480-V system is supplied by 4160-V bus 1-6 through 4160-V breaker 1-607.

MCC 1-5262 may be fed from either 480-V switchgear bus 1-52 or 480-V switchgear bus 1-62 through breakers 15209 or 16209 respectively, via a manually operated transfer switch. 480-V switchgear bus 1-52 may ultimately be fed from on-site power source Diesel Generator 1A and

480-V switchgear bus 1-62 may ultimately be fed from on-site power source Diesel Generator 1B. 480-V MCC 1-5262 does function as a swing bus between the two redundant on-site power distribution systems as discussed in Safety Guide 6. However, the mechanical operation of this switch allows MCC 1-5262 to be connected to only one 480-V switchgear bus at a time. Hence the redundant 480-V switchgear buses can never be paralleled through the operation of this switch. Therefore, the transfer of MCC 1-5262 does conform to the criteria outlined in Safety Guide 6. Furthermore, the components fed from MCC 1-5262 are not part of a redundant load group. The components are:

- Turbine Turning Gear
- Turbine HP Hydrogen Seal Oil Backup Pump
- Condensate Bypass All Heaters to Feedwater Pumps Motor-Operated Valve
- Waste Gas Compressor 1B
- Station and Instrument Air Compressor 1A

The intent of the transfer switch is to provide maintenance flexibility for these loads.

It should also be noted that 480-V switchgear buses are each protected from the transfer switch with a breaker.

The automatic transfer switch associated with MCC 1-52E provides BRA-106 with two possible sources of power, MCC 1-52E or MCC 1-52C. However, both MCC 1-52E and MCC 1-52C are associated with the same on-site power source (Diesel Generator 1A). Therefore, the transfer is not between power systems of redundant load groups and does not fall into the category of swing buses discussed in Safety Guide 6.

Likewise, the automatic transfer switch associated with 1-62E allows BRB-106 to be fed from either MCC 1-62E or MCC 1-62C. These MCCs are both within the same load group.

125-V and 250-V D-C System

The 125V and 250-V DC system is divided into five buses (two safeguard and three non-safeguard, see Figure 8.2-3) each with one battery and a battery charger, distribution panels and inverters. Components prefixed with BRA and BRB make up the safeguard DC system and those prefixed with BRC, BRD and BRE make up the non-safeguard system.

The DC power requirements of the engineered safety features (ESF) and other vital plant loads are supplied by the safeguard batteries (numbered BRA-101 and BRB-101). Each safeguard battery consists of 59 cells, each of which are the lead calcium type. The batteries are rated 125V DC, 1304 ampere-hours at the eight-hour rate without discharging below 1.78V per cell. Two main DC distribution panels (BRA-102 and BRB-102) are fed from these batteries via main fuses. The main distribution panels connect the battery to the battery charger, to the sub-distribution panels, and allow for the interconnection of the two buses through the bus tiebreakers.

BRA-102 supplies sub-distribution panel BRA-104. BRA-104 in turn supplies the control and excitation power for diesel generator 1A, the control power for ESF buses 1-5, 1-51, and 1-52, control and power to one-half of the redundant essential plant equipment required for safe shutdown in the event of loss of AC power and provides a standby power source for the safeguard inverters BRA-111 and BRA-112.

BRB-102 supplies sub-distribution panel BRB-104. BRB-104, in turn, supplies the control and excitation power for diesel generator 1B, the control power for ESF buses 1-6, 1-61, and 1-62, control and power to one-half of the redundant essential plant equipment required for safe shutdown in the event of loss of AC power and provides a standby power source for the safeguard inverters BRB-111 and BRB-112.

The balance of plant DC power requirements are supplied by three non-safeguard batteries (designated BRC-101, BRD-101 and BRE-101). Batteries BRC-101 and BRD-101 each consist of 59 cells and are of the lead calcium type, rated at 125V DC, 1680 ampere-hours at the eight hour rate to reach 1.78V per cell. Battery BRE-101 consists of 120 cells of the lead calcium type, rated at 250 V d-c, 694 ampere-hours at the two hour rate to reach 1.75V per cell. Each battery is connected to a main distribution panel (BRC-102, BRD-102 and BRE-102). The main distribution panel connects each battery to a battery charger, sub-distribution panel, bus tie (125V d-c batteries only) and inverter(s).

Distribution panels BRC-102 and BRD-102 supply sub-distribution panels BRC-103 and BRD-103, respectively. Panel BRC-102 is also a standby source for inverter BRC-109. The BRC and D-103 panels, in turn, supply Technical Support Center diesel generator control and excitation power, control power for non-ESF buses 1-1, 1-2, 1-3, 1-4, 1-32, 1-33, 1-35, 1-42, 1-43, 1-45, and 1-46, other non-safety related equipment sensitive to a loss of AC power and are a standby source for inverter BRD-109 and a proprietary inverter.

Distribution panel BRE-102 supplies AC drive units BRE-109 and BRE-110, which convert the 250V d-c input power to 230-V a-c output power for the turbine emergency oil pump and air side seal oil backup pump motors.

Each of the five battery buses is served by one connected battery charger. Each safeguard battery has provisions for connection of a spare portable charger. The spare safeguard charger can be moved to its designated mounting in either safeguard battery room and connected to the DC bus in the event of charger failure. The battery life to minimum voltage under maximum load will allow sufficient time to make this connection. The non-safeguard spare charger is permanently mounted between the normal 125V d-c chargers and can be connected to either bus as required. There is no spare 250V d-c charger.

The two bus tie breakers between 125-V d-c distribution cabinet BRA-102 and 125-V d-c distribution cabinet BRB-102, are manually operated. These breakers are strictly administratively controlled to prevent them from being closed during plant operation.

Instrument Bus

The 120-V a-c instrument supply is split into several buses as shown on the one line diagram Figure 8.2-3. There are four independent instrument buses, each fed by an inverter which, in turn, is fed from each of the d-c buses. A fifth independent non-interruptible bus, fed by an ESF motor control center, supplies the rod position indicators. The sixth and seventh independent buses, each fed by an inverter, supply the plant process control computer and cabinet BRD-115, respectively. There are two additional independent buses; each fed from an ESF motor control center through a transformer.

Evaluation of Layout and Load Distribution

The physical location of electrical distribution system equipment is such as to minimize vulnerability of vital circuits to physical damage as a result of accidents.

The Main Auxiliary, Reserve Auxiliary and Tertiary Auxiliary Transformers are located outdoors and are physically separated from one another by firewalls. Each transformer cell, formed by the firewalls, has an automatic water spray system to extinguish and prevent the spread of fires.

The 4160-V switchgear and 480-V load centers are located in areas which minimize their exposure to mechanical, fire and water damage. This equipment is coordinated electrically to permit safe operation of the equipment under normal and short-circuit conditions.

The 480-V motor control centers are located in the areas of electrical load concentration. Those associated with the turbine-generator auxiliary system in general are located in the Turbine Building. Those associated with the nuclear steam system are located in the Auxiliary Building.

The application and routing of control, instrumentation and power cables are such as to minimize their vulnerability to damage from any source. The construction design drawings had second level review in accordance with the Kewaunee Construction Quality Assurance Program.

All cables are specified using conservative margins with respect to their current-carrying capacities, insulation properties and mechanical construction. The power conductors are three conductor, galvanized armored and installed in a single layer in ladder type cable trays, and clamped to insure that ample ventilation spacing is maintained throughout the run.

Bulk control power supply cables are treated as noted in the previous paragraph.

Control cables normally employ minimum size of #12 AWG when run in multi conductor cables in control trays. As there are few continuous loads on these circuits no attempt was made at derating. Continuously loaded circuits (current transformer secondaries) are sized by burden requirements of the circuit. In special cases #14 AWG multi-conductor control cable was allowed in cable trays. In these identified cases, a safety evaluation has been performed. Special cases may continue to allow #14 AWG multi-conductor cable to be used. These will require a safety evaluation and approval by the responsible engineer.

Cables that are run in trays have fire resistant jackets. Safety-related cables meet the environmental qualifications required by 10 CFR 50.49. Appropriate instrumentation cables are shielded as required to minimize induced-voltage interference. Wire cables related to engineered safety features and reactor protective systems are routed and installed to maintain the integrity of their respective redundant channels and protect them from physical damage.

Supports and cable trays for safety feature power cable systems are designed for 100% loading plus the forces generated by a seismic disturbance. Other cable systems are designed for 100% loading. The ladder fill is restricted to one layer, clamped in place to maintain 2 to 12 inch spacing between cables.

Cable trays for control and signal cable support systems and other safety-related systems are designed for forces generated by a seismic disturbance assuming maximum fill. Other cable systems are seismically designed for maximum fill. The tray fill is restricted to 50% of the tray's cross sectional area for safety-related cable and 60% for non-safety related cable.

Separation Criteria

Cable separation provides sufficient isolation between redundant systems so that no single failure or electrical incident can render both redundant systems inoperable or remove them from service.

To assure complete separation of Class IE circuitry that initiates and controls the transfer of power sources to the emergency a-c and emergency d-c distribution system, the following cable and cable tray separation techniques are used:

a. Each tray section of the cable tray system has an identifying code indicated on the electrical design drawings and this same identification is stenciled on the tray after it is installed. Stenciling is applied at each straight section of tray where the identifying code changes.

   On the electrical design drawings, the trays are identified by a number placed in a rectangular symbol. The trays used for the redundant safeguards equipment are further identified with a vertical line adjacent to the tray name symbol.

   The identifying code contains the designation "S5" or 'S6" where applicable to identify the safeguard train.

b. The two Safeguard tray systems are independent of each other such that they are physically separated a minimum of 3 feet horizontally and 3 feet vertically, except in the Relay Room where 1-foot minimum horizontal and vertical separation is required.

c. Each electrical cable has an identifying code indicated on the electrical design drawings and cable routing lists. This same cable number is affixed at each end of the cable with permanent tags.

The identifying code is based on the following systems.

The cable codes for the 4160-V switchgear; the 480-V switchgear and the 480-V motor control centers are a combination of letters and numbers that form a four part coding containing:

- Unit Number 1
- Source of Power
- Power or Control
- A serially assigned number which provides uniqueness.

A "5" in the second digit from the left is a cable in the "Safeguards 5" system; a "6" a cable in the "Safeguards 6" system (Example 1S5 xxx or 1S6 xxx).

Normal cable being fed from safeguards distribution equipment should have cable codes starting with 1NP followed by a serially assigned number which provides uniqueness. Some normal cables remain with their original safeguards power source type cable codes, since no comprehensive re-labeling program was implemented.

Redundant circuitry for reactor protection and engineered safety systems are separated into groups as follows.

Class IE cables are divided into the following groups:

- Group 1 (color code red)
  Red Instrument Channel
  Instrument Bus I

- Group 2 (color code yellow)
  Yellow Instrument Channel
  Instrument Bus IV

- Group 3 (color code green)
  Safeguard Train A, Battery 1A, Diesel Generator 1A, 4160-V Bus 1-5, 480-V Bus 1-51, 480-V Bus 1-52 and its associated Motor Control Centers.

- Group 4 (color code blue)
  Blue Instrument Channel
  Instrument Bus III

- Group 5 (color code white)
  White Instrument Channel
  Instrument Bus II

- Group 6 (color code orange)
  Safeguard Train B, Battery 1B, Diesel Generator 1B, 4160-V Bus 1-6, 480-V Bus 1-61, 480-V Bus 1-62 and its associated Motor Control Centers.

Each group is run in a separate tray, ladder, trough or conduit. These trays are identified on electrical drawings for engineered safety features, and are marked and color-coded on the actual hardware. All trays for the engineered safety feature equipment are Class I structures.

Within containment the tray systems for the four reactor protection instrument channels are separated 3 feet horizontally where they involve 2/4 logic and are separated approximately 20 feet where they involve 2/3 logic. Vertical separation is 5 feet where practical, and where impractical, barriers are installed. These barriers are solid metal covers on the lower trays.

For the non-Class-IE systems throughout the remainder of the plant, trays installed in stacks are spaced vertically with a minimum of 12 inches bottom to bottom in all areas. However, Class IE trays have a minimum of 15 inches bottom to bottom between trays of the same train. Class IE trays containing instrument, control or power cables have a minimum horizontal separation between redundant circuits of 36 inches. Redundant circuits are not permitted in the same tray or conduit. If closer spacing than 36 inches cannot be avoided, an approved barrier must be placed between the circuits. Cable trays are routed to avoid a fire hazard area, such as oil storage rooms, oil tanks, etc., whenever possible. When this cannot be done, the cable tray system is protected by fire resisting barriers. Where practical, these barriers will be tray covers. Whenever possible, a wall or floor has been introducted between trays carrying redundant safeguard circuits. Barriers are required where mutually redundant trays cross. The barriers shall extend to each side of the protected tray by a distance equal to approximately three times the wider of the two trays.

Mixing of power cables with control or instrument cable in the same tray is not permitted throughout the plant. Whenever a control and/or instrument cable tray and a power tray are in the same stack, the power tray is located in the top tier.

Trays for Train A and Train B are separated 3 feet horizontally and vertically except in the Relay Room where practical design considerations require 1-foot vertical and horizontal separation. The two trains are separated by 40 feet at the reactor containment vessel penetrations.

Power cables for engineered safeguards are kept strictly in cable trays so designated. Occasionally, a non-safety-related power cable may be run in a safeguards cable tray but a safeguards cable will never run in any tray other than its own system. Control cables are similarly separated and control and instrumentation of the same train designation may be run in the same control cable tray. Non-safety-related power, control or instrumentation cable shall not be permitted to cross over from one safeguards tray to another.

Where the wiring for redundant engineered safety features is within a single panel or panel section, this wiring is separated one group from the other, by a 6-inch air space or a fireproof barrier. The barriers are sheet metal or flexible metallic conduit. The flexible conduit may be applied to one train to separate it from the other train. Wiring not associated with either train may be grouped with one train but may not cross from one train bundle to the other train.

Where the approved logic required recognition of input signals from both A train and B train devices into common terminal blocks or operational devices, the interconnecting wiring can no longer retain train identity. Train A and B wiring shall maintain physical and electrical separation up to the termination point prior to where the interconnecting wiring loses its identity. This is an allowable exception to the above paragraph. The interconnecting wiring common to both trains shall not be termed "Normal", nor shall it be routed with normal wiring.

Cable trays used for redundant reactor protection systems, engineered safeguards systems and Class IE electrical systems have an identifying code number stenciled on them in color paint after they are installed. The number is applied whenever there is a change in identity or when passing through floor or wall openings. This number is applied prior to the pulling of any cables, and the color establishes the system to which it is assigned.

During the cable pulling operation, an intermittent colored stripe is applied to the cable as it leaves the reel. This color must match the color of the tray system in which it is installed. Normal (or non-Class IE) cables, if pulled into a colored tray system, will not have an identifying stripe on them.

The following colors are used for identification:

- Green - Safeguard Train A circuits
- Orange - Safeguard Train B circuits
- Red - Reactor Protection system
- White - Reactor Protection system
- Blue - Reactor Protection system
- Yellow - Reactor Protection system

Relay Room

The main (original) relay room is arranged in two groups of four rows of cabinets. The group to the east contains blue and white reactor protection channels, reactor protection train A, engineered safeguard train A, and other miscellaneous relay circuits (including two cabinets on the east wall). The group to the west contains yellow and red reactor protection channels, reactor protection train B, engineered safeguard B, and miscellaneous relay and metering circuits.

A relay room expansion to the south has two rows of cabinets with provision for a third. The cabinets contain plant process computer input/output and miscellaneous monitoring equipment.

The upper levels of the relay room are used for cable routing, as there is no separate room labeled "cable routing room".

Within the room the trays are arranged in four tiers. The arrangement of these tiers is such that instrument and control circuits of reactor protection and engineered safety features of like trains are stacked together or with normal instrumentation and control. The colored instrument channels for reactor protection are converted from trays to rigid conduit systems where they enter the relay room and then to 6-inch metal raceway as they pass over the instrument racks to facilitate rack output interconnections.

Horizontal fire barriers are provided between the Control Room and the Relay Room at the control consoles and panels.

Testing

Testing of the operator-activated Class IE circuitry that initiates and controls the connection of the buses to the power sources for the a-c emergency power system; the Main Auxiliary Transformer, the Tertiary Auxiliary Transformer, the Reserve Auxiliary Transformer, and the Diesel Generators; can be done by transferring the buses, one at a time, from one source to another with controls available to the operator in the Control Room. Manual switching of these source breakers occurs during periodic breaker maintenance, bus maintenance and diesel generator testing. Testing of the automatic-initiated Class IE circuitry that initiates and controls the connection of the buses to these sources can be performed by utilizing a switch in the Control Room which disconnects two bus undervoltage relays of one safeguard bus, thereby simulating loss of voltage, and activates the associated logic circuitry that is required to automatically restore power to the bus.

A sequential events recorder prints out the operation of each relay in the scheme providing printed proof of the circuitry's proper response. Upon successful completion of this test a green light glows to the right of the test switch.

Both the automatic circuitry and the manual circuitry can be tested by these methods during plant operation. Portions of the circuitry that cannot be conveniently tested with the plant in operation without temporarily interrupting circuit protection are the bus lockouts and the transformer lockouts. A switch is provided for each lockout, which isolates its output contacts allowing the lockout to operate without actually tripping any breakers. The continuity of transformer lockout relays is continually monitored by indicating lamps located in the Control Room.

The power sources for the d-c emergency power system associated with the Train A load group are Station Battery 1A and 480-V MCC 1-52C (via Battery Charger 1A). The power sources for the d-c emergency power system associated with the Train B load group are Station Battery 1B and 480-V MCC 1-62C (via battery charger 1B). The transfer of a d-c bus between its respective battery and battery charger can be tested by opening the 480-V breaker supplying power to the charger, thus simulating a loss of power to the 480-V bus. The d-c system should transfer to the battery as its source of power. If, after opening the 480-V breaker to the charger the d-c bus retains its voltage, the transfer was successful.

8.2.3 EMERGENCY POWER

Sources Description

Power sources for the engineered safety features are 4160-V Bus 1-5 and Bus 1-6. The normal source of power to Bus 1-5 is the Tertiary Auxiliary Transformer. The Reserve Auxiliary and Main Auxiliary Transformers provide backup sources, in that order. The normal source of power to Bus 1-6 is the Reserve Auxiliary Transformer. The Tertiary Auxiliary and Main Auxiliary Transformers provide backup sources, in that order. Thus, since the normal source of power for these buses is the 138/345-kV Kewaunee Substation, no transfer is required in the event of a turbine-generator trip.

If all other power sources should fail, two diesel generators are provided, one connected to 4160-V Bus 1-5 and one connected to 4160-V Bus 1-6. Each of these is a General Motors Corporation, Electro-Motive Division, Model A-20-C1, diesel engine-generator unit rated at 2600-kW, (2860-kW, 110% Overload, two thousand hours per year) 0.8 pf, 900 rpm, 4160-V, 3 phase, 60 Hertz. The generator has emergency ratings of 2950-kW for seven days continuous and 3050-kW for thirty minutes per year.

Each diesel generator, as a backup to the normal standby a-c power supply, is capable of sequentially starting and supplying the power requirements of one complete set of engineered safety features equipment. The electrical emergency power system logic diagrams are shown in Figures 8.2-4, 8.2-5, and 8.2-6. The units are located in separate rooms in Class I portion of the Administration Building. These rooms are heated; assuring that the diesel generators can be started in cold weather.

Service water for the Diesel Engine Cooling Water Heat Exchanger is supplied from separate service water headers for Diesel Generator 1A and 1B. The Cooling Water Heat Exchanger is an engine mounted water-to-water heat exchanger providing cooling for the engine jacket water and for the engine oil heat exchanger. Vent fans for each room provide a supply of combustion air into the Diesel Room. Separate startup air receivers and compressors are located just external to the rooms. Primary and reserve tanks of the air receivers supply compressed air to the dual Air Start System, the DG cooling water isolation valve actuators, and the Diesel Room Ventilation and combustion air dampers.

Each diesel generator is automatically started by either one of two pairs of air motors mounted on each side of the diesel (four air motors per engine). Each unit has its own independent starting system including a bank of four air storage tanks, two primary and reserve tanks, and one compressor powered from the 480-V emergency bus. An air cooler/dryer is installed on the discharge of each air start system compressor. The dry air improves the starting performance of the diesel engine. The primary or reserve tanks have sufficient storage to crank the engine for twenty seconds. The generator is capable of being started and ready to accept load in ten seconds.

Starting air is admitted from the storage tanks to the starting system through a pressure-reducing valve to supply air to the starters.

The following describes a typical diesel engine start sequence. The sequence stated assumes the air start motor priority selector switch is in the #1 position. The air start motor priority selector switch is typically rotated from set #1 to set #2 on a monthly basis. This ensures even run time on the air start motors.

When the diesel start signal is initiated, a start attempt is made through air start motor set #1. If the air start motor set #1 fails to engage within 2 seconds, a second start attempt is made with the same set of motors. If the air start motors still do not engage, and then after 5 seconds a third start attempt is made, this time using the second pair of air start motors (set #2). Air start motors set #2 will continue to attempt to start the diesel generator on a two second cycle, until the engine starts or 15 seconds after the start signal, whichever occurs first. The start signal also initiates starting of the fuel priming pump and the governor booster pump. If, after fifteen seconds, the diesel has not reached 200 rpm, a start failure signal opens the fault relay. Starting air is cut off, the fuel priming and governor booster pump are stopped. Operator action is then required for further start attempts. The fault relay in the diesel generator room must be reset and any faults causing the fault lockout must be corrected before the start signal will be effective again.

The start failure relay serves to indicate an abnormally long period of engine cranking without an engine start (fifteen seconds) and to prevent subsequent engine starting attempts until the cause of the engine start failure has been determined by operating personnel. The total air capacity available to crank the engine is twenty seconds per air starter-tank combination.

The following interlocks must be satisfied to automatically start the diesel engine:

a. Engine mounted LOCAL/REMOTE (AUTO/OFF/MAN) switch must be in REMOTE (AUTO) position for Diesel Generator 1A(1B). Control Room and local annunciation is given when this local switch is not in the REMOTE (AUTO) position.

b. Control Room PULLOUT/STOP/AUTO/START switch must be in AUTO position (maintained position). (The other maintained position of this switch is the PULLOUT position, which disables the engine starting circuit during engine maintenance).

c. 125-V d-c control power must be available at the diesel engine control panel. (The engine starters cannot be engaged if control power is not available; loss of control power is annunciated in the Control Room).

d. The engine must not be running.

e. Air pressure must be available to the starting air system (loss of air pressure on each starting air system is annunciated locally and in the Control Room).

The diesel engine interlocks itemized above can be periodically tested as follows:

a. Turning the engine mounted LOCAL/REMOTE (AUTO/OFF/MAN) switch from the REMOTE (AUTO) position will alarm the local and Control Room annunciators for Diesel Generator 1A(1B). Diesel generator operational testing verifies annunciator status; therefore, any failure to start due to mispositioning of the switch would identify the annunciator failure.

b. The stable PULLOUT position of the Control Room PULLOUT/STOP/AUTO/START switch is visibly different than the stable AUTO position in that the switch handle is slanted to the left of its normally vertical position (AUTO) and a distinctive silver colored switch shaft extension is visible.

c. Opening the 125-V d-c distribution breaker to the engine control cabinet alarms the Control Room annunciator.

d. Response of the air receiver pressure switches can be tested and calibrated by valving in the standby air receivers, valving out the on-line receivers, opening the air compressor circuit breaker, and opening the receiver drain valve until an alarm occurs on the local and Control Room annunciators.

The motor-driven compressor associated with each diesel is fed from the emergency bus supplied from the same diesel. The control voltage for each diesel starting system is from its associated 125-V d-c station battery.

An audible and visual alarm system is located in the control room and will alarm off-normal conditions of jacket water temperature, lube oil temperature, fuel oil level, starting air pressure and Diesel Generator stator hi temperature (1 of 12 inputs feeding the 4160 Volt Stator Temperature Hot annunciator). An alarm also sounds if a starting circuit is locked out, a control switch is not in "auto" position, or d-c power for the controls at the diesel generator is lost. The alarm in the control room also alerts the operator to other various off-normal conditions including jacket water expansion tank level and pressure, engine crankcase pressure, and fuel oil pressure. Local audio and visual alarms are also provided at each diesel generator.

Reference 2 is a safety evaluation in which the NRC has concluded that, based on the review of submitted information and on-site inspections, the status annunciators for the diesel generators are acceptable. The review was specifically intended to ensure that any deliberately induced condition which may disable the diesel generators, and which is expected to occur more frequently than once per year, is automatically annunciated in the Control Room with devices worded to alert the operator of their abnormal status.

Two 850-gallon "day" tanks are located in enclosures within each diesel generator room. The two tanks provide capacity for approximately four hours operation for one generator at full load. Two 35,000-gallon underground storage tanks supply fuel oil through immersion pumps to either pair of day tanks. Combined fuel capability of one storage tank and two day tanks would provide a minimum of 7 days fuel supply for one diesel generator (36,000 gallons of fuel oil), thus assuring adequate time to restore off-site power or to replenish fuel. The diesel fuel oil storage capacity requirements are consistent with those specified in ANSI N195-1976/ANS-59.51, Sections 5.2, 5.4 and 6.1. See Reference 3 and Technical Specification 3.7 for fuel oil storage requirements.

Loading Description

Each diesel generator is automatically started on the occurrence of either of the following incidents:

a. Undervoltage on the associated 4160-V bus (Bus 1-5 or Bus 1-6) provided that the low voltage is not caused by a fault which operates the bus lockout relay (see Reference 5 and Technical Specification);

b. Initiation of a Safety Injection Signal which will start both diesel generators.

With the occurrence of undervoltage on 4160-V Bus 1-5, whose normal source of power is the Tertiary Auxiliary Transformer, the automatic sequence is as follows:

1) Start Diesel Generator 1A,

2) Close Reserve Auxiliary Source Breaker (BKR503), if voltage is present. If this source is not available, then

3) Close Tertiary Auxiliary Source Breaker (BKR501), if voltage is present and BKR611 is tripped. If this source is not available, then

4) Shed load on the 4160-V and 480-V buses and close Diesel Generator 1A Breaker (BKR509), if diesel generator voltage and frequency meet established criteria (maximum ten seconds from diesel engine start signal).

The automatic restoration of voltage sequence for 4160-V Bus 1-6, whose normal source of power is the Reserve Auxiliary Transformer, is as follows:

1) Start Diesel Generator 1B

2) Close Tertiary Auxiliary Source Breaker (BKR611), if voltage is present and BKR501 is tripped. If this source is not available, then

3) Close Reserve Auxiliary Transformer Breaker (BKR601), if voltage is present. If this source is not available, then

4) Shed load on the 4160-V and 480-V buses and close Diesel Generator 1B Breaker (BKR603) if voltage and frequency meet established criteria (maximum ten seconds from diesel engine start signal).

Once started, the diesel continues to run even though voltage may be restored from an off-site source of power. Manual shutdown of the diesels by the Control Room operator is always required (except for engine protection shutdowns).

Circuit breaker interlocks are provided to preclude interconnection of redundant emergency buses.

Breakers 1-501 and 1-611 allow the load groups associated with Diesel Generator 1A and Diesel Generator 1B, respectively, to be connected to a preferred power source, the Tertiary Auxiliary Transformer. This combination of load group connections is referred to in Safety Guide 6, Section D.2, which states: "A preferred power source bus, however, may serve redundant load groups".

Breaker 1-501 must be tripped before Diesel Generator 1A can be automatically connected to Bus 1-5 and BKR 1-611 must be tripped before Diesel Generator 1B can be automatically connected to Bus 1-6. Therefore, the redundant standby power sources cannot be automatically paralleled, satisfying Section D.4a of Safety Guide 6.

Breakers 1-510 and 1-602 provide a bus tie between the load group associated with Diesel Generator 1A and the load group associated with Diesel Generator 1B. These breakers can be closed by operator action only. To close BKR 1-510 or BKR 1-602, the following conditions must exist:

- No bus fault on Bus 1-5.
- No bus fault on Bus 1-6.
- No fault on cable between 1-510 and 1-602 (as monitored by independent lockout circuits).
- Breakers 1-503, 1-501, 1-511, 1-601, 1-610, and 1-611 are tripped.
- Either Diesel Generator 1A is supplying power to Bus 1-5 and BKR 1-603 (for Diesel Generator 1B) is tripped or Diesel Generator 1B is supplying power to Bus 1-6 and BKR 1-509 (for Diesel Generator 1A) is tripped.

These interlocks provide the necessary isolation as identified in Safety Guide 6 between redundant load groups.

Breakers 15211 and 16211 are bus tie breakers.

Both breakers can be closed only by operator action. The operator can close BKR 15211 or BKR 16211 only if:

- No fault has occurred on either Bus 1-52 or Bus 1-62, and
- No fault has occurred on the section of cable between 15211 and 16211, and
- Breaker 15201 and/or 16201 is open.

Breakers 15111 and 16111 can be closed by operator action only. The operator can close BKR 15111 or BKR 16111 only if:

- No fault has occurred on either Bus 1-51 or 1-61, and
- No fault has occurred on the section of cable between 15111 and 16111, and
- Breaker 15101 and/or 16101 is open.

Using 4160-V Bus 1-5 and assuming the loss of off-site power, the following steps take place:

a. Start Diesel Generator 1A

b. Trip all 4160-V source breakers and the bus tie breaker (BKRs 501, 503, 509, 510 and 511);

c. Trip all 4160-V motor loads (BKRs 502, 504, 506, 507 and 508);

d. Trip selected 480-V loads (BKRs 15203, 15104, 15105, 15108, 15109 and 15212).

e. Close the diesel breaker (BKR 509) after the unit comes up to speed and voltage (maximum ten seconds from diesel engine start signal).

If there is a requirement for engineered safety features operation coincident with bus undervoltage, step "e" above is automatically followed by the sequential starting of the engineered safety feature equipment. A group of equipment is directly connected to the bus (see Table 8.2-1, Sequence 1.1, 1.2, 1.3 and 1.4) and the loads are picked up by the diesel generators immediately upon closing of the diesel breakers. This total load is minimal, < 330-kW. (Should the requirement for engineered safety features operation occur when voltage is present, the diesel generator is started and this same sequence is followed with the exception that the containment spray pump is allowed to start immediately if containment Hi-Hi pressure is present.). This loading sequence for Diesel Generator 1A is as follows (major loads only), continuing from step "e" (see Table 8.2-1 and Figure 8.2-7):

	Max. Time Lapse (Sec)
f. (Step "0") Motor Operated Valves	0
g. (Step "1") Start Safety Injection Pump 1A	7
h. (Step "2") Start Residual Heat Removal Pump 1A	12
i. (Step "3") Start Shield Bldg. Fan 1A, Start Zone SV equipment, Start Containment Spray Pump 1A if containment Hi-Hi pressure is present	20
j. (Step "4") Start Service Water Pump 1A1	25
k. (Step "5") Start Containment Fan Coil Units 1A/1B	30
l. (Step "6") Start Auxiliary Feedwater Pump 1A	35
m. (Step "7") Start Component Cooling Pump 1A	40
n. (Step "8") Start Service Water Pump 1A2	45
o. (Step "9") Manual or Auto start of any auxiliary as required for safe plant operation	53
p. (Step "10") Manual or Auto start of any auxiliary as required for safe plant operation	63

NOTE: "Max Time Lapse" is the maximum time to initiation of closing the branch feeder breaker following the closure of the diesel generator source breaker. The "Max Time Lapse" specified above through step "n" is a maximum time that should not be exceeded. The "Max Time Lapse" specified above for step "o", as one input to service water isolation to the Turbine Building, is a maximum time that should not be exceeded. Because all other step "o" and "p" loads are defined above as those loads that will manually or automatically start as required, the "Max Time Lapse" specified above should not be considered absolute with nominal deviation of minor safety significance.

Starting of the containment spray pumps, initiated by Hi-Hi containment pressure, is accomplished simultaneously with any of the above steps following the starting of the residual heat removal pump when the diesel is required to supply power to the bus. When the bus is supplied from a transformer source the containment spray pump is started immediately on Hi-Hi containment pressure. The diesel generator automatic loading sequence through step 8, including 10 seconds for engine starting, will be accomplished in approximately fifty-five seconds as shown by Figure 8.2-7. As stated in Section 14.3.4, the containment pressure analysis assumes a delay of 137.7 and 85.3 seconds respectively, to supply design containment cooling from containment spray and fan-coil units.

The automatic sequences for Bus 1-6 and for Buses 1-61 and 1-62 associated with Diesel Generator 1B are similar to those described for Bus 1-5. Loads to be carried by a diesel generator are summarized in Table 8.2-1.

Should any of the feeder breakers, associated with the above (safety features) large (non-MCC feed) pump or fan motors, trip due to overcurrent, they can be re-closed from the control room. The electrical overload protection for the engineered safety feature fan, pump, and valve motors are not actually applied as overload protection. The motors are conservatively operated with respect to their rating and an overload occurs only as a major malfunction. Therefore, the overload protection isolates the malfunctioning component before it can make the bus breaker trip, causing loss of power to all other components in that circuit. Overload trip elements on the reversing starters associated with the various motor-operated valves and non-reversing starters associated with small pump or fan motors can and must be reset at the motor control centers. If the diesel generator is overloaded, an alarm is annunciated in the control room. The diesel generator is not protected by overload devices.

Load Evaluation

Diesel Generators

Each diesel generator is sized to start and carry the engineered safety features required for a post-blowdown containment pressure transient.

Selected generator nameplate data is as follows:

- Electro-Motive Division of General Motors Corporation
- Model A-20-C1, Serial Nos. 70-J1-1029 and 1039
- 2400/4160-V, 60 Hertz, Amps 782/452, 3 phase
- 3250-kVA, Temperature rise 85°C Stator-Thermometer
- Temperature rise 60°C, Rotor-Resistance
- 900 RPM, Power Factor 0.8
- 3575-kVA Peak, 2000 hours per year
- Temperature rise 105°C, Stator-Thermometer
- Temperature rise 70°C, Rotor-Resistance.
- Insulation Class, H-Stator and F-Rotor

Additional operating characteristics of the generator follow:

Capable of being started and ready to accept load in ten seconds and capable of being fully loaded within twenty seconds.

Capable of operating continuously at rated kVA output at any power factor between rated lagging and unity, at any voltage within ±5% of rated voltage.

Capable of tolerating for thirty seconds without injury a three-phase short circuit at its terminals when operating at rated kVA and power factor, 5% overvoltage and fixed excitation.

Compliance to Regulatory Guide 1.9

1. Sizing of generator power requirements.

Motors - All motors are the standard rating above the normal load. The service factor is added to the motor to cover fan and pump run-out. Checks have been made to assure the run-out is within the service factor. Motor power requirements were calculated as follows:

100 hp and larger, at 93% efficiency from manufacturers certified test data and handbooks. Less than 100 hp, at 88% efficiency from handbook data.

Brake hp was used for larger motors where certified test data was available.

KVA - Loads for transformers, etc., were calculated at 80% power factor.

Heater-Loads were taken at rated kW.

2. Generating load ratings:

Continuous	3250 kVA	100.0%
Continuous	2600 kW at 0.8 P.F.	100.0%
Overload, 2000 hours per year	2860 kW at 0.8 P.F.	110.0%
Overload, 7 days per year	2950 kW at 0.8 P.F.	113.5%
Overload, 30 minutes per year	3050 kW at 0.8 P.F.	117.3%

3. Generator rating criteria:

Regulatory Guide 1.9, dated December 1979 states that:

"Conformance with the requirements of IEEE Std 387-1977, "IEEE Standard criteria for Diesel Generator Units applied as Standby Power Supplies for Nuclear Power Generating Stations", dated June 17, 1977 is acceptable for meeting the requirements of the principle design criteria and qualification testing of diesel generator units used as on-site electric power systems for nuclear power plants".

The IEEE Standard 387-1977 states that:

"5.2.3 Operation Application Rule (see 3.7.1 and 3.7.2). The diesel generator units may be utilized to the limit of their power capabilities as defined by the continuous and short time ratings".

"3.7.1 Continuous Rating - The electric power output capability that the diesel generator unit can maintain in the service environment for 8760 h of operation per (common) year with only scheduled outages for maintenance".

"3.7.2 Short Time Rating - The electric power output capability that the diesel generator unit can maintain in the service environment for 2 h in any 24 h period, without exceeding the manufacturer's design limits and without reducing the maintenance interval established for the continuous rating".

NOTE: "Operation at this rating does not limit the use of the diesel generator unit at its continuous rating".

Table 8.2-1 lists the diesel-generator loads and the times that they will sequence on if required. The maximum connected loads are 3701.4 kW for DG 1A and 3523.3 kW for DG 1B. Table 8.2-1 also gives a time dependent load list, which shows that the highest estimated loads are 2919.8 and 2899.1 kW for each respective diesel generator, which occurs from one to sixty minute into the loading sequence. After adding safeguard station service transformer loss loads of 25.5 kW and 20.4 kW the maximum diesel generator loads are 2945.3 kW for DG 1A and 2919.5 kW for DG 1B. These loads are both less than the seven-day per year overload rating of 2950 kW for the diesel generators.

Operation of the safeguard diesel generators at frequencies other than 60 hertz, as allowed by the governor speed setting, have been shown by calculation to be within the various generator ratings.

The diesel generator ratings given in Item 2 above do not match the Short Time Rating definition of IEEE Std. 387-1977, as they were determined before 1977. We do, however, meet the intent of the Standard in that the diesel generators do not exceed the defined (by the manufacturer) load ratings. Therefore, the requirements of Regulatory Guide 1.9 are met.

4. Generator loading and sequence on safety injection signal is shown in Figure 8.2-7. The time sequence is after the closing of the diesel generator breaker. The maximum allowable time lapse for the load to come on was originally specified by Westinghouse and is used to support transient analysis. The normal time to pick up the load is the automatic timer setting. Table 8.2-1 lists the specific loads.

The criteria used in determining the a-c loads assigned to the emergency buses were:

a. Those loads, which are essential to safety-related functions and which if the power source failed, could affect public health and safety.

b. Those loads which if the power source failed would cause severe economic loss or cause the plant to experience an extended outage.

Batteries and Battery Chargers

Each of the plant's four 125-V station batteries has been sized to carry the expected shutdown loads following a plant trip and a loss of all AC power for a period of eight hours without the battery terminal voltage falling below 105 V. The 250-V station battery has been sized to carry its loads following a plant trip and a loss of all AC power for a period of two hours.

The safeguard batteries (BRA101 and BRB101) are C and D Charter Power Type LCR-19 1304 AH (8 hour), 1054 AH (3 hour), 647 AH (1 hour), and 1234 AH (1 min.). Major loads, with their approximate operating times on each battery, are listed in Table 8.2-2. The non-safeguard batteries (BRC101 and BRD101) are Exide Corp. Type FTC-21 1680 AH (8 hour), 1236 AH (3 hour), 750 AH (1 hour), and 1260 AH (1 min.). The 250-V non-safeguard battery (BRE101) is a C and D Power Systems Type 2LCR-15, 700 AH (2 hour).

Each of the three safeguard battery chargers has been sized to recharge either of the above partially discharged safeguard batteries within twenty-four hours, while carrying its normal load. Partially discharged is defined as any condition between fully discharged (105-V) and nominal (125-V). Normal voltage when on charger is 129 to 135-V (2.19 to 2.29-V per cell).

The battery chargers are each supplied with a d-c ammeter to continuously indicate the charger's current output. Each battery charger is also supplied with a d-c voltmeter on the line side of the charger output circuit breaker. This voltmeter will indicate the charger or battery voltage whichever is higher. A d-c ammeter on each of the main load side buses of the batteries continually indicate the total load current on each battery train. On a monthly basis the specific gravities of the pilot cells are checked and recorded.

The actual charge stored in the batteries can be related to the monitored parameters in the following ways:

a. The batteries are of the lead calcium type and are floated at 132-V d-c. As long as the specific gravities of the cells are at least 1.200 and the d-c voltmeter reads 132-V then the battery is considered fully charged.

b. The battery chargers are each rated at 150-ampere d-c output and will supply the plant load under normal conditions. The current-limiting feature on these chargers is set at approximately 172.5 amperes. In a situation when the ammeter on the battery main bus reads above 0.0 amperes then the battery would be discharging as indicated on the main bus ammeter.

Since the battery and charger share the loads on the bus under the normal condition, the battery must have sufficient ampere-hour capacity to carry the total loads consisting of two classes as follows:

a. The momentary load, such as closing and tripping of switchgear, involves the one-minute rating of a battery, though the time duration of the operation is but a few cycles.

b. The continuous load usually involves the battery's three to eight-hour rating, although longer time periods are sometimes used. The load consists of indicating lamps, holding coils for relays and any other equipment continuously drawing current from the control bus.

Reliability Assurance

The electrical system equipment is arranged so that no single incident can inactivate enough engineered safety features equipment to jeopardize plant safety. The 4160-V equipment is supplied from 6 buses, the 480-V equipment from 11 buses.

All Class 1E electrical equipment complies with IEEE Standard 344-1971, Trial Use Guide for Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations.

Two separate off-site power sources serve the 4160-V buses supplying power to the engineered safety features equipment. One of these is from the 138-kV portion of the substation; the second is from the tertiary winding of the substation autotransformer via an underground 13.8-kV circuit to the plant.

Separation is maintained in both the 4160-V and the 480-V systems to allow the plant auxiliary equipment to be arranged electrically so that redundant items receive power from two different buses.

For example, one complement of engineered safety features equipment is supplied from Bus 1-5 (4160-V) and Buses 1-51 and 1-52 (480-V) while the other complement is supplied from Bus 1-6 (4160-V) and Buses 1-61 and 1-62 (480-V). The cable tray system for one complement is independent of the cable tray system for the other complement; there are no crossties. This design assures the separation and independence of the two systems.

One off-site source of power can supply sufficient power to run normal operating equipment. Any one of the four transmission lines can supply all the plant auxiliary power. A low-voltage station auxiliary transformer can supply all the auxiliary loads for the plant.

Each diesel generator has capacity enough to sequentially start and run a fully loaded set of engineered safety features equipment. These safety features can adequately cool the core for any loss-of-coolant incident, and maintain the containment pressure within the design value.

One battery charger is in service on each battery so that the batteries are always at full charge in anticipation of loss of a-c power. This insures that adequate d-c power is available for starting the diesel generators and for other emergency uses.

The physical barrier provided between the emergency diesel engine generator sets consists in part of a Class I reinforced concrete wall 18 inches thick and the remainder a reinforced concrete block wall 12 inches thick. The doors and ventilation exhaust louvers are all Underwriters' Laboratories construction. All other openings in the barrier are sealed with fire retardant materials to maintain fire separation of the two diesel generator units.

The only potential for an explosion in the diesel generator rooms exists within a diesel engine crankcase. The rooms have sufficient volume and are vented to preclude a pressure rise that would endanger the integrity of the room walls.

In the event of a service water line break in the area between diesel generator rooms some water leakage would occur into the diesel generator rooms. Leakage through the door into diesel generator room 1B would flow to the floor drain if enough water was present to overflow the curb. Leakage through the door and through the trench into diesel generator room 1A would flow to the trench drain. All other openings into the diesel generator rooms from the tunnel are at higher elevations.

Both diesel generator rooms have double doors appropriately strengthened to prevent possible flooding. The trench into diesel generator room 1A was plugged to restrict leakage into the room.

Water flowing from the hypothetical service water line break would return to the screenhouse along the floor of the tunnel. Water entering the screenhouse would drain to the circulating water pump elevation where approximately 382,000 gallons are required to flood to the 586-foot elevation. Maximum possible service water pump run-out for two pumps would be less than 20,000 gpm total. The operator has over nineteen minutes to respond to the low service water pressure alarm on one header, isolate that header from the auxiliary building and then trip the pumps.

The rupture of a service water line in an Emergency Diesel Generator Room could result in the loss of the generator or the Safeguards bus in that room. Administrative operation from the Control Room of Type I Service Water valving would isolate the break and if required, realign the Service Water supplies through the intact piping from the operating Service Water Pumps.

Surveillance Requirements

The monthly tests specified for the diesel generators will demonstrate their continued capability to start and carry rated load. The fuel supplies and starting circuits and controls are continuously monitored, and abnormal conditions in these systems would be indicated by an alarm without need for test startup (Reference 2).

The less frequent overall system test demonstrates that the emergency power system and the control system for the engineered safety features equipment function automatically in the event of loss of all other sources of a-c power, and that the diesel generators start automatically in the event of a loss-of-coolant accident. This test demonstrates proper tripping of motor feeder breakers, main supply and tie breakers on the affected bus, and sequential starting of essential equipment, to the extent possible, as well as the operability of the diesel generators.

The specified test frequencies provide reasonable assurance that any mechanical or electrical deficiency is detected and corrected before it can result in failure of one emergency power supply to respond when called upon to function.

Station batteries will deteriorate with time, but precipitous failure is extremely unlikely. The continuous and periodic surveillance performed on the batteries will demonstrate battery degradation long before a cell becomes unserviceable or fails.

If a battery cell has deteriorated, or if a connection is loose, the voltage under load will drop excessively, indicating need for replacement or maintenance.

8.2.4 STATION BLACKOUT

Introduction

On July 21, 1988 the Code of Federal Regulations 10 CFR Part 50, was amended to include a new Section 50.63 entitled, "Loss of All Alternating Current Power", (Station Blackout). The station blackout (SBO) rule requires that each light-water-cooled nuclear power plant be able to withstand and recover from an SBO of specified duration.

Station Blackout Duration

The Kewaunee SBO duration is 4 hours, based on a plant AC power design characteristic Group P1, an emergency AC (EAC) power configuration Group C, and a target Emergency Diesel Generator (EDG) reliability of 0.95 (see NRC SER in reference 4).

Alternate AC (AAC) Power Source

The existing Technical Support Center (TSC) diesel generator will be used as an AAC source. The TSC diesel generator is an independent, non-class 1E, 600 kW (1000 hr/year standby rating) diesel generator that provides emergency power to 480-V Bus 1-46 for TSC equipment. For SBO purposes, a connection can be made between this bus and the 480-V safety Bus 1-52. Normal isolation between the two buses is provided by a Class 1E breaker at Bus 1-52 and a non-class 1E breaker at Bus 1-46. For SBO, selected non-essential loads will be stripped from each of the two buses and the two breakers will close to provide power to essential loads on both buses. The total load to be powered within 1 hour following the onset of the SBO is calculated to be approximately 587 kW.

Condensate Inventory

The Technical Specifications (TS) provide for a minimum of 39,000 gallons of condensate inventory. This is sufficient for 4 hours of decay heat removal.

Class 1E Battery Capacity

Battery capacity calculations exist for the BRA101 and BRB101 Class 1E batteries and on the BRD101 non-class 1E battery. They are based on the IEEE-485 methodology and a duty cycle of 8 hours. The 8-hour duty cycle capability has been verified by test per the guidance of IEEE 450-1987. Based on the above, the batteries are adequate, with considerable margin for the required 4-hour SBO duration.

Compressed Air

The air-operated valves relied upon to cope with an SBO event for 4 hours will be equipped with DC powered solenoid valves and backup air (nitrogen) supplies.

Effects of Loss of Ventilation

Steady state heat-up analyses were performed to determine the effects of loss of ventilation in the battery rooms, control room, relay room, charging pump room, turbine driven auxiliary feedwater pump room, containment and steam generator power operated relief valve areas. The calculated steady state temperatures for these rooms are well below the temperature limits described in NUMARC 87-00, Section 2.7. Kewaunee has procedures and operator training to ensure opening of doors #45 and #48 to Battery Rooms 1A and 1B, respectively, and relay room cabinet doors within 30 minutes of the onset of an SBO event.

Containment Isolation

Table 5.2-2, "Containment Penetrations" in the Updated Safety Analysis Report has been reviewed in accordance with the guidelines described in NRC Regulatory Guide (RG) 1.155 to ensure that appropriate containment integrity will be provided during an SBO event.

Reactor Coolant Inventory

The TSC diesel generator will power one of two charging pumps each having a capability of supplying water at the rate of 60 gpm. This will provide makeup for a total of 50 gpm reactor coolant pumps seal leakage (25 gpm per pump) and 10 gpm reactor coolant system leakage (maximum allowed by the TS). The water supply for the pumps will be from the refueling water storage tank, which has a TS minimum of 272,500 gallons of water.

Procedures and Training

Operator actions will be required for an SBO event. In addition to the opening of selected room and cabinet doors, and the valve operations required to align the charging pump and AFW pump, a number of electrical breaker operations are required to shed loads not required during an SBO, and to align the TSC diesel generator to the required loads.

The physical location of the buses and MCCs involved are in close proximity to each other. The inability of an operator to open any single motor control center (MCC) breaker will not result in the TSC diesel generator exceeding its overload rating. In addition to proceduralized operator actions and training provided to individuals, the breakers which are required to be opened are also locally identified.

The actions required during an SBO event can be accomplished in the 1-hour time frame specified for the AAC source to power the SBO loads.

SBO Modifications

An air (nitrogen) supply has been provided for RCS inventory valve CVC-7 in order to provide control room control of the amount of charging flow to the reactor coolant loop versus the reactor coolant pump seals.

The steam generator power operated relief valves SD3A and SD3B have each been equipped with a DC solenoid valve and a backup air (nitrogen) supply. Additional lighting has been provided at MCC 1-52E (charging pump 1A) and outside of Battery Room A.

Quality Assurance and Technical Specifications

The QA type 2 classification is consistent with the requirements of RG 1.155, Section 3.5.

EDG Reliability Program

The EDG reliability program, including the TSC diesel generator, meets the intent and guidance provided in RG 1.155, Section 1.2.

REFERENCES - SECTION 8.2

1. NRC Safety Evaluation Report, M.B. Fairtile (NRC) to D.C. Hintz (WPS), Letter No. K-86-136 dated July 3, 1986

2. NRC Safety Evaluation Report, S.A. Varga (NRC) to E.R. Mathews (WPS), Letter No. K-81-189 dated November 12, 1981

3. NRC Safety Evaluation Report, A.T. Gody, Jr. (NRC) to K.H. Evers (WPS), Letter No. K-89-212 dated October 25, 1989

4. NRC Supplemental Safety Evaluation Report, A.G. Hansen (NRC) to C.A. Schrock (WPS), Letter No. K-92-215 dated November 19, 1992

5. NRC Safety Evaluation Reports:

    a. A. Schwencer (NRC) to E.W. James (WPS), Letter dated June 3, 1977
    b. R.B. Licciardo (NRC) to E.R. Matthews (WPS), Letter No. K-82-074 dated April 30, 1982
    c. J.D. Neighbors (NRC) to C.W. Giesler (WPS), Letter No. K-84-091 dated April 30, 1982
    d. R.J. Laufer (NRC) to C.A. Schrock (WPS), Letter No. K-93-203 dated September 30, 1993
    e. R.J. Laufer (NRC) to M.L. Marchi (WPS), Letter No. K-96-098 dated June 12, 1996

6. "Stability Studies Associated With Nuclear Power Plants In Wisconsin", E.R. Mathews, Wisconsin Public Service Corporation, October 11, 1968

8.2 ELECTRICAL SYSTEM

8.2.1 NETWORK INTERCONNECTIONS

Electrical energy generated at 20-kV is transformed to 345-kV by the Main Transformer, a bank of three single-phase units, which in total are rated at 580-mVA. It is delivered through a 345-kV, 25,000-mVA (interrupting rating), 1600-ampere (552-mVA) circuit breaker to the 345/138-kV switching station located at the plant site, as shown in Figure 8.2-1. The electrical output is integrated into the American Transmission Company's 345-kV and 138-kV transmission systems. These systems are interconnected at the plant site substation by a transformer rated at 300 mVA. The 345-kV transmission system has interconnections with Northern States Power Company. Two 345-kV transmission lines are connected to the plant switching station and are on separate line structures in order to minimize the possibility of losing more than one circuit at a time. Either line is capable of carrying the full output of the generator. In addition, two 138-kV lines are connected to the plant switching station from the 138-kV grid system. These two lines together are capable of carrying the full output of the generator, but would be limited by the 300-mVA Autotransformer on a loss of both 345-kV lines.

An analysis of the integrated 345/138-kV power system has been made and shows that a fault on any one of the transmission lines, any bus section at the Kewaunee Substation, or the loss of the Kewaunee generator will not cause a cascading failure on the transmission system, thereby insuring an off-site power supply to the plant for any of the aforementioned failures. (See Stability Study in Reference 6).

The centerline of the two 138-kV lines is 265 feet south of the southern-most two 345-kV lines on a separate right-of-way as they leave the plant property. About one-half mile west of the substation the two 345-kV lines turn: one, line R-304, goes north and the other, line Q-303, turns south at a dead-end tower. Line Q-303 crosses over both 138-kV lines, one before the dead-end tower and one after (see Figure 8.2-1a). Since dead-end tower failure is not considered a credible accident, failure of a 345-kV line structure at some point could only cause a failure of one 138-kV line. There is no area where a failure of a 138-kV line structure can cause a failure of either 345-kV line. Thus, there is no single failure that can disable more than two lines. This results in three pairs of physically independent sources of offsite power (see NRC SER in Reference 1).

The 138-kV overhead transmission line from the substation into the plant on-site distribution system through the Reserve (Startup) Auxiliary Transformer is bifurcated at the substation to connect to either the East or West 138-kV buses. Each leg of the bifurcation is separated from the respective bus with a 138-kV oil circuit breaker. The controls for the 138-kV breakers are separated into two distinct 125-V d-c branch circuits; one serving the breaker closing circuit and one of the two trip circuits with the primary relaying; the other serving the second trip coil and the backup relaying. These breakers are manually controlled from the plant Electrical Vertical Panel A.

Similar connections to the East and West 138-kV buses are made to the 345-138-kV autotransformer, which can be used to energize the autotransformer if the need arises. These breakers are manually controlled from the plant Electrical Vertical Panel A, or local control in the substation control house and by operator choice can be transferred to system operating office supervisory control.

The West 138-kV bus is connected to a capacitor bank with a 138-kV oil circuit breaker (see Figure 8.2-1). Each of the four-capacitor banks have switching devices. The breaker and switching devices are normally under system operating office supervisory control and by operator choice, can be transferred to control from Electrical Vertical Panel A.

The 345-kV system can be and normally will be used to energize the 345-138-kV Autotransformer. The 345-KV oil circuit breakers are controlled similar to the East and West 138-kV bus oil circuit breakers described above.

The tertiary winding of the autotransformer is used to furnish power to the 13.8-kV Tertiary Auxiliary Transformer via an underground insulated power cable. This cable becomes the second of the two physically independent circuits to provide off-site power to the on-site distribution systems.

Both of the above circuits will normally be energized at all times and will be connected to one or the other of the engineered safeguards buses at all times. Thus, loss of the reactor, turbine generator, Main Station Auxiliary source of power does not even require a transfer for the safeguards buses. In the case of an engineered safeguards bus energized by one of the on-site power sources, i.e., the diesel-generator, and subsequent loss of the diesel-generator, the automatic transfer system will search for a transformer source and automatically close in its breaker. The above postulated condition could only occur after a loss of all off-site power and subsequent restoration of at least one source, prior to manual operator actions per emergency instructions.

Thus, loss of power from the nuclear unit should not affect the availability of power from the two off-site transmission circuits or the two standby power sources. Loss of either transmission circuit should not affect the other transmission circuit, the two standby power sources or the nuclear unit as a source. Finally, loss of one or both of the standby power sources should not affect the availability of the transmission sources or the nuclear unit.

Substation D-C System

The substation d-c system consists of two separate systems; the 48-V d-c distribution system furnishing power to the solid-state relay systems and the 125-V d-c distribution system furnishing control power and additional electro-mechanical and microprocessor based relay systems.

The 48-V d-c distribution system consists of redundant battery chargers, one battery and two distribution cabinets.

The 125-V d-c distribution system consists of redundant battery chargers, one battery and four distribution cabinets.

The battery chargers are furnished with a-c and d-c (output) failure relays and redundant a-c sources. High and low voltage alarms are provided for the d-c circuit.

The 125-V d-c distribution system is arranged so that the two circuits used to control each high-voltage circuit breaker emanate from different distribution cabinets. Each branch circuit in the three breaker distribution cabinets is individually monitored and alarmed on loss of d-c voltage. These alarms are displayed in the substation control house and are transmitted to the plant control room annunciator as a substation alarm. The branch circuits in the fuse distribution cabinet are not individually monitored and alarmed on loss of d-c voltage. A loss of power to any of the microprocessor based relays fed from the fuse cabinet would be indicated locally on the relay and transmit an alarm to the transmission system operator. The overall effect is that of a dual supply to the high-voltage breaker control trip elements with alarm should any portion of the supply system become abnormal.

The oil circuit breakers can close on 90-V d-c and trip on 70-V d-c. Each battery charger is equipped with a low voltage alarm set at 122-V d-c. The battery is considered to be fully discharged when the voltage reaches 105-V d-c. The actual closing energy is supplied from air compressors also integral with each circuit breaker. The air compressors are a-c powered. The storage cylinders have sufficient capacity when fully charged for five operations if a-c were lost. These breakers can be manually tripped at the breaker.

Loss of the substation battery and a concurrent fault on one of the transmission lines wherein the plant would continue to supply power to the fault is a set of postulated conditions, which would lead to an undetected failure. Under these postulated conditions, the fault would clear when the transmission-line remote terminal breakers opened. The Kewaunee breakers can be manually tripped and, thus, isolate the fault, thereby restoring off-site power to the plant via the remaining transmission lines.

Thus, given an undetected failure, the capability to clear and restore off-site power to the Kewaunee plant will not be lost, and the restoration of off-site power (assuming the grid is available) will be made within the time period (seven days) in which the plant can be maintained in a safe condition without off-site a-c power.

8.2.2 PLANT DISTRIBUTION SYSTEM

The Auxiliary Electrical System is designed to provide a simple arrangement of buses requiring the minimum of switching to restore power to a bus in the event that the normal supply to that bus is lost.

Single Line Diagrams

The basic components of the plant electrical system are shown on the Single Line or Circuit Diagrams, Figures 8.2-2 and 8.2-3. These figures show the 20-kV, 4160-V, 480-V and instrument bus a-c systems and the 125-V and 250-V d-c systems. In addition, Figures 8.2-1, 8.2-2 and 8.2-3 show the basic elements of the 13.8-kV, 138-kV and 345-kV substation systems.

Main Auxiliary, Reserve Auxiliary and Tertiary Auxiliary Transformers

The plant turbine-generator serves as the primary source of auxiliary electrical power during "on-the-line" operation. Power is supplied via a 20-4.16-kV, three-winding, Main Auxiliary Transformer, which is connected to the main leads from the turbine generator.

The primary sources of electrical power for the auxiliaries associated with engineered safety features during "on-the-line" operation of the plant are the Reserve Auxiliary Transformer and the Tertiary Auxiliary Transformer. Power is normally supplied to one bus (Bus 1-6) through the 138-4.16-kV, three-winding Reserve Auxiliary Transformer which is connected to the 138-kV portion of the Kewaunee Substation. Power is normally supplied to the second bus (Bus 1-5) through the 13.8-4.16-kV, two-winding Tertiary Auxiliary Transformer which is connected, by an underground line, to the 13.8-kV tertiary winding of the 345/138/13.8-kV auto transformer in the Kewaunee Substation. Either source can supply both buses.

Auxiliary power required during plant startup, shutdown and after reactor trip is supplied from the 13.8/138/345-kV Kewaunee Substation via the Reserve Auxiliary and Tertiary Auxiliary transformers. After turbine-generator trip, the auxiliaries on the 4160-V buses being fed by the Main Auxiliary Transformer are transferred by a fast bus transfer scheme using stored energy breakers to the Reserve Auxiliary Transformer. Control power for the plant auxiliary breakers is supplied by the plant batteries. The high-side (substation) breakers use the substation battery for control power.

4160-V System

The 4160-V system is divided into six buses, as shown in Figure 8.2-2. Buses 1-1 and 1-2 are connected via bus main breakers to the Main Auxiliary and Reserve Auxiliary Transformers. These buses supply power to the Reactor Coolant Pumps and the Feedwater Pumps.

Buses 1-3 and 1-4 are also connected via bus main breakers to the Main Auxiliary and Reserve Auxiliary Transformers. These buses supply power to the normal balance-of-plant auxiliaries, and each bus supplies power to three 4160 - 480-V station service transformers. A fourth transformer connected to bus 1-4 supplies power to the Technical Support Center. In addition, the Circulating Water Pumps, Condensate Pumps and the Heater Drain Pumps are directly connected to buses 1-3 and 1-4.

Buses 1-5 and 1-6 are connected via bus main breakers to the Main Auxiliary, Reserve Auxiliary, and Tertiary Auxiliary Transformers. In addition, each bus is directly fed via a main breaker by a diesel generator. The two buses are tied together via two bus tiebreakers in series, one on each bus. Each bus supplies two of the four 4160 - 480-V station service transformers for the plant's 480-V engineered safety features equipment. In addition, the Service Water Pumps, Auxiliary Feedwater Pumps, Safety Injection Pumps and the Residual Heat Removal Pumps are directly connected to buses 1-5 and 1-6.

Bus 1-5 is normally supplied from the Tertiary Auxiliary Transformer and Bus 1-6 is normally supplied from the Reserve Auxiliary Transformer. Thus, no transfer is required for the engineered safety features in the event of an incident.

The bus tie breakers between Bus 1-5 and Bus 1-6 can only be manually closed, but are interlocked so that the diesel generators cannot be operated in parallel.

480-V System

The 480-V system is divided into 11 load center or switchgear buses, as shown in Figure 8.2-2. Those fed from 4160-V buses 1-3 and 1-4 serve balance-of-plant loads, those fed from 4160-V buses 1-5 and 1-6 serve the loads associated with the engineered safety features equipment.

Transformers 1-32 and 1-42 are connected to 4160-V buses 1-3 and 1-4, respectively. Transformer 1-32 feeds 480-V bus 1-32; transformer 1-42 feeds 480-V bus 1-42. These components including the 480-V bus tie are assembled as a conventional, double-ended switchgear unit. In a similar manner buses 1-33/1-43 and 1-35/1-45 are connected to 4160-V buses 1-3 and 1-4. Bus 1-46, supplying the TSC, is connected to 4160-V bus 1-4.

The various motor control centers throughout the plant are then connected to these switchgear buses.

The power required for the 480-V engineered safety features and other vital plant loads is supplied from four 480-V buses fed from 4160-V buses 1-5 and 1-6. Transformer 1-51 is fed from 4160-V bus 1-5 through breaker 1-505 and supplies bus 1-51. This transformer, bus and breakers, including one bus tie, are assembled as a switchgear unit. In a similar manner, bus 1-52 is also connected via breaker 1-505 to 4160-V bus 1-5.

The large 480-V engineered safety features motors are connected to bus 1-51. Motor Control Centers supplying the smaller loads are fed from bus 1-52.

A redundant 480-V system is supplied by 4160-V bus 1-6 through 4160-V breaker 1-607.

MCC 1-5262 may be fed from either 480-V switchgear bus 1-52 or 480-V switchgear bus 1-62 through breakers 15209 or 16209 respectively, via a manually operated transfer switch. 480-V switchgear bus 1-52 may ultimately be fed from on-site power source Diesel Generator 1A and

480-V switchgear bus 1-62 may ultimately be fed from on-site power source Diesel Generator 1B. 480-V MCC 1-5262 does function as a swing bus between the two redundant on-site power distribution systems as discussed in Safety Guide 6. However, the mechanical operation of this switch allows MCC 1-5262 to be connected to only one 480-V switchgear bus at a time. Hence the redundant 480-V switchgear buses can never be paralleled through the operation of this switch. Therefore, the transfer of MCC 1-5262 does conform to the criteria outlined in Safety Guide 6. Furthermore, the components fed from MCC 1-5262 are not part of a redundant load group. The components are:

- Turbine Turning Gear
- Turbine HP Hydrogen Seal Oil Backup Pump
- Condensate Bypass All Heaters to Feedwater Pumps Motor-Operated Valve
- Waste Gas Compressor 1B
- Station and Instrument Air Compressor 1A

The intent of the transfer switch is to provide maintenance flexibility for these loads.

It should also be noted that 480-V switchgear buses are each protected from the transfer switch with a breaker.

The automatic transfer switch associated with MCC 1-52E provides BRA-106 with two possible sources of power, MCC 1-52E or MCC 1-52C. However, both MCC 1-52E and MCC 1-52C are associated with the same on-site power source (Diesel Generator 1A). Therefore, the transfer is not between power systems of redundant load groups and does not fall into the category of swing buses discussed in Safety Guide 6.

Likewise, the automatic transfer switch associated with 1-62E allows BRB-106 to be fed from either MCC 1-62E or MCC 1-62C. These MCCs are both within the same load group.

125-V and 250-V D-C System

The 125V and 250-V DC system is divided into five buses (two safeguard and three non-safeguard, see Figure 8.2-3) each with one battery and a battery charger, distribution panels and inverters. Components prefixed with BRA and BRB make up the safeguard DC system and those prefixed with BRC, BRD and BRE make up the non-safeguard system.

The DC power requirements of the engineered safety features (ESF) and other vital plant loads are supplied by the safeguard batteries (numbered BRA-101 and BRB-101). Each safeguard battery consists of 59 cells, each of which are the lead calcium type. The batteries are rated 125V DC, 1304 ampere-hours at the eight-hour rate without discharging below 1.78V per cell. Two main DC distribution panels (BRA-102 and BRB-102) are fed from these batteries via main fuses. The main distribution panels connect the battery to the battery charger, to the sub-distribution panels, and allow for the interconnection of the two buses through the bus tiebreakers.

BRA-102 supplies sub-distribution panel BRA-104. BRA-104 in turn supplies the control and excitation power for diesel generator 1A, the control power for ESF buses 1-5, 1-51, and 1-52, control and power to one-half of the redundant essential plant equipment required for safe shutdown in the event of loss of AC power and provides a standby power source for the safeguard inverters BRA-111 and BRA-112.

BRB-102 supplies sub-distribution panel BRB-104. BRB-104, in turn, supplies the control and excitation power for diesel generator 1B, the control power for ESF buses 1-6, 1-61, and 1-62, control and power to one-half of the redundant essential plant equipment required for safe shutdown in the event of loss of AC power and provides a standby power source for the safeguard inverters BRB-111 and BRB-112.

The balance of plant DC power requirements are supplied by three non-safeguard batteries (designated BRC-101, BRD-101 and BRE-101). Batteries BRC-101 and BRD-101 each consist of 59 cells and are of the lead calcium type, rated at 125V DC, 1680 ampere-hours at the eight hour rate to reach 1.78V per cell. Battery BRE-101 consists of 120 cells of the lead calcium type, rated at 250 V d-c, 694 ampere-hours at the two hour rate to reach 1.75V per cell. Each battery is connected to a main distribution panel (BRC-102, BRD-102 and BRE-102). The main distribution panel connects each battery to a battery charger, sub-distribution panel, bus tie (125V d-c batteries only) and inverter(s).

Distribution panels BRC-102 and BRD-102 supply sub-distribution panels BRC-103 and BRD-103, respectively. Panel BRC-102 is also a standby source for inverter BRC-109. The BRC and D-103 panels, in turn, supply Technical Support Center diesel generator control and excitation power, control power for non-ESF buses 1-1, 1-2, 1-3, 1-4, 1-32, 1-33, 1-35, 1-42, 1-43, 1-45, and 1-46, other non-safety related equipment sensitive to a loss of AC power and are a standby source for inverter BRD-109 and a proprietary inverter.

Distribution panel BRE-102 supplies AC drive units BRE-109 and BRE-110, which convert the 250V d-c input power to 230-V a-c output power for the turbine emergency oil pump and air side seal oil backup pump motors.

Each of the five battery buses is served by one connected battery charger. Each safeguard battery has provisions for connection of a spare portable charger. The spare safeguard charger can be moved to its designated mounting in either safeguard battery room and connected to the DC bus in the event of charger failure. The battery life to minimum voltage under maximum load will allow sufficient time to make this connection. The non-safeguard spare charger is permanently mounted between the normal 125V d-c chargers and can be connected to either bus as required. There is no spare 250V d-c charger.

The two bus tie breakers between 125-V d-c distribution cabinet BRA-102 and 125-V d-c distribution cabinet BRB-102, are manually operated. These breakers are strictly administratively controlled to prevent them from being closed during plant operation.

Instrument Bus

The 120-V a-c instrument supply is split into several buses as shown on the one line diagram Figure 8.2-3. There are four independent instrument buses, each fed by an inverter which, in turn, is fed from each of the d-c buses. A fifth independent non-interruptible bus, fed by an ESF motor control center, supplies the rod position indicators. The sixth and seventh independent buses, each fed by an inverter, supply the plant process control computer and cabinet BRD-115, respectively. There are two additional independent buses; each fed from an ESF motor control center through a transformer.

Evaluation of Layout and Load Distribution

The physical location of electrical distribution system equipment is such as to minimize vulnerability of vital circuits to physical damage as a result of accidents.

The Main Auxiliary, Reserve Auxiliary and Tertiary Auxiliary Transformers are located outdoors and are physically separated from one another by firewalls. Each transformer cell, formed by the firewalls, has an automatic water spray system to extinguish and prevent the spread of fires.

The 4160-V switchgear and 480-V load centers are located in areas which minimize their exposure to mechanical, fire and water damage. This equipment is coordinated electrically to permit safe operation of the equipment under normal and short-circuit conditions.

The 480-V motor control centers are located in the areas of electrical load concentration. Those associated with the turbine-generator auxiliary system in general are located in the Turbine Building. Those associated with the nuclear steam system are located in the Auxiliary Building.

The application and routing of control, instrumentation and power cables are such as to minimize their vulnerability to damage from any source. The construction design drawings had second level review in accordance with the Kewaunee Construction Quality Assurance Program.

All cables are specified using conservative margins with respect to their current-carrying capacities, insulation properties and mechanical construction. The power conductors are three conductor, galvanized armored and installed in a single layer in ladder type cable trays, and clamped to insure that ample ventilation spacing is maintained throughout the run.

Bulk control power supply cables are treated as noted in the previous paragraph.

Control cables normally employ minimum size of #12 AWG when run in multi conductor cables in control trays. As there are few continuous loads on these circuits no attempt was made at derating. Continuously loaded circuits (current transformer secondaries) are sized by burden requirements of the circuit. In special cases #14 AWG multi-conductor control cable was allowed in cable trays. In these identified cases, a safety evaluation has been performed. Special cases may continue to allow #14 AWG multi-conductor cable to be used. These will require a safety evaluation and approval by the responsible engineer.

Cables that are run in trays have fire resistant jackets. Safety-related cables meet the environmental qualifications required by 10 CFR 50.49. Appropriate instrumentation cables are shielded as required to minimize induced-voltage interference. Wire cables related to engineered safety features and reactor protective systems are routed and installed to maintain the integrity of their respective redundant channels and protect them from physical damage.

Supports and cable trays for safety feature power cable systems are designed for 100% loading plus the forces generated by a seismic disturbance. Other cable systems are designed for 100% loading. The ladder fill is restricted to one layer, clamped in place to maintain 2 to 12 inch spacing between cables.

Cable trays for control and signal cable support systems and other safety-related systems are designed for forces generated by a seismic disturbance assuming maximum fill. Other cable systems are seismically designed for maximum fill. The tray fill is restricted to 50% of the tray's cross sectional area for safety-related cable and 60% for non-safety related cable.

Separation Criteria

Cable separation provides sufficient isolation between redundant systems so that no single failure or electrical incident can render both redundant systems inoperable or remove them from service.

To assure complete separation of Class IE circuitry that initiates and controls the transfer of power sources to the emergency a-c and emergency d-c distribution system, the following cable and cable tray separation techniques are used:

a. Each tray section of the cable tray system has an identifying code indicated on the electrical design drawings and this same identification is stenciled on the tray after it is installed. Stenciling is applied at each straight section of tray where the identifying code changes.

   On the electrical design drawings, the trays are identified by a number placed in a rectangular symbol. The trays used for the redundant safeguards equipment are further identified with a vertical line adjacent to the tray name symbol.

   The identifying code contains the designation "S5" or 'S6" where applicable to identify the safeguard train.

b. The two Safeguard tray systems are independent of each other such that they are physically separated a minimum of 3 feet horizontally and 3 feet vertically, except in the Relay Room where 1-foot minimum horizontal and vertical separation is required.

c. Each electrical cable has an identifying code indicated on the electrical design drawings and cable routing lists. This same cable number is affixed at each end of the cable with permanent tags.

The identifying code is based on the following systems.

The cable codes for the 4160-V switchgear; the 480-V switchgear and the 480-V motor control centers are a combination of letters and numbers that form a four part coding containing:

- Unit Number 1
- Source of Power
- Power or Control
- A serially assigned number which provides uniqueness.

A "5" in the second digit from the left is a cable in the "Safeguards 5" system; a "6" a cable in the "Safeguards 6" system (Example 1S5 xxx or 1S6 xxx).

Normal cable being fed from safeguards distribution equipment should have cable codes starting with 1NP followed by a serially assigned number which provides uniqueness. Some normal cables remain with their original safeguards power source type cable codes, since no comprehensive re-labeling program was implemented.

Redundant circuitry for reactor protection and engineered safety systems are separated into groups as follows.

Class IE cables are divided into the following groups:

- Group 1 (color code red)
  Red Instrument Channel
  Instrument Bus I

- Group 2 (color code yellow)
  Yellow Instrument Channel
  Instrument Bus IV

- Group 3 (color code green)
  Safeguard Train A, Battery 1A, Diesel Generator 1A, 4160-V Bus 1-5, 480-V Bus 1-51, 480-V Bus 1-52 and its associated Motor Control Centers.

- Group 4 (color code blue)
  Blue Instrument Channel
  Instrument Bus III

- Group 5 (color code white)
  White Instrument Channel
  Instrument Bus II

- Group 6 (color code orange)
  Safeguard Train B, Battery 1B, Diesel Generator 1B, 4160-V Bus 1-6, 480-V Bus 1-61, 480-V Bus 1-62 and its associated Motor Control Centers.

Each group is run in a separate tray, ladder, trough or conduit. These trays are identified on electrical drawings for engineered safety features, and are marked and color-coded on the actual hardware. All trays for the engineered safety feature equipment are Class I structures.

Within containment the tray systems for the four reactor protection instrument channels are separated 3 feet horizontally where they involve 2/4 logic and are separated approximately 20 feet where they involve 2/3 logic. Vertical separation is 5 feet where practical, and where impractical, barriers are installed. These barriers are solid metal covers on the lower trays.

For the non-Class-IE systems throughout the remainder of the plant, trays installed in stacks are spaced vertically with a minimum of 12 inches bottom to bottom in all areas. However, Class IE trays have a minimum of 15 inches bottom to bottom between trays of the same train. Class IE trays containing instrument, control or power cables have a minimum horizontal separation between redundant circuits of 36 inches. Redundant circuits are not permitted in the same tray or conduit. If closer spacing than 36 inches cannot be avoided, an approved barrier must be placed between the circuits. Cable trays are routed to avoid a fire hazard area, such as oil storage rooms, oil tanks, etc., whenever possible. When this cannot be done, the cable tray system is protected by fire resisting barriers. Where practical, these barriers will be tray covers. Whenever possible, a wall or floor has been introducted between trays carrying redundant safeguard circuits. Barriers are required where mutually redundant trays cross. The barriers shall extend to each side of the protected tray by a distance equal to approximately three times the wider of the two trays.

Mixing of power cables with control or instrument cable in the same tray is not permitted throughout the plant. Whenever a control and/or instrument cable tray and a power tray are in the same stack, the power tray is located in the top tier.

Trays for Train A and Train B are separated 3 feet horizontally and vertically except in the Relay Room where practical design considerations require 1-foot vertical and horizontal separation. The two trains are separated by 40 feet at the reactor containment vessel penetrations.

Power cables for engineered safeguards are kept strictly in cable trays so designated. Occasionally, a non-safety-related power cable may be run in a safeguards cable tray but a safeguards cable will never run in any tray other than its own system. Control cables are similarly separated and control and instrumentation of the same train designation may be run in the same control cable tray. Non-safety-related power, control or instrumentation cable shall not be permitted to cross over from one safeguards tray to another.

Where the wiring for redundant engineered safety features is within a single panel or panel section, this wiring is separated one group from the other, by a 6-inch air space or a fireproof barrier. The barriers are sheet metal or flexible metallic conduit. The flexible conduit may be applied to one train to separate it from the other train. Wiring not associated with either train may be grouped with one train but may not cross from one train bundle to the other train.

Where the approved logic required recognition of input signals from both A train and B train devices into common terminal blocks or operational devices, the interconnecting wiring can no longer retain train identity. Train A and B wiring shall maintain physical and electrical separation up to the termination point prior to where the interconnecting wiring loses its identity. This is an allowable exception to the above paragraph. The interconnecting wiring common to both trains shall not be termed "Normal", nor shall it be routed with normal wiring.

Cable trays used for redundant reactor protection systems, engineered safeguards systems and Class IE electrical systems have an identifying code number stenciled on them in color paint after they are installed. The number is applied whenever there is a change in identity or when passing through floor or wall openings. This number is applied prior to the pulling of any cables, and the color establishes the system to which it is assigned.

During the cable pulling operation, an intermittent colored stripe is applied to the cable as it leaves the reel. This color must match the color of the tray system in which it is installed. Normal (or non-Class IE) cables, if pulled into a colored tray system, will not have an identifying stripe on them.

The following colors are used for identification:

- Green - Safeguard Train A circuits
- Orange - Safeguard Train B circuits
- Red - Reactor Protection system
- White - Reactor Protection system
- Blue - Reactor Protection system
- Yellow - Reactor Protection system

Relay Room

The main (original) relay room is arranged in two groups of four rows of cabinets. The group to the east contains blue and white reactor protection channels, reactor protection train A, engineered safeguard train A, and other miscellaneous relay circuits (including two cabinets on the east wall). The group to the west contains yellow and red reactor protection channels, reactor protection train B, engineered safeguard B, and miscellaneous relay and metering circuits.

A relay room expansion to the south has two rows of cabinets with provision for a third. The cabinets contain plant process computer input/output and miscellaneous monitoring equipment.

The upper levels of the relay room are used for cable routing, as there is no separate room labeled "cable routing room".

Within the room the trays are arranged in four tiers. The arrangement of these tiers is such that instrument and control circuits of reactor protection and engineered safety features of like trains are stacked together or with normal instrumentation and control. The colored instrument channels for reactor protection are converted from trays to rigid conduit systems where they enter the relay room and then to 6-inch metal raceway as they pass over the instrument racks to facilitate rack output interconnections.

Horizontal fire barriers are provided between the Control Room and the Relay Room at the control consoles and panels.

Testing

Testing of the operator-activated Class IE circuitry that initiates and controls the connection of the buses to the power sources for the a-c emergency power system; the Main Auxiliary Transformer, the Tertiary Auxiliary Transformer, the Reserve Auxiliary Transformer, and the Diesel Generators; can be done by transferring the buses, one at a time, from one source to another with controls available to the operator in the Control Room. Manual switching of these source breakers occurs during periodic breaker maintenance, bus maintenance and diesel generator testing. Testing of the automatic-initiated Class IE circuitry that initiates and controls the connection of the buses to these sources can be performed by utilizing a switch in the Control Room which disconnects two bus undervoltage relays of one safeguard bus, thereby simulating loss of voltage, and activates the associated logic circuitry that is required to automatically restore power to the bus.

A sequential events recorder prints out the operation of each relay in the scheme providing printed proof of the circuitry's proper response. Upon successful completion of this test a green light glows to the right of the test switch.

Both the automatic circuitry and the manual circuitry can be tested by these methods during plant operation. Portions of the circuitry that cannot be conveniently tested with the plant in operation without temporarily interrupting circuit protection are the bus lockouts and the transformer lockouts. A switch is provided for each lockout, which isolates its output contacts allowing the lockout to operate without actually tripping any breakers. The continuity of transformer lockout relays is continually monitored by indicating lamps located in the Control Room.

The power sources for the d-c emergency power system associated with the Train A load group are Station Battery 1A and 480-V MCC 1-52C (via Battery Charger 1A). The power sources for the d-c emergency power system associated with the Train B load group are Station Battery 1B and 480-V MCC 1-62C (via battery charger 1B). The transfer of a d-c bus between its respective battery and battery charger can be tested by opening the 480-V breaker supplying power to the charger, thus simulating a loss of power to the 480-V bus. The d-c system should transfer to the battery as its source of power. If, after opening the 480-V breaker to the charger the d-c bus retains its voltage, the transfer was successful.

8.2.3 EMERGENCY POWER

Sources Description

Power sources for the engineered safety features are 4160-V Bus 1-5 and Bus 1-6. The normal source of power to Bus 1-5 is the Tertiary Auxiliary Transformer. The Reserve Auxiliary and Main Auxiliary Transformers provide backup sources, in that order. The normal source of power to Bus 1-6 is the Reserve Auxiliary Transformer. The Tertiary Auxiliary and Main Auxiliary Transformers provide backup sources, in that order. Thus, since the normal source of power for these buses is the 138/345-kV Kewaunee Substation, no transfer is required in the event of a turbine-generator trip.

If all other power sources should fail, two diesel generators are provided, one connected to 4160-V Bus 1-5 and one connected to 4160-V Bus 1-6. Each of these is a General Motors Corporation, Electro-Motive Division, Model A-20-C1, diesel engine-generator unit rated at 2600-kW, (2860-kW, 110% Overload, two thousand hours per year) 0.8 pf, 900 rpm, 4160-V, 3 phase, 60 Hertz. The generator has emergency ratings of 2950-kW for seven days continuous and 3050-kW for thirty minutes per year.

Each diesel generator, as a backup to the normal standby a-c power supply, is capable of sequentially starting and supplying the power requirements of one complete set of engineered safety features equipment. The electrical emergency power system logic diagrams are shown in Figures 8.2-4, 8.2-5, and 8.2-6. The units are located in separate rooms in Class I portion of the Administration Building. These rooms are heated; assuring that the diesel generators can be started in cold weather.

Service water for the Diesel Engine Cooling Water Heat Exchanger is supplied from separate service water headers for Diesel Generator 1A and 1B. The Cooling Water Heat Exchanger is an engine mounted water-to-water heat exchanger providing cooling for the engine jacket water and for the engine oil heat exchanger. Vent fans for each room provide a supply of combustion air into the Diesel Room. Separate startup air receivers and compressors are located just external to the rooms. Primary and reserve tanks of the air receivers supply compressed air to the dual Air Start System, the DG cooling water isolation valve actuators, and the Diesel Room Ventilation and combustion air dampers.

Each diesel generator is automatically started by either one of two pairs of air motors mounted on each side of the diesel (four air motors per engine). Each unit has its own independent starting system including a bank of four air storage tanks, two primary and reserve tanks, and one compressor powered from the 480-V emergency bus. An air cooler/dryer is installed on the discharge of each air start system compressor. The dry air improves the starting performance of the diesel engine. The primary or reserve tanks have sufficient storage to crank the engine for twenty seconds. The generator is capable of being started and ready to accept load in ten seconds.

Starting air is admitted from the storage tanks to the starting system through a pressure-reducing valve to supply air to the starters.

The following describes a typical diesel engine start sequence. The sequence stated assumes the air start motor priority selector switch is in the #1 position. The air start motor priority selector switch is typically rotated from set #1 to set #2 on a monthly basis. This ensures even run time on the air start motors.

When the diesel start signal is initiated, a start attempt is made through air start motor set #1. If the air start motor set #1 fails to engage within 2 seconds, a second start attempt is made with the same set of motors. If the air start motors still do not engage, and then after 5 seconds a third start attempt is made, this time using the second pair of air start motors (set #2). Air start motors set #2 will continue to attempt to start the diesel generator on a two second cycle, until the engine starts or 15 seconds after the start signal, whichever occurs first. The start signal also initiates starting of the fuel priming pump and the governor booster pump. If, after fifteen seconds, the diesel has not reached 200 rpm, a start failure signal opens the fault relay. Starting air is cut off, the fuel priming and governor booster pump are stopped. Operator action is then required for further start attempts. The fault relay in the diesel generator room must be reset and any faults causing the fault lockout must be corrected before the start signal will be effective again.

The start failure relay serves to indicate an abnormally long period of engine cranking without an engine start (fifteen seconds) and to prevent subsequent engine starting attempts until the cause of the engine start failure has been determined by operating personnel. The total air capacity available to crank the engine is twenty seconds per air starter-tank combination.

The following interlocks must be satisfied to automatically start the diesel engine:

a. Engine mounted LOCAL/REMOTE (AUTO/OFF/MAN) switch must be in REMOTE (AUTO) position for Diesel Generator 1A(1B). Control Room and local annunciation is given when this local switch is not in the REMOTE (AUTO) position.

b. Control Room PULLOUT/STOP/AUTO/START switch must be in AUTO position (maintained position). (The other maintained position of this switch is the PULLOUT position, which disables the engine starting circuit during engine maintenance).

c. 125-V d-c control power must be available at the diesel engine control panel. (The engine starters cannot be engaged if control power is not available; loss of control power is annunciated in the Control Room).

d. The engine must not be running.

e. Air pressure must be available to the starting air system (loss of air pressure on each starting air system is annunciated locally and in the Control Room).

The diesel engine interlocks itemized above can be periodically tested as follows:

a. Turning the engine mounted LOCAL/REMOTE (AUTO/OFF/MAN) switch from the REMOTE (AUTO) position will alarm the local and Control Room annunciators for Diesel Generator 1A(1B). Diesel generator operational testing verifies annunciator status; therefore, any failure to start due to mispositioning of the switch would identify the annunciator failure.

b. The stable PULLOUT position of the Control Room PULLOUT/STOP/AUTO/START switch is visibly different than the stable AUTO position in that the switch handle is slanted to the left of its normally vertical position (AUTO) and a distinctive silver colored switch shaft extension is visible.

c. Opening the 125-V d-c distribution breaker to the engine control cabinet alarms the Control Room annunciator.

d. Response of the air receiver pressure switches can be tested and calibrated by valving in the standby air receivers, valving out the on-line receivers, opening the air compressor circuit breaker, and opening the receiver drain valve until an alarm occurs on the local and Control Room annunciators.

The motor-driven compressor associated with each diesel is fed from the emergency bus supplied from the same diesel. The control voltage for each diesel starting system is from its associated 125-V d-c station battery.

An audible and visual alarm system is located in the control room and will alarm off-normal conditions of jacket water temperature, lube oil temperature, fuel oil level, starting air pressure and Diesel Generator stator hi temperature (1 of 12 inputs feeding the 4160 Volt Stator Temperature Hot annunciator). An alarm also sounds if a starting circuit is locked out, a control switch is not in "auto" position, or d-c power for the controls at the diesel generator is lost. The alarm in the control room also alerts the operator to other various off-normal conditions including jacket water expansion tank level and pressure, engine crankcase pressure, and fuel oil pressure. Local audio and visual alarms are also provided at each diesel generator.

Reference 2 is a safety evaluation in which the NRC has concluded that, based on the review of submitted information and on-site inspections, the status annunciators for the diesel generators are acceptable. The review was specifically intended to ensure that any deliberately induced condition which may disable the diesel generators, and which is expected to occur more frequently than once per year, is automatically annunciated in the Control Room with devices worded to alert the operator of their abnormal status.

Two 850-gallon "day" tanks are located in enclosures within each diesel generator room. The two tanks provide capacity for approximately four hours operation for one generator at full load. Two 35,000-gallon underground storage tanks supply fuel oil through immersion pumps to either pair of day tanks. Combined fuel capability of one storage tank and two day tanks would provide a minimum of 7 days fuel supply for one diesel generator (36,000 gallons of fuel oil), thus assuring adequate time to restore off-site power or to replenish fuel. The diesel fuel oil storage capacity requirements are consistent with those specified in ANSI N195-1976/ANS-59.51, Sections 5.2, 5.4 and 6.1. See Reference 3 and Technical Specification 3.7 for fuel oil storage requirements.

Loading Description

Each diesel generator is automatically started on the occurrence of either of the following incidents:

a. Undervoltage on the associated 4160-V bus (Bus 1-5 or Bus 1-6) provided that the low voltage is not caused by a fault which operates the bus lockout relay (see Reference 5 and Technical Specification);

b. Initiation of a Safety Injection Signal which will start both diesel generators.

With the occurrence of undervoltage on 4160-V Bus 1-5, whose normal source of power is the Tertiary Auxiliary Transformer, the automatic sequence is as follows:

1) Start Diesel Generator 1A,

2) Close Reserve Auxiliary Source Breaker (BKR503), if voltage is present. If this source is not available, then

3) Close Tertiary Auxiliary Source Breaker (BKR501), if voltage is present and BKR611 is tripped. If this source is not available, then

4) Shed load on the 4160-V and 480-V buses and close Diesel Generator 1A Breaker (BKR509), if diesel generator voltage and frequency meet established criteria (maximum ten seconds from diesel engine start signal).

The automatic restoration of voltage sequence for 4160-V Bus 1-6, whose normal source of power is the Reserve Auxiliary Transformer, is as follows:

1) Start Diesel Generator 1B

2) Close Tertiary Auxiliary Source Breaker (BKR611), if voltage is present and BKR501 is tripped. If this source is not available, then

3) Close Reserve Auxiliary Transformer Breaker (BKR601), if voltage is present. If this source is not available, then

4) Shed load on the 4160-V and 480-V buses and close Diesel Generator 1B Breaker (BKR603) if voltage and frequency meet established criteria (maximum ten seconds from diesel engine start signal).

Once started, the diesel continues to run even though voltage may be restored from an off-site source of power. Manual shutdown of the diesels by the Control Room operator is always required (except for engine protection shutdowns).

Circuit breaker interlocks are provided to preclude interconnection of redundant emergency buses.

Breakers 1-501 and 1-611 allow the load groups associated with Diesel Generator 1A and Diesel Generator 1B, respectively, to be connected to a preferred power source, the Tertiary Auxiliary Transformer. This combination of load group connections is referred to in Safety Guide 6, Section D.2, which states: "A preferred power source bus, however, may serve redundant load groups".

Breaker 1-501 must be tripped before Diesel Generator 1A can be automatically connected to Bus 1-5 and BKR 1-611 must be tripped before Diesel Generator 1B can be automatically connected to Bus 1-6. Therefore, the redundant standby power sources cannot be automatically paralleled, satisfying Section D.4a of Safety Guide 6.

Breakers 1-510 and 1-602 provide a bus tie between the load group associated with Diesel Generator 1A and the load group associated with Diesel Generator 1B. These breakers can be closed by operator action only. To close BKR 1-510 or BKR 1-602, the following conditions must exist:

- No bus fault on Bus 1-5.
- No bus fault on Bus 1-6.
- No fault on cable between 1-510 and 1-602 (as monitored by independent lockout circuits).
- Breakers 1-503, 1-501, 1-511, 1-601, 1-610, and 1-611 are tripped.
- Either Diesel Generator 1A is supplying power to Bus 1-5 and BKR 1-603 (for Diesel Generator 1B) is tripped or Diesel Generator 1B is supplying power to Bus 1-6 and BKR 1-509 (for Diesel Generator 1A) is tripped.

These interlocks provide the necessary isolation as identified in Safety Guide 6 between redundant load groups.

Breakers 15211 and 16211 are bus tie breakers.

Both breakers can be closed only by operator action. The operator can close BKR 15211 or BKR 16211 only if:

- No fault has occurred on either Bus 1-52 or Bus 1-62, and
- No fault has occurred on the section of cable between 15211 and 16211, and
- Breaker 15201 and/or 16201 is open.

Breakers 15111 and 16111 can be closed by operator action only. The operator can close BKR 15111 or BKR 16111 only if:

- No fault has occurred on either Bus 1-51 or 1-61, and
- No fault has occurred on the section of cable between 15111 and 16111, and
- Breaker 15101 and/or 16101 is open.

Using 4160-V Bus 1-5 and assuming the loss of off-site power, the following steps take place:

a. Start Diesel Generator 1A

b. Trip all 4160-V source breakers and the bus tie breaker (BKRs 501, 503, 509, 510 and 511);

c. Trip all 4160-V motor loads (BKRs 502, 504, 506, 507 and 508);

d. Trip selected 480-V loads (BKRs 15203, 15104, 15105, 15108, 15109 and 15212).

e. Close the diesel breaker (BKR 509) after the unit comes up to speed and voltage (maximum ten seconds from diesel engine start signal).

If there is a requirement for engineered safety features operation coincident with bus undervoltage, step "e" above is automatically followed by the sequential starting of the engineered safety feature equipment. A group of equipment is directly connected to the bus (see Table 8.2-1, Sequence 1.1, 1.2, 1.3 and 1.4) and the loads are picked up by the diesel generators immediately upon closing of the diesel breakers. This total load is minimal, < 330-kW. (Should the requirement for engineered safety features operation occur when voltage is present, the diesel generator is started and this same sequence is followed with the exception that the containment spray pump is allowed to start immediately if containment Hi-Hi pressure is present.). This loading sequence for Diesel Generator 1A is as follows (major loads only), continuing from step "e" (see Table 8.2-1 and Figure 8.2-7):

		Max. Time Lapse (Sec)
f.	(Step "0") Motor Operated Valves	0
g.	(Step "1") Start Safety Injection Pump 1A	7
h.	(Step "2") Start Residual Heat Removal Pump 1A	12
i.	(Step "3") Start Shield Bldg. Fan 1A, Start Zone SV equipment, Start Containment Spray Pump 1A if containment Hi-Hi pressure is present	20
j.	(Step "4") Start Service Water Pump 1A1	25
k.	(Step "5") Start Containment Fan Coil Units 1A/1B	30
l.	(Step "6") Start Auxiliary Feedwater Pump 1A	35
m.	(Step "7") Start Component Cooling Pump 1A	40
n.	(Step "8") Start Service Water Pump 1A2	45
o.	(Step "9") Manual or Auto start of any auxiliary as required for safe plant operation	53
p.	(Step "10") Manual or Auto start of any auxiliary as required for safe plant operation	63

NOTE: "Max Time Lapse" is the maximum time to initiation of closing the branch feeder breaker following the closure of the diesel generator source breaker. The "Max Time Lapse" specified above through step "n" is a maximum time that should not be exceeded. The "Max Time Lapse" specified above for step "o", as one input to service water isolation to the Turbine Building, is a maximum time that should not be exceeded. Because all other step "o" and "p" loads are defined above as those loads that will manually or automatically start as required, the "Max Time Lapse" specified above should not be considered absolute with nominal deviation of minor safety significance.

Starting of the containment spray pumps, initiated by Hi-Hi containment pressure, is accomplished simultaneously with any of the above steps following the starting of the residual heat removal pump when the diesel is required to supply power to the bus. When the bus is supplied from a transformer source the containment spray pump is started immediately on Hi-Hi containment pressure. The diesel generator automatic loading sequence through step 8, including 10 seconds for engine starting, will be accomplished in approximately fifty-five seconds as shown by Figure 8.2-7. As stated in Section 14.3.4, the containment pressure analysis assumes a delay of 137.7 and 85.3 seconds respectively, to supply design containment cooling from containment spray and fan-coil units.

The automatic sequences for Bus 1-6 and for Buses 1-61 and 1-62 associated with Diesel Generator 1B are similar to those described for Bus 1-5. Loads to be carried by a diesel generator are summarized in Table 8.2-1.

Should any of the feeder breakers, associated with the above (safety features) large (non-MCC feed) pump or fan motors, trip due to overcurrent, they can be re-closed from the control room. The electrical overload protection for the engineered safety feature fan, pump, and valve motors are not actually applied as overload protection. The motors are conservatively operated with respect to their rating and an overload occurs only as a major malfunction. Therefore, the overload protection isolates the malfunctioning component before it can make the bus breaker trip, causing loss of power to all other components in that circuit. Overload trip elements on the reversing starters associated with the various motor-operated valves and non-reversing starters associated with small pump or fan motors can and must be reset at the motor control centers. If the diesel generator is overloaded, an alarm is annunciated in the control room. The diesel generator is not protected by overload devices.

Load Evaluation

Diesel Generators

Each diesel generator is sized to start and carry the engineered safety features required for a post-blowdown containment pressure transient.

Selected generator nameplate data is as follows:

- Electro-Motive Division of General Motors Corporation
- Model A-20-C1, Serial Nos. 70-J1-1029 and 1039
- 2400/4160-V, 60 Hertz, Amps 782/452, 3 phase
- 3250-kVA, Temperature rise 85°C Stator-Thermometer
- Temperature rise 60°C, Rotor-Resistance
- 900 RPM, Power Factor 0.8
- 3575-kVA Peak, 2000 hours per year
- Temperature rise 105°C, Stator-Thermometer
- Temperature rise 70°C, Rotor-Resistance.
- Insulation Class, H-Stator and F-Rotor

Additional operating characteristics of the generator follow:

Capable of being started and ready to accept load in ten seconds and capable of being fully loaded within twenty seconds.

Capable of operating continuously at rated kVA output at any power factor between rated lagging and unity, at any voltage within ±5% of rated voltage.

Capable of tolerating for thirty seconds without injury a three-phase short circuit at its terminals when operating at rated kVA and power factor, 5% overvoltage and fixed excitation.

Compliance to Regulatory Guide 1.9

1. Sizing of generator power requirements.

Motors - All motors are the standard rating above the normal load. The service factor is added to the motor to cover fan and pump run-out. Checks have been made to assure the run-out is within the service factor. Motor power requirements were calculated as follows:

100 hp and larger, at 93% efficiency from manufacturers certified test data and handbooks. Less than 100 hp, at 88% efficiency from handbook data.

Brake hp was used for larger motors where certified test data was available.

KVA - Loads for transformers, etc., were calculated at 80% power factor.

Heater-Loads were taken at rated kW.

2. Generating load ratings:

Continuous	3250 kVA	100.0%
Continuous	2600 kW at 0.8 P.F.	100.0%
Overload, 2000 hours per year	2860 kW at 0.8 P.F.	110.0%
Overload, 7 days per year	2950 kW at 0.8 P.F.	113.5%
Overload, 30 minutes per year	3050 kW at 0.8 P.F.	117.3%

3. Generator rating criteria:

Regulatory Guide 1.9, dated December 1979 states that:

"Conformance with the requirements of IEEE Std 387-1977, "IEEE Standard criteria for Diesel Generator Units applied as Standby Power Supplies for Nuclear Power Generating Stations", dated June 17, 1977 is acceptable for meeting the requirements of the principle design criteria and qualification testing of diesel generator units used as on-site electric power systems for nuclear power plants".

The IEEE Standard 387-1977 states that:

"5.2.3 Operation Application Rule (see 3.7.1 and 3.7.2). The diesel generator units may be utilized to the limit of their power capabilities as defined by the continuous and short time ratings".

"3.7.1 Continuous Rating - The electric power output capability that the diesel generator unit can maintain in the service environment for 8760 h of operation per (common) year with only scheduled outages for maintenance".

"3.7.2 Short Time Rating - The electric power output capability that the diesel generator unit can maintain in the service environment for 2 h in any 24 h period, without exceeding the manufacturer's design limits and without reducing the maintenance interval established for the continuous rating".

NOTE: "Operation at this rating does not limit the use of the diesel generator unit at its continuous rating".

Table 8.2-1 lists the diesel-generator loads and the times that they will sequence on if required. The maximum connected loads are 3701.4 kW for DG 1A and 3523.3 kW for DG 1B. Table 8.2-1 also gives a time dependent load list, which shows that the highest estimated loads are 2919.8 and 2899.1 kW for each respective diesel generator, which occurs from one to sixty minute into the loading sequence. After adding safeguard station service transformer loss loads of 25.5 kW and 20.4 kW the maximum diesel generator loads are 2945.3 kW for DG 1A and 2919.5 kW for DG 1B. These loads are both less than the seven-day per year overload rating of 2950 kW for the diesel generators.

Operation of the safeguard diesel generators at frequencies other than 60 hertz, as allowed by the governor speed setting, have been shown by calculation to be within the various generator ratings.

The diesel generator ratings given in Item 2 above do not match the Short Time Rating definition of IEEE Std. 387-1977, as they were determined before 1977. We do, however, meet the intent of the Standard in that the diesel generators do not exceed the defined (by the manufacturer) load ratings. Therefore, the requirements of Regulatory Guide 1.9 are met.

4. Generator loading and sequence on safety injection signal is shown in Figure 8.2-7. The time sequence is after the closing of the diesel generator breaker. The maximum allowable time lapse for the load to come on was originally specified by Westinghouse and is used to support transient analysis. The normal time to pick up the load is the automatic timer setting. Table 8.2-1 lists the specific loads.

The criteria used in determining the a-c loads assigned to the emergency buses were:

a. Those loads, which are essential to safety-related functions and which if the power source failed, could affect public health and safety.

b. Those loads which if the power source failed would cause severe economic loss or cause the plant to experience an extended outage.

Batteries and Battery Chargers

Each of the plant's four 125-V station batteries has been sized to carry the expected shutdown loads following a plant trip and a loss of all AC power for a period of eight hours without the battery terminal voltage falling below 105 V. The 250-V station battery has been sized to carry its loads following a plant trip and a loss of all AC power for a period of two hours.

The safeguard batteries (BRA101 and BRB101) are C and D Charter Power Type LCR-19 1304 AH (8 hour), 1054 AH (3 hour), 647 AH (1 hour), and 1234 AH (1 min.). Major loads, with their approximate operating times on each battery, are listed in Table 8.2-2. The non-safeguard batteries (BRC101 and BRD101) are Exide Corp. Type FTC-21 1680 AH (8 hour), 1236 AH (3 hour), 750 AH (1 hour), and 1260 AH (1 min.). The 250-V non-safeguard battery (BRE101) is a C and D Power Systems Type 2LCR-15, 700 AH (2 hour).

Each of the three safeguard battery chargers has been sized to recharge either of the above partially discharged safeguard batteries within twenty-four hours, while carrying its normal load. Partially discharged is defined as any condition between fully discharged (105-V) and nominal (125-V). Normal voltage when on charger is 129 to 135-V (2.19 to 2.29-V per cell).

The battery chargers are each supplied with a d-c ammeter to continuously indicate the charger's current output. Each battery charger is also supplied with a d-c voltmeter on the line side of the charger output circuit breaker. This voltmeter will indicate the charger or battery voltage whichever is higher. A d-c ammeter on each of the main load side buses of the batteries continually indicate the total load current on each battery train. On a monthly basis the specific gravities of the pilot cells are checked and recorded.

The actual charge stored in the batteries can be related to the monitored parameters in the following ways:

a. The batteries are of the lead calcium type and are floated at 132-V d-c. As long as the specific gravities of the cells are at least 1.200 and the d-c voltmeter reads 132-V then the battery is considered fully charged.

b. The battery chargers are each rated at 150-ampere d-c output and will supply the plant load under normal conditions. The current-limiting feature on these chargers is set at approximately 172.5 amperes. In a situation when the ammeter on the battery main bus reads above 0.0 amperes then the battery would be discharging as indicated on the main bus ammeter.

Since the battery and charger share the loads on the bus under the normal condition, the battery must have sufficient ampere-hour capacity to carry the total loads consisting of two classes as follows:

a. The momentary load, such as closing and tripping of switchgear, involves the one-minute rating of a battery, though the time duration of the operation is but a few cycles.

b. The continuous load usually involves the battery's three to eight-hour rating, although longer time periods are sometimes used. The load consists of indicating lamps, holding coils for relays and any other equipment continuously drawing current from the control bus.

Reliability Assurance

The electrical system equipment is arranged so that no single incident can inactivate enough engineered safety features equipment to jeopardize plant safety. The 4160-V equipment is supplied from 6 buses, the 480-V equipment from 11 buses.

All Class 1E electrical equipment complies with IEEE Standard 344-1971, Trial Use Guide for Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations.

Two separate off-site power sources serve the 4160-V buses supplying power to the engineered safety features equipment. One of these is from the 138-kV portion of the substation; the second is from the tertiary winding of the substation autotransformer via an underground 13.8-kV circuit to the plant.

Separation is maintained in both the 4160-V and the 480-V systems to allow the plant auxiliary equipment to be arranged electrically so that redundant items receive power from two different buses.

For example, one complement of engineered safety features equipment is supplied from Bus 1-5 (4160-V) and Buses 1-51 and 1-52 (480-V) while the other complement is supplied from Bus 1-6 (4160-V) and Buses 1-61 and 1-62 (480-V). The cable tray system for one complement is independent of the cable tray system for the other complement; there are no crossties. This design assures the separation and independence of the two systems.

One off-site source of power can supply sufficient power to run normal operating equipment. Any one of the four transmission lines can supply all the plant auxiliary power. A low-voltage station auxiliary transformer can supply all the auxiliary loads for the plant.

Each diesel generator has capacity enough to sequentially start and run a fully loaded set of engineered safety features equipment. These safety features can adequately cool the core for any loss-of-coolant incident, and maintain the containment pressure within the design value.

One battery charger is in service on each battery so that the batteries are always at full charge in anticipation of loss of a-c power. This insures that adequate d-c power is available for starting the diesel generators and for other emergency uses.

The physical barrier provided between the emergency diesel engine generator sets consists in part of a Class I reinforced concrete wall 18 inches thick and the remainder a reinforced concrete block wall 12 inches thick. The doors and ventilation exhaust louvers are all Underwriters' Laboratories construction. All other openings in the barrier are sealed with fire retardant materials to maintain fire separation of the two diesel generator units.

The only potential for an explosion in the diesel generator rooms exists within a diesel engine crankcase. The rooms have sufficient volume and are vented to preclude a pressure rise that would endanger the integrity of the room walls.

In the event of a service water line break in the area between diesel generator rooms some water leakage would occur into the diesel generator rooms. Leakage through the door into diesel generator room 1B would flow to the floor drain if enough water was present to overflow the curb. Leakage through the door and through the trench into diesel generator room 1A would flow to the trench drain. All other openings into the diesel generator rooms from the tunnel are at higher elevations.

Both diesel generator rooms have double doors appropriately strengthened to prevent possible flooding. The trench into diesel generator room 1A was plugged to restrict leakage into the room.

Water flowing from the hypothetical service water line break would return to the screenhouse along the floor of the tunnel. Water entering the screenhouse would drain to the circulating water pump elevation where approximately 382,000 gallons are required to flood to the 586-foot elevation. Maximum possible service water pump run-out for two pumps would be less than 20,000 gpm total. The operator has over nineteen minutes to respond to the low service water pressure alarm on one header, isolate that header from the auxiliary building and then trip the pumps.

The rupture of a service water line in an Emergency Diesel Generator Room could result in the loss of the generator or the Safeguards bus in that room. Administrative operation from the Control Room of Type I Service Water valving would isolate the break and if required, realign the Service Water supplies through the intact piping from the operating Service Water Pumps.

Surveillance Requirements

The monthly tests specified for the diesel generators will demonstrate their continued capability to start and carry rated load. The fuel supplies and starting circuits and controls are continuously monitored, and abnormal conditions in these systems would be indicated by an alarm without need for test startup (Reference 2).

The less frequent overall system test demonstrates that the emergency power system and the control system for the engineered safety features equipment function automatically in the event of loss of all other sources of a-c power, and that the diesel generators start automatically in the event of a loss-of-coolant accident. This test demonstrates proper tripping of motor feeder breakers, main supply and tie breakers on the affected bus, and sequential starting of essential equipment, to the extent possible, as well as the operability of the diesel generators.

The specified test frequencies provide reasonable assurance that any mechanical or electrical deficiency is detected and corrected before it can result in failure of one emergency power supply to respond when called upon to function.

Station batteries will deteriorate with time, but precipitous failure is extremely unlikely. The continuous and periodic surveillance performed on the batteries will demonstrate battery degradation long before a cell becomes unserviceable or fails.

If a battery cell has deteriorated, or if a connection is loose, the voltage under load will drop excessively, indicating need for replacement or maintenance.

8.2.4 STATION BLACKOUT

Introduction

On July 21, 1988 the Code of Federal Regulations 10 CFR Part 50, was amended to include a new Section 50.63 entitled, "Loss of All Alternating Current Power", (Station Blackout). The station blackout (SBO) rule requires that each light-water-cooled nuclear power plant be able to withstand and recover from an SBO of specified duration.

Station Blackout Duration

The Kewaunee SBO duration is 4 hours, based on a plant AC power design characteristic Group P1, an emergency AC (EAC) power configuration Group C, and a target Emergency Diesel Generator (EDG) reliability of 0.95 (see NRC SER in reference 4).

Alternate AC (AAC) Power Source

The existing Technical Support Center (TSC) diesel generator will be used as an AAC source. The TSC diesel generator is an independent, non-class 1E, 600 kW (1000 hr/year standby rating) diesel generator that provides emergency power to 480-V Bus 1-46 for TSC equipment. For SBO purposes, a connection can be made between this bus and the 480-V safety Bus 1-52. Normal isolation between the two buses is provided by a Class 1E breaker at Bus 1-52 and a non-class 1E breaker at Bus 1-46. For SBO, selected non-essential loads will be stripped from each of the two buses and the two breakers will close to provide power to essential loads on both buses. The total load to be powered within 1 hour following the onset of the SBO is calculated to be approximately 587 kW.

Condensate Inventory

The Technical Specifications (TS) provide for a minimum of 39,000 gallons of condensate inventory. This is sufficient for 4 hours of decay heat removal.

Class 1E Battery Capacity

Battery capacity calculations exist for the BRA101 and BRB101 Class 1E batteries and on the BRD101 non-class 1E battery. They are based on the IEEE-485 methodology and a duty cycle of 8 hours. The 8-hour duty cycle capability has been verified by test per the guidance of IEEE 450-1987. Based on the above, the batteries are adequate, with considerable margin for the required 4-hour SBO duration.

Compressed Air

The air-operated valves relied upon to cope with an SBO event for 4 hours will be equipped with DC powered solenoid valves and backup air (nitrogen) supplies.

Effects of Loss of Ventilation

Steady state heat-up analyses were performed to determine the effects of loss of ventilation in the battery rooms, control room, relay room, charging pump room, turbine driven auxiliary feedwater pump room, containment and steam generator power operated relief valve areas. The calculated steady state temperatures for these rooms are well below the temperature limits described in NUMARC 87-00, Section 2.7. Kewaunee has procedures and operator training to ensure opening of doors #45 and #48 to Battery Rooms 1A and 1B, respectively, and relay room cabinet doors within 30 minutes of the onset of an SBO event.

Containment Isolation

Table 5.2-2, "Containment Penetrations" in the Updated Safety Analysis Report has been reviewed in accordance with the guidelines described in NRC Regulatory Guide (RG) 1.155 to ensure that appropriate containment integrity will be provided during an SBO event.

Reactor Coolant Inventory

The TSC diesel generator will power one of two charging pumps each having a capability of supplying water at the rate of 60 gpm. This will provide makeup for a total of 50 gpm reactor coolant pumps seal leakage (25 gpm per pump) and 10 gpm reactor coolant system leakage (maximum allowed by the TS). The water supply for the pumps will be from the refueling water storage tank, which has a TS minimum of 272,500 gallons of water.

Procedures and Training

Operator actions will be required for an SBO event. In addition to the opening of selected room and cabinet doors, and the valve operations required to align the charging pump and AFW pump, a number of electrical breaker operations are required to shed loads not required during an SBO, and to align the TSC diesel generator to the required loads.

The physical location of the buses and MCCs involved are in close proximity to each other. The inability of an operator to open any single motor control center (MCC) breaker will not result in the TSC diesel generator exceeding its overload rating. In addition to proceduralized operator actions and training provided to individuals, the breakers which are required to be opened are also locally identified.

The actions required during an SBO event can be accomplished in the 1-hour time frame specified for the AAC source to power the SBO loads.

SBO Modifications

An air (nitrogen) supply has been provided for RCS inventory valve CVC-7 in order to provide control room control of the amount of charging flow to the reactor coolant loop versus the reactor coolant pump seals.

The steam generator power operated relief valves SD3A and SD3B have each been equipped with a DC solenoid valve and a backup air (nitrogen) supply. Additional lighting has been provided at MCC 1-52E (charging pump 1A) and outside of Battery Room A.

Quality Assurance and Technical Specifications

The QA type 2 classification is consistent with the requirements of RG 1.155, Section 3.5.

EDG Reliability Program

The EDG reliability program, including the TSC diesel generator, meets the intent and guidance provided in RG 1.155, Section 1.2.

REFERENCES - SECTION 8.2

1. NRC Safety Evaluation Report, M.B. Fairtile (NRC) to D.C. Hintz (WPS), Letter No. K-86-136 dated July 3, 1986

2. NRC Safety Evaluation Report, S.A. Varga (NRC) to E.R. Mathews (WPS), Letter No. K-81-189 dated November 12, 1981

3. NRC Safety Evaluation Report, A.T. Gody, Jr. (NRC) to K.H. Evers (WPS), Letter No. K-89-212 dated October 25, 1989

4. NRC Supplemental Safety Evaluation Report, A.G. Hansen (NRC) to C.A. Schrock (WPS), Letter No. K-92-215 dated November 19, 1992

5. NRC Safety Evaluation Reports:

   a. A. Schwencer (NRC) to E.W. James (WPS), Letter dated June 3, 1977
   b. R.B. Licciardo (NRC) to E.R. Matthews (WPS), Letter No. K-82-074 dated April 30, 1982
   c. J.D. Neighbors (NRC) to C.W. Giesler (WPS), Letter No. K-84-091 dated April 30, 1982
   d. R.J. Laufer (NRC) to C.A. Schrock (WPS), Letter No. K-93-203 dated September 30, 1993
   e. R.J. Laufer (NRC) to M.L. Marchi (WPS), Letter No. K-96-098 dated June 12, 1996

6. "Stability Studies Associated With Nuclear Power Plants In Wisconsin", E.R. Mathews, Wisconsin Public Service Corporation, October 11, 1968

9.2 CHEMICAL AND VOLUME CONTROL SYSTEM

The Chemical and Volume Control System:

a. adjusts the concentration of chemical neutron absorber for chemical reactivity control;

b. maintains the proper water inventory in the Reactor Coolant System;

c. provides the required seal water flow for the reactor coolant pump shaft seals;

d. processes reactor coolant letdown for reuse of boric acid;

e. maintains the proper concentration of corrosion inhibiting chemicals in the reactor coolant; and

f. keeps the reactor coolant fission product and corrosion product activities to within design levels.

The system is also used to fill and hydrostatically test the Reactor Coolant System.

During normal operation, therefore, this system has provisions for supplying:

a. Hydrogen to the volume control tank

b. Nitrogen as required for purging the volume control tank

c. Hydrazine or pH control chemical, as required, via the chemical mixing tank to the charging pumps suction.

9.2.1 DESIGN BASES

Redundancy of Reactivity Control

Criterion: Two independent reactivity control systems, preferably of different principles, shall be provided (GDC 27).

In addition to the reactivity control achieved by the Rod Control System described in Section 7, reactivity control is provided by the Chemical and Volume Control System which regulates the concentration of boric acid solution neutron absorber in the Reactor Coolant System. The system is designed to prevent uncontrolled or inadvertent reactivity changes, which might cause system parameters to exceed design limits.

Reactivity Hot Shutdown Capability

Criterion: The reactivity control system provided shall be capable of making and holding the core sub-critical from any hot standby or hot operating condition (GDC 28).

The reactivity control systems provided are capable of making and holding the core sub-critical from any hot standby or hot operating condition, including conditions resulting from power changes. The maximum excess reactivity expected for the core occurs for the cold, clean condition at the beginning of life of the core. The full-length RCC assemblies are divided into two categories comprising control and shutdown groups.

The control group used in combination with boric acid as a chemical shim provides control of the reactivity changes of the core throughout the life of the core at power conditions. This group of RCC assemblies is used to compensate for short term reactivity changes at power such as those produced due to variations in reactor power requirements or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion, fission product buildup and decay, and the xenon transient associated with power level changes.

Reactivity Shutdown Capability

Criterion: One of the reactivity control systems provided shall be capable of making the core sub-critical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure sub-criticality with the most reactive control rod fully withdrawn (GDC 29).

The reactor core, together with the reactor control and protection system is designed so that the minimum allowable DNBR is no less than 1.30 and there is no fuel melting during normal operation, including anticipated transients.

The shutdown groups are provided to supplement the control group of RCC assemblies to make the reactor at least one percent sub-critical ($k(eff) \leq 0.99$) following trip from any credible operating condition to the hot, zero power condition assuming the most reactive RCC assembly remains in the fully withdrawn position.

Sufficient shutdown capability is also provided to maintain the core sub-critical, with the most reactive rod assumed to be fully withdrawn, for the most severe anticipated cooldown transient associated with a single active failure, e.g., accidental opening of a steam bypass or relief valve. This is achieved with combination of control rods and automatic boron addition via the Safety Injection System.

Reactivity Hold-Down Capability

Criterion: The reactivity control systems provided shall be capable of making the core sub-critical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public (GDC 30).

Normal reactivity shutdown capability is provided by control rods with boric acid injection used for xenon transients and for plant cooldown. When the plant is at power, the quantity of boric acid available will exceed that quantity required to establish the cold shutdown boron concentration. This quantity will always exceed the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

The normal supply of boric acid is maintained in the boric acid storage tanks. The boric acid solution is transferred from the boric acid tanks to the suction of the charging pumps by either boric acid transfer pump. The charging pumps then inject the boric acid solution to the reactor coolant. Any charging pump and any boric acid transfer pump can be energized from diesel generator power. The Refueling Water Storage Tank (RWST) provides a backup supply of borated water that can be aligned to the suction of the charging pumps. The RWST normally has adequate quantity of borated water to place the plant in the cold shutdown condition if necessary. Either boric acid supply to any charging pump will provide sufficient negative reactivity to the reactor coolant to compensate for xenon decay.

Boric acid could be injected to shutdown the reactor independent of the RCC assemblies, which normally serve this function in the short term. The quantity of acid maintained to establish the cold shutdown boron concentration is more than that needed to initially shutdown the reactor without using RCC assemblies.

On the basis of the above, the injection of boric acid is shown to afford backup reactivity shutdown capability and provides adequate long term hold-down capability necessary to compensate for xenon transients and for plant cooldown.

Codes and Classifications

All pressure retaining components (or compartments of components) which are exposed to reactor coolant, comply with the following codes:

a. System pressure vessels - ASME Boiler and Pressure Vessel Code, Section III, Class C.

b. System valves, fittings, and piping - USAS B31.1, including nuclear code cases.

System component code requirements are tabulated in Table 9.2-1.

The tube and shell sides on the regenerative heat exchanger and the tube side of the excess letdown heat exchanger are designed to ASME Section III, Class C. This designation is based on the following considerations:

a. each exchanger is connected to the Reactor Coolant System by lines equal to or less than 2 inches; and

b. each is located inside the Reactor Containment Vessel.

Analyses show that the accident associated with a 2-inch line break does not result in clad damage or failure. Additionally, previously contaminated reactor coolant escaping from the Reactor Coolant System during such an accident is confined to the Reactor Containment Vessel and no public hazard results.

9.2.2 SYSTEM DESIGN AND OPERATION

The Chemical and Volume Control System, shown in Figures 9.2-1 through 9.2-5, provides a means for injection of the neutron control chemical in the form of boric acid solution, chemical additions for corrosion control, and reactor coolant cleanup, degasification and deboration. This system also adds makeup water to the Reactor Coolant System, reprocesses water letdown from the Reactor Coolant System, and provides seal water injection to the reactor coolant pump seals. Design seal injection to each reactor coolant pump is 8 gpm with 5 gpm leaking through the labyrinth seal into the reactor coolant system. The seal is designed to leak 3 gpm or less back to the Chemical and Volume Control System. Materials in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

System components whose design pressure and temperature are less than the Reactor Coolant System design limits are provided with overpressure protective devices.

System discharges from overpressure protective devices (safety valves) and system leakage are directed to closed systems. Effluents removed from such closed systems are monitored and discharged under controlled conditions.

During plant operation, reactor coolant flows through the letdown line from a loop cold-leg on the suction side of the reactor coolant pump and, after processing is returned to the cold-leg of the loop on the discharge side of the pump via a charging line and through the in leakage in the reactor coolant pump seals. An excess letdown line is also provided for removing coolant from the Reactor Coolant System.

Each of the connections to the Reactor Coolant System has an isolation valve located close to the loop piping. In addition, a check valve is located downstream of each charging line isolation valve. Reactor coolant entering the Chemical and Volume Control System flows through the shell side of the regenerative heat exchanger where its temperature is reduced. The coolant then flows through letdown orifices, which reduce the coolant pressure. The cooled, low-pressure water leaves the Reactor Containment Vessel and enters the Auxiliary Building where it undergoes a second temperature reduction in the tube side of the letdown heat exchanger followed by a second pressure reduction by the low-pressure letdown valve. After passing through the letdown filter and one of the mixed bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filters and enters the volume control tank through a spray nozzle.

Hydrogen is automatically supplied, as determined by pressure control, to the vapor space in the volume control tank, which is predominantly hydrogen and water vapor. The hydrogen within this tank is supplied to the reactor coolant for maintaining a low oxygen concentration. If required, fission gases can be removed from the system by venting the volume control tank to the Waste Disposal System.

From the volume control tank the coolant flows to the charging pumps which raise the pressure above that in the Reactor Coolant System. The coolant then enters the containment, passes through the tube side of the regenerative heat exchanger, and is returned to the Reactor Coolant System.

The cation bed demineralizer, located downstream of the mixed bed demineralizers, is used intermittently to control cesium activity in the coolant and also to remove excess lithium, which is formed from B^{10} (n, α) Li^7 reaction.

Boric acid is dissolved in hot water in the batching tank to a concentration of nominal 8.0% by weight. The lower portion of the batching tank is jacketed to permit heating of the batching tank solution with low-pressure steam. A transfer pump is used to transfer the batch to the boric acid tanks. Small quantities of boric acid solution are metered from the discharge of an operating transfer pump for blending with reactor makeup water as makeup for normal leakage or for increasing the reactor coolant boron concentration during normal operation. Electric heaters maintain the temperature of the boric acid tanks solution high enough to prevent precipitation. The boric acid piping is heat traced to prevent precipitation.

Excess liquid effluents containing boric acid flow from the Reactor Coolant System through the letdown line and are collected in the holdup tanks. As liquid enters the holdup tanks, the nitrogen cover gas is displaced to the gas decay tanks in the Waste Disposal System through the waste vent header. The concentration of boric acid in the holdup tanks varies throughout core life from the refueling concentration to essentially zero at the end of the core cycle. A recirculation pump is provided to transfer liquid from one holdup tank to another and to recirculate the contents of individual holdup tanks.

Liquid effluent in the holdup tanks is processed as a batch operation. This liquid is pumped through the evaporation feed ion exchangers, which primarily remove lithium hydroxide (Li7OH), and fission products such as long-lived cesium. It then flows through the ion exchanger filter and into the gas stripper/boric acid evaporator package.

The dissolved gases are removed from the liquid by the gas stripper section and are vented to the Waste Disposal System. The liquid effluent from the gas stripper section then enters the evaporator section. The distillate produced in the boric acid evaporator leaves the evaporator condenser and is pumped through a condensate cooler where the distillate is cooled to the operating temperature of the two-evaporator condensate demineralizers. If it is required, evaporator carry over is removed by one of the two-evaporator condensate demineralizers, and then the condensate flows through the condensate filter and accumulates in one of two monitor tanks. The dilute boric acid solution originally in the boric acid evaporator remains as the bottoms of the distillation process and is concentrated to nominal weight 8.0% boric acid.

Subsequent handling of the condensate is dependent on the results of sample analysis. Discharge from the monitor tanks is recycled through the evaporator condensate demineralizers, returned to the holdup tanks for reprocessing in the evaporator train or discharged to the environment with the condenser circulating water, when within the allowable activity concentration as discussed in Section 11. If the sample analysis of the monitor tank contents indicates that it may be discharged safely to the environment, at least two valves must be opened to provide a discharge path. As the effluent leaves, it is continuously monitored by the Waste Disposal System liquid effluent monitor. If an unexpected increase in radioactivity is sensed, one of the valves in the discharge line to the service water discharge header closes automatically and an alarm sounds in the control room.

Boric acid evaporator bottoms are discharged through a concentrate filter to the concentrates holding tank. Solution collected in the concentrates holding tank is sampled and then transferred to the boric acid tanks if analysis indicates that it meets specifications for use as boric acid makeup. Otherwise the solution is pumped to the holdup tanks for reprocessing by the evaporator train.

The concentrated solution can also be pumped from the evaporator to the Waste Disposal System to be placed in containers. These containers can then be stored at the plant site for ultimate shipment off-site for disposal.

The deborating demineralizers can be used intermittently to remove boron from the reactor coolant near the end of the core life. When the deborating demineralizers are in operation, the letdown stream passes from the mixed bed demineralizers, through a deborating demineralizer and into the volume control tank after passing through the reactor coolant filter.

During plant cooldown, when the residual heat removal loop is operating and the letdown orifices are not in service, a flow path is provided to remove corrosion impurities and fission products. A portion of the flow leaving the residual heat exchangers passes through the letdown heat exchanger, letdown filter, mixed bed demineralizers, reactor coolant filter, and volume control tank. The fluid is then pumped, via the charging pump, through the tube side of the regenerative heat exchanger into the Reactor Coolant System. Flow may also bypass the regenerative heat exchanger and letdown orifices, by way of a direct connection from the 1A RHR heat exchanger to the letdown heat exchanger.

Expected Operating Conditions

Tables 9.2-2 and 9.2-3 list the system performance requirements, and data for individual system components. Reactor coolant equilibrium activities are given in Appendix D.

Reactor Coolant Activity Concentration

The parameters used in the calculation of the reactor coolant fission product inventory, including pertinent information concerning the expected coolant cleanup flow rate and demineralizer effectiveness, are presented in Appendix D. In these calculations, 1% defects are assumed to be present in the fuel rods at initial core loading and are uniformly distributed throughout the core. The fission product escape rate coefficients are therefore based upon an average fuel temperature.

The fission product activity in the reactor coolant during operation with small cladding pinholes or cracks in 1% of the fuel rods is computed using the following differential equations:

For parent nuclides in the coolant,

$$\frac{dN_{w_i}}{dt} = D\upsilon_i N_{C_i} - \left(\lambda_i + R\eta_i + \frac{B'}{B_o - tB'}\right) N_{w_i}$$

For daughter nuclides in the coolant,

$$\frac{dN_{w_j}}{dt} = D\upsilon_j N_{C_j} - \left(\lambda_j + R\eta_j + \frac{B'}{B_o - tB'}\right) N_{w_j} + \lambda_i N_{w_i}$$

where:

N = population of nuclide
D = fraction of fuel rods having defective cladding
R = purification flow, coolant system volumes per sec.
B_o = initial boron concentration, ppm
B' = boron concentration reduction rate by feed and bleed, ppm per sec.
η = removal efficiency of purification cycle for nuclide
λ = radioactive decay constant
υ = escape rate coefficient for diffusion into coolant

Subscript C refers to core
Subscript w refers to coolant
Subscript i refers to parent nuclide
Subscript j refers to daughter nuclide

Tritium is produced in the reactor from ternary fission in the fuel, irradiation of boron in the burnable poison rods and irradiation of boron, lithium, and deuterium in the coolant. The deuterium contribution is less than 0.1 curie per year and may be neglected. The parameters used in the calculation of tritium production rate are also presented in Appendix D.

Reactor Makeup Control

The reactor makeup control consists of a group of instruments arranged to provide a manually pre-selected makeup water composition to the charging pump suction header or the volume control tank. The makeup control functions are to maintain desired operating fluid inventory in the volume control tank and to adjust reactor coolant boron concentration for reactivity and shim control.

Makeup for normal plant leakage is regulated by the reactor makeup control, which is set by the operator to blend water from the reactor makeup water tank with concentrated boric acid to match the reactor coolant boron concentration.

The makeup system also provides concentrated boric acid or reactor makeup water to either increase or decrease the boric acid concentration in the Reactor Coolant System. To maintain the reactor coolant volume constant, an equal amount of reactor coolant at existing reactor coolant boric acid concentration is letdown to the holdup tanks. Should the letdown line be out of service during operation, the excess letdown would be available which would aid in providing sufficient volume in the pressurizer to accept the amount of boric acid necessary for cold shutdown.

Makeup water to the Reactor Coolant System is provided by the Chemical and Volume Control System from the following sources:

a. The reactor makeup water tanks, which provide water for dilution when the reactor coolant boron concentration is to be reduced.

b. The boric acid tanks, which supply concentrated boric acid solution when reactor coolant boron concentration is to be increased.

c. The refueling water storage tank, which supplies borated water for emergency makeup.

d. The chemical mixing tank, which is used to inject small quantities of solution when additions of hydrazine or pH control chemical are necessary.

The reactor makeup control is operated from the control room by manually pre-selecting the desired makeup composition to the charging pump suction header or the volume control tank in order to adjust the reactor coolant boron concentration for reactivity control. Makeup is provided to maintain the desired operating fluid inventory in the Reactor Coolant System. The operator can stop the makeup operation at any time in any operating mode. The reactor makeup water supply and boric acid transfer pumps are normally lined up for automatic operation as required by the makeup controller.

A portion of the high-pressure charging flow is injected into the reactor coolant pumps between the pump impeller and the shaft seal so that the seals and the lower radial bearing are not exposed to high temperature reactor coolant. The injection flow splits and part becomes the shaft seal leak-off flow and the remainder enters the Reactor Coolant System through a labyrinth seal on the pump shaft. The shaft seal leak-off flow cools the lower radial bearing, passes through the seals, is cooled in the seal water heat exchanger, filtered, and returned to the volume control tank.

Seal water inleakage to the Reactor Coolant System requires a continuous letdown of reactor coolant to maintain the desired inventory. In addition, bleed and feed of reactor coolant are required for removal of impurities and adjustment of boric acid in the reactor coolant.

Automatic Makeup

The "automatic makeup" mode of operation of the reactor makeup control provides boric acid solution preset by the operator to match the boron concentration in the Reactor Coolant System. The automatic makeup compensates for minor leakage of reactor coolant without causing significant changes in the coolant boron concentration.

Under normal plant operating conditions, the mode selector switch is set in the "Automatic Makeup" position. A preset low level signal (17%) from the volume control tank level controller causes the automatic makeup control action to open the makeup stop valve to the charging pump suction, modulate the concentrated boric acid control valve, open the reactor makeup water control valve and to switch the boric acid transfer pumps to high speed operation. The flow controllers then blend the makeup stream according to the preset concentration. Makeup addition to the charging pump suction header causes the water level in the volume control tank to rise. At a preset high level set-point (27%), the makeup is stopped; the reactor makeup water control valve closes, the concentrated boric acid control valve returns to its normal position (open), the makeup stop valve to charging pump suction closes, and the boric acid transfer pumps are returned to their previous mode of operation.

Dilution

The "dilute" mode of operation permits the addition of a pre-selected quantity of reactor makeup water at pre-selected flow rate to the Reactor Coolant System. The operator sets the mode selector switch to "dilute", the reactor makeup water flow controller set point to the desired flow rate, and the reactor makeup water batch integrator to the desired quantity. Upon manual start of the system the makeup stop valve to the volume control tank opens, and the reactor makeup water control valve opens. Makeup water is added to the volume control tank and then goes to the charging pump suction header. Excessive rise of the volume control tank water level is prevented by automatic actuation (by the tank level controller) of a three-way diversion valve, which routes the reactor coolant letdown flow to the holdup tanks. When the preset quantity of reactor makeup water has been added, the batch integrator causes the reactor makeup water control valve and makeup stop valve to close.

Alternate Dilute

The "alternate dilute" mode is similar to the dilute mode except the dilution water, after passing through the blender, splits and a portion flows directly to the charging pump suction and a portion flows into the volume control tank via the spray nozzle and then flows to the charging pump suction. The operator sets the mode selector switch to "alternate dilute", the reactor makeup water flow controller set point to the desired flow rate, the reactor makeup water batch integrator to the desired quantity, and actuates the makeup start. The start signal causes the makeup control to open the makeup stop valve to the volume control tank, the makeup stop valve to the charging pump suction header, and the reactor makeup control valve. Reactor makeup water is simultaneously added to the volume control tank and to the charging pump suction header. This mode is used for load follow and permits the dilution water to follow the initial xenon transient and simultaneously dilute the volume control tank. Excessive water level in the volume control tank is prevented by automatic actuation of the volume control tank level controller which routes the reactor coolant letdown flow to the holdup tanks. When the preset quantity of reactor makeup water has been added, the batch integrator causes the reactor makeup water control valve and the reactor makeup stop valves to close. This operation may be stopped manually by actuating the makeup stop.

Boration

The "borate" mode of operation permits the addition of a pre-selected quantity of concentrated boric acid solution at a pre-selected flow rate to the Reactor Coolant System. The operator sets the mode selector switch to "borate", the concentrated boric acid flow controller set point to the desired flow rate, and the concentrated boric acid batch integrater to the desired quantity. Upon manual start of the system the makeup stop valve to the charging pumps opens, the concentrated boric acid control valve modulates, the boric acid transfer pumps start, if not already running, and the concentrated boric acid is added to the charging pump suction header. The total quantity added in most cases is so small that it has only a minor effect on the volume control tank level. When the preset quantity of concentrated boric acid solution has been added, the batch integrater causes the boric acid control valve and the makeup stop valve to the charging pump suction to return to their normal position, and stops the Boric Acid Transfer Pumps.

The capability to add boron to the reactor coolant is such that it imposes no limitation on the rate of cooldown of the reactor upon shutdown. The maximum rates of boration and the equivalent coolant cooldown rates are given in Table 9.2-2. One set of values is given for the addition of boric acid from a boric acid tank with one transfer and one charging pump operating. The other set assumed the use of refueling water but with two-of-the-three charging pumps operating. The rates are based on full operating temperature and on the end of the core life when the moderator temperature coefficient is most negative.

By manual action of the operator, the boric acid transfer pumps can discharge directly to the charging pump suction and bypass the blender and volume control tank.

Alarm Functions

The reactor makeup control is provided with alarm functions to call attention to the following conditions:

- Deviation of reactor makeup water flow rate from the control set point by ±5 gpm.
- Deviation of concentrated boric acid flow rate from the control set point by ± 0.2 gpm.
- Low level (makeup initiation point) in the volume control tank if the level decreases to the low level makeup initiation set point.

Charging Pump Control

Three positive-displacement variable-speed-drive-charging pumps are used to supply charging flow to the Reactor Coolant System.

The speed of each pump can be controlled manually or automatically. During normal operation, charging pumps are operated as necessary to maintain inventory in the reactor coolant system. During load changes, the pressurizer level set point is varied automatically to compensate partially for the expansion or contraction of the reactor coolant associated with the T_{avg} changes. Automatic control of charging pump speed does not change rapidly with pressurizer level variations due to the reset action of the pressurizer level controller.

If the pressurizer level increases, the speed of the pump decreases; likewise, if the level decreases, the speed increases. If the charging pump on automatic control reaches the high speed limit, an alarm is actuated and a second pump is manually started. The speed of the second pump is manually regulated. If the speed of the charging pump on automatic control does not decrease and the second charging pump is operating at maximum speed, the third charging pump can be started and its speed manually regulated. If the speed of the charging pump on automatic control decreases to its minimum value, an alarm is actuated and the speed of the pumps on manual control is reduced.

A selector switch is used to choose one-of-two pressurizer water level channel signals for input to a single level controller, which in turn controls charging pump speed.

The charging pump cannot overpressurize the system because:

1. The spray system in the pressurizer can suppress maximum charging flow.

2. Assuming a spray failure (single failure), the power operated relief valves can handle more than the maximum charging flow (2 channels).

3. Assuming three failures (spray and two power operated relief valves), the code relief valves can handle more than the maximum charging flow.

In addition, there are many redundant and diverse alarms to bring the operator's attention to the situation (high pressurizer pressure, high pressurizer water level, pressurizer level deviation, etc.). There is also an alarm on reaching the high or low limits imposed on the automatic control signal to the pump controller. The operator would then go to manual control of the charging pumps.

To ensure that the charging pump flow is always sufficient to meet the RCP seal water requirements, the pump has a control stop which can be preset and does not permit pump flow lower than the specified minimum. This control stop is adjustable to permit higher minimum flow limits to be set if mechanical seal leakage increases during plant life.

Components

A summary of principal component data is given in Table 9.2-3.

Regenerative Heat Exchanger

The regenerative heat exchanger is designed to recover the heat from the letdown stream by re-heating the charging stream during normal operation. This exchanger also limits the temperature rise, which occurs at the letdown orifices during periods when letdown flow exceeds charging flow by a greater margin than at normal letdown conditions.

The letdown stream flows through the shell of the regenerative heat exchanger and the charging stream flows through the tubes to place the lower pressure requirements on the shell. The unit is a three shell, multiple tube pass heat exchanger made of austenitic stainless steel, and is of all-welded construction.

Letdown Orifices

One of the three letdown orifices controls flow of the letdown stream during normal operation and reduces the pressure to a value compatible with the letdown heat exchanger design. Either of two letdown orifices, each 40 gpm, is used to pass normal letdown flow. The third orifice, 80 gpm, is designed to be used for maximum purification flow at normal Reactor Coolant System operating pressure and can pass twice the normal letdown flow. The orifices are placed in and taken out of service by remote manual operation of their respective isolation valves. One or both of the standby orifices may be used in parallel with the normally operating orifice in order to bring the letdown flow up to normal when the Reactor Coolant System pressure driving force is below normal. This arrangement provides a full standby capacity for control of letdown flow. Each orifice consists of bored pipe made of austenitic stainless steel.

Letdown Heat Exchanger

The letdown heat exchanger cools the letdown stream to the operating temperature of the mixed bed demineralizers. Reactor coolant flows through the tube side of the exchanger while component-cooling water flows through the shell. The letdown stream outlet temperature is automatically controlled by a temperature control valve in the component cooling water outlet stream. The unit is a multiple-pass tube-and-shell heat exchanger. All surfaces in contact with the reactor coolant are austenitic stainless steel, and the shell is carbon steel. Since the maximum operating temperature for the components in question is considerably less than 200°F (maximum operating temperatures are 125°F for the letdown heat exchanger shell, 130°F for the charging pump, and 100°F for the monitor tank pump), a design temperature of 200°F provides adequate margin. An increase in design temperature to 250°F would not provide any significant changes in equipment design and would not affect reactor safety.

If a significant leak develops in the letdown heat exchanger, such that the charging system is unable to maintain pressurizer water level, the letdown line isolation valves near the reactor coolant loop will trip closed. The excess letdown path can then be placed in service while maintenance is performed on the letdown heat exchanger.

Mixed Bed Demineralizers

Two flushable mixed bed demineralizers maintain reactor coolant purity. A lithium-7 (or H+ form) cation resin and a hydroxyl form anion resin are initially charged into the demineralizers. Both forms of resin remove fission and corrosion products, and, in addition, the reactor coolant causes the anion resin to be converted to the borate form. The resin bed is designed to reduce the concentration of ionic isotopes in the purification stream.

Each demineralizer is sized to accommodate the normal letdown flow. One demineralizer serves as a standby unit for use if the operating demineralizer becomes exhausted during operation.

The demineralizer vessels are made of austenitic stainless steel, and are provided with suitable connections to facilitate resin replacement when required. The vessels are equipped with a resin retention screen. Each demineralizer has sufficient capacity, after operation for one core cycle with 1% defective fuel rods, to reduce the activity of the reactor coolant to refueling concentration.

Cation Bed Demineralizer

A flushable cation resin bed in the hydrogen form is located downstream of the mixed bed demineralizers and is used intermittently to control the concentration of lithium-7 which builds up in the coolant from the B^{10} (n, α) Li^7 reaction. The demineralizer also has sufficient capacity to maintain the cesium-137 concentration in the coolant below 1.0 μCi/cc with 1% defective fuel. The demineralizer would be used intermittently to control cesium.

The demineralizer is made of austenitic stainless steel and is provided with suitable connections to facilitate resin replacement when required. The vessel is equipped with a resin retention screen.

Deborating Demineralizers

When required, two anion demineralizers remove boric acid from the Reactor Coolant System fluid. The demineralizers are provided for use near the end of a core cycle, but can be used at any time. Hydroxyl-form ion-exchange resin is used to reduce Reactor Coolant System boron concentration. Facilities are provided for regeneration. When regeneration is no longer feasible the resin is flushed to the spent resin-shipping cask.

Each demineralizer can remove the quantity of boric acid that must be removed from the Reactor Coolant System to maintain full power operation near the end of core life without the use of the holdup tanks or evaporators.

Resin Fill Tank

The resin fill tank is used to charge fresh resin to the demineralizers. The line from the conical bottom of the tank is fitted with a dump valve and may be connected to any one of the demineralizer fill lines. The demineralized water and resin slurry can be sluiced into the demineralizer by opening the dump valve. The tank, designed to hold approximately one-third the resin volume of one mixed bed demineralizer, is made of austenitic stainless steel.

Reactor Coolant Filter

This filter collects resin fines and particulates larger than 25 microns from the letdown stream. The vessel is made of austenitic stainless steel, and is provided with connections for draining and venting. Design flow capacity of the filter is equal to the maximum purification flow rate. Disposable synthetic bag filter elements are used.

Volume Control Tank

The volume control tank collects the excess water, released from zero power to full power that is not accommodated by the pressurizer. It also receives the excess coolant release caused by the deadband in the reactor control temperature instrumentation. Overpressure of hydrogen gas is maintained in the volume control tank to control the hydrogen concentration in the reactor coolant at 25 to 50 cc per kg of water (standard conditions).

A spray nozzle is located inside the tank on the VCT inlet line coming from the reactor coolant filter. This spray nozzle provides intimate contact to equilibrate the gas and liquid phases. A remotely operated vent valve discharging to the Waste Disposal System permits removal of gaseous fission products which are stripped from the reactor coolant and collected in this tank. Post-accident the volume control tank vents gaseous fission products to containment. The volume control tank also acts as a head tank for the charging pumps and a reservoir for the leakage from the reactor coolant pump controlled-leakage seal. The tank is constructed of austenitic stainless steel.

Charging Pumps

Three charging pumps inject coolant into the Reactor Coolant System. The pumps are the variable-speed positive-displacement type, and all parts in contact with the reactor coolant are fabricated of austenitic stainless steel and other material of adequate corrosion resistance. Special low-chloride packing is used in the pump glands. These pumps have mechanical packing followed by a leakoff to collect reactor coolant before it can leak to the outside atmosphere. Pump leakage is piped to the RHR sump pump pit. The pump design precludes the possibility of lubricating oil contaminating the charging flow, and the integral discharge valves act as check valves.

Each charging pump has sufficient capacity (60 gpm) to compensate for normal letdown purification flow (40 gpm) and No. 1 seal leakage (normally 3 gpm/pump) and, thus, maintain the proper reactor coolant total inventory, temperature and pressure. Each pump is designed to provide rated flow against a pressure equal to the sum of the Reactor Coolant System maximum pressure (existing when the pressurizer power operated relief valve is operating) and the piping, valve and equipment pressure losses of the charging system at the design charging flows.

A suction stabilizer and pulsation dampener have been installed on the inlet and discharge piping of each of the three charging pumps. The installation significantly reduces vibration and stress levels on the charging pumps.

One of the three charging pumps can be used to hydrotest the Reactor Coolant System. A small motor can be directly coupled to the pump for hydrostatic test purposes. A design change removed the small motor, and disconnected the power and control cables; however, the hydrotest capability still exists.

Chemical Mixing Tank

The primary use of the chemical mixing tank is in the preparation of pH control chemical solutions and hydrazine for oxygen scavenging.

The capacity of the chemical mixing tank is determined by the quantity of 35% hydrazine solution necessary to increase the concentration in the reactor coolant by 10 ppm. This capacity is more than sufficient to prepare the solution of pH control chemical for the Reactor Coolant System. The chemical mixing tank is made of austenitic stainless steel.

The mixing tank is provided with an orifice located in the reactor makeup water inlet line to limit the flow rate through the tank as the solution is flushed to the charging pump suction. The orifice is designed to pass the tank volume within 2.5 minutes by the reactor makeup water pressure.

Demineralizer Letdown Pre-Filter

The filters, a normal and a bypass, collect particulates larger than 15 microns from the letdown stream before it enters the demineralizers. The vessels are made of austenitic stainless steel, and are provided with connections for draining and venting. Design flow capacity of each filter is equal to the maximum purification flow rate. Disposable filter elements are used.

Excess Letdown Heat Exchanger

The excess letdown heat exchanger cools reactor coolant letdown flow at a rate equal to the nominal injection rate through the reactor coolant pump labyrinth seal, if letdown through the normal letdown path is blocked. The unit is designed to reduce the letdown stream temperature from the cold-leg temperature to 195°F. The letdown stream flows through the tube side and component cooling water is circulated through the shell side. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel. All tube joints are welded. Flanges have been installed in the excess letdown line to the heat exchanger to permit tube bundle removal.

Seal Water Heat Exchanger

The seal water heat exchanger removes heat from the reactor coolant pump seal water returning to the volume control tank and reactor coolant discharge from the excess letdown heat exchanger. Reactor coolant flows through the tubes and component cooling water is circulated through the shell side. The tubes are welded to the tube sheet to prevent leakage in either direction, which would result in undesirable contamination of the reactor coolant or component cooling water. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

The unit is designed to cool the excess letdown flow and the seal water flow to the temperature normally maintained in the volume control tank if all the reactor coolant pump seals are leaking at the maximum design leakage rate.

Seal Water Filter

The filter collects particulates larger than 25 microns from the reactor coolant pump seal water return and from the excess letdown heat exchanger flow. The filter is designed to pass the sum of the excess letdown flow and the maximum design leakage from the reactor coolant pump controlled-leakage seals. The vessel is constructed of austenitic stainless steel and is provided with connections for draining and venting. Disposable synthetic filter cartridges are used.

Seal Water Injection Filters

Two filters are provided in parallel, each sized for the injection flow. They collect particulates larger than 5 microns from the water supplied to the reactor coolant pump seal. The vessel is constructed of stainless steel and the filter elements are disposable cartridges.

Boric Acid Filter

The boric acid filter collects particulates larger than 25 microns from the boric acid solution being pumped to the charging pump suction line. The filter is designed to pass the design flow of two boric acid transfer pumps operating simultaneously and at high speed. The vessel is constructed of austenitic stainless steel and the filter elements are disposable synthetic cartridges. Provisions are available for venting and draining the filter.

Boric Acid Tanks

The boric acid tank capacities are sized to store sufficient boric acid solution for refueling plus enough boric acid solution for cold shutdown shortly after full power operation is achieved. In addition, each tank has sufficient boric acid solution to achieve cold shutdown if the most reactive RCCA is not inserted.

The concentration of boric acid solution in storage is maintained between 7.5 and 8.5% by weight (13,000 to 15,000 ppm boron) at a temperature of at least 125°F. Periodic manual sampling is performed and corrective action is taken, if necessary, to ensure that these limits are maintained.

As a consequence, measured amounts of boric acid solution can be delivered to the reactor coolant to control the chemical poison concentration. The combination overflow and breather vent connection has a water loop seal to minimize vapor discharge during storage of the solution. The tanks are constructed of austenitic stainless steel.

Boric Acid Tank Heaters

Two 100% capacity electric immersion heaters located near the bottom of each boric acid tank are designed to maintain the temperature of the boric acid solution at nominal 135°F with an ambient air temperature of 40°F. Thus ensuring a temperature in excess of the solubility limit (for 8.5% weight boric acid solution, crystallization occurs at 105°F). The temperature is monitored and is alarmed (high and low temperature alarms) in the control room. The heaters are sheathed in austenitic stainless steel.

Batching Tank

The batching tank is sized to hold 6 days makeup supply of nominal 8.0% boric acid solution for the boric acid tank. The basis for makeup is reactor coolant leakage of ½ gpm at beginning of core life. The tank is also used for solution storage. A local sampling point is provided for verifying the solution concentration prior to transferring it to the boric acid tank.

A tank manway is provided, with a removable screen to prevent entry of foreign material. In addition, the tank is provided with an agitator to improve mixing during batching operation. The tank is constructed of austenitic stainless steel and is not used to handle radioactive substances. The tank is provided with a steam jacket for heating the boric acid solution to 135°F.

The source of heat for the steam-jacketed boric acid batching tank is the process steam.

The boric acid batching tank is not required under post-accident conditions, since the boric acid tanks or the refueling water storage tank are used to supply the borated water required for safe shutdown.

Boric Acid Transfer Pumps

Two centrifugal pumps with two-speed motors are used to circulate or transfer boric acid. The pumps circulate boric acid solution through the boric acid tanks and inject boric acid into the charging pump suction header.

Normally one pump is used for boric acid batching and transfer. Both pumps are used for boric acid injection. The design capacity of each pump is equal to the normal letdown flow rate. The design head at high speed is sufficient, considering line and valve losses, to deliver rated flow to the charging pump suction header when volume control tank pressure is at the maximum operating value (relief valve setting). All parts in contact with the solutions are austenitic stainless steel and other adequately corrosion-resistant materials.

The enclosures around the Boric Acid Transfer Pumps are heated to prevent boric acid solidification. Each pump is heated by two cartridge heaters installed in stainless steel blocks mounted to the base of the pump. The suction piping within the enclosure is heated by fin strip heaters located on the sides of the piping. The electrical configuration maintains train separation of the heaters for each pump.

The transfer pumps are operated either automatically or manually from the main control room. The reactor makeup control operates both pumps automatically at high speed when boric acid solution is required for makeup or boration.

Boric Acid Blender

The boric acid blender enhances thorough mixing of boric acid solution and reactor makeup water from the reactor makeup supply circuit. The blender consists of a conventional pipe elbow fitted with a perforated tube insert. All material is austenitic stainless steel. The blender decreases the pipe length required to homogenize the mixture.

Recycle Process

Holdup Tanks

Three holdup tanks contain radioactive liquid, which enters from the letdown line. The liquid is released from the Reactor Coolant System during startup, shutdown, and load changes and from boron dilution to compensate for burnup. The contents of one tank may be processed by the gas stripper/boric acid evaporator package while another tank is available as a standby. The total liquid storage capacity of the three holdup tanks is designed to meet the liquid dilution required to return to power from cold shutdown when the unit is at approximately 90% of core life. The tanks are constructed of austenitic stainless steel.

The tanks utilize the gas decay tanks, with nitrogen as a backup, to prevent oxygen contamination.

Holdup Tank Recirculation Pump

The holdup tank recirculation pump is a centrifugal type used to mix the contents of a holdup tank or transfer the contents of one holdup tank to another. The pump is sized to transfer the contents of one tank to another in less than two hours. The wetted surface of this pump is constructed of austenitic stainless steel.

Gas Stripper Feed Pumps

Two canned rotor centrifugal gas stripper feed pumps supply feed from a holdup tank to the gas stripper/evaporator train. The capacity of each pump is equal to twice the gas stripper/evaporator capacity. The non-operating pump is a standby and is available for operation in the event the operating pump malfunctions. These centrifugal pumps are constructed of austenitic stainless steel.

Evaporator Feed Ion Exchangers

Three flushable evaporator feed ion exchangers remove cation contamination from the holdup tank effluent. Experiments performed by Westinghouse indicate that the decontamination factor for cesium (see Appendix D) is conservative. One ion exchanger has sufficient capacity to supply the gas stripper/evaporator train. These ion exchangers may be operated in parallel or series. Each vessel is constructed of austenitic stainless steel and contains a resin retention screen.

Ion Exchanger Filters

These filters collect resin fines and particulates from the evaporator feed ion exchanger. The vessels are made of austenitic stainless steel and are provided with connections for draining and venting. Disposable synthetic filter cartridges are used. The design flow capacity is equal to the boric acid evaporator flow rate.

Boric Acid Evaporator/Gas Stripper Package

One boric acid evaporator/gas stripper package is provided to remove radioactive gases and concentrate boric acid for reuse in the reactor coolant system.

Liquid effluent from the holdup tanks is preheated and then passed through the gas stripper column where dissolved and entrained gases are removed. The feed stream leaving the column enters the evaporator where the water and boric acid is concentrated in the evaporator shell to a nominal 8.0% weight boric acid solution and then pumped out. All liberated noncondensible gases flow to the vent header.

The evaporator/gas stripper package consists of a feed pre-heater, vent condenser, stripping column, evaporator, absorption tower, evaporator condenser, distillate cooler and the following pumps: two evaporator concentrates pumps and two distillate pumps. The package also includes valves, piping and associated component and process instrumentation.

All evaporator/gas stripper package equipment in contact with the process fluid is constructed of austenitic stainless steel.

Evaporator Condensate Demineralizers

Two demineralizers remove the radioactive contaminants carried over with the evaporator condensate. When the resin is exhausted, it is flushed to the Waste Disposal System. Normally one demineralizer is used as needed, with one available as a standby. The demineralizer vessels are constructed of austenitic stainless steel.

Condensate Filter

A filter collects resin fines and particulates larger than 25 microns from the boric acid evaporator condensate stream. The required flow capacity of the filter is based on the boric acid evaporator flow rate. The vessel is made of austenitic stainless steel and is provided with a connection for draining and venting. Disposable synthetic filter elements are used. The design flow capacity of the filter is equal to the boric acid evaporator flow rate.

Monitor Tanks

The monitor tanks permit continuous operation of the evaporator train. When one tank is filled, the contents are analyzed and discharged to the Waste Disposal System. The monitor tanks can also be filled by water from the makeup water treatment system. The monitor tank capacity permits the continuous evaporator operation at 15 gpm and requires sampling and laboratory analysis three times per day. These tanks are constructed of austenitic stainless steel.

Monitor Tank Pumps

Two monitor tank pumps discharge water from the monitor tanks. Each pump is designed to empty a monitor tank in 1.5 hours. The pumps are constructed of austenitic stainless steel.

Reactor Makeup Water Tanks

Two reactor makeup-water tanks contain liquid supplied from the makeup water demineralizers. The monitor tanks, after sampling has shown the liquid to be of proper quality and acceptable radioactivity level, could be transferred to the reactor makeup tank, if desired. These stainless steel tanks serve as a source of Reactor Coolant System makeup water.

Reactor Makeup Water Pumps

Two reactor makeup water pumps serve as a supply source for the Reactor Coolant Makeup System. These austenitic stainless steel pumps take suction from the reactor makeup water tanks.

Concentrates Filter

A disposable synthetic cartridge type filter removes particulates larger than 25 microns from the evaporator concentrates. Design flow capacity of the filter is equal to the boric acid evaporator concentrates transfer pump capacity. The vessel is made of austenitic stainless steel.

Concentrates Holding Tank

The concentrates holding tank is sized to hold one batch of concentrates from operation of the evaporator. The tank is supplied with an electrical heater, which prevents boric acid precipitation and is constructed of austenitic stainless steel.

Concentrates Holding-Tank Transfer Pumps

Two holding-tank transfer pumps discharge boric acid to the boric acid tanks or to the holdup tanks for recycling.

Electrical Heat Tracing

Electrical heat tracing is installed under the insulation on all piping, valves, line-mounted instrumentation, and components normally containing concentrated boric acid solution. The heat tracing compensates for heat loss due to cooling and prevents boric acid precipitation. The heat tracing system was designed in accordance with the following criteria:

1. 100% redundant and separate heat tracing systems are provided, with the exceptions that the concentrates holding tank is equipped with a single immersion heater, and the boric acid batching tank is steam heated.

2. The heat tracing system is designed to maintain the fluid temperature between 125°F and 135°F with an ambient air temperature of 40°F.

3. Each redundant heat tracing system is supplied from a separate power source connected to the redundant emergency diesel generators.

4. Normally, only one heat tracing system is energized. Failure of the energized heat tracing system is annunciated in the control room. An automatic transfer of control from the primary to the standby heat tracing system is achieved by energizing the redundant system by different settings in the two separate control thermostats, such that on failure of the primary system, the standby system will pick-up automatically without reaching the low temperature alarm setting.

5. The boric acid tanks are equipped with individual means of heating by immersion heaters supplied, as in (3) above.

The lines and components of the Chemical and Volume Control System (CVCS) which are provided with heat tracing or heater enclosures are shown in Figures 9.2-1 through 9.2-5.

The boric acid tanks, boric acid batching tank, and the concentrates holding tank are provided with individual means of heating and need not be electrically heat traced.

Redundant electrical heat tracing is installed on all sections of the CVCS normally containing boric acid solution, to provide standby capacity if the operating section malfunctions. The power supply for the redundant lines of heat tracing is connected to the diesel-powered buses to ensure continuous operation during a prolonged outage of normal power supplies.

The combination of electrical heat tracing and insulation maintains the temperature of the piping and contents at 125°F to 135°F with an ambient air temperature of 40°F. Separate thermostatic controls are provided for each of the duplicate sets of heat tracing to maintain the temperature within the specified control band. A high/low alarm is provided in the control room to warn of failure to maintain the normal temperature control band for the piping and equipment containing concentrated boric acid solution. Transfer of control between the redundant heat tracing is an automatic operation. Any single failure of a heat tracing line, heater controller or alarm will not result in a reduction of temperature below the point where precipitation of concentrated boric acid might occur.

Valves

Valves for radioactive service that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection that discharges to the Waste Disposal System. Manual and motor operated valves larger than 2 inches for radioactive service, with fluids at temperatures above 212°F, also have an intermediate leakoff connection that discharges to the Waste Disposal System. Globe valves are installed with flow over the seats when such an arrangement reduces the possibility of leakage. Basic material of construction is stainless steel for all valves except the batching tank steam jacket valves, which are carbon steel.

Isolation valves are provided at all connections to the Reactor Coolant System. Lines with flow into the reactor containment also have check valves inside the containment to prevent reverse flow from the containment.

Relief valves are provided for lines and components that might be pressurized above design pressure by improper operation or component malfunction. Pressure relief for the tube side of the regenerative heat exchanger, is provided by the check valve, which bypasses the charging line isolation valve.

Piping

All CVCS piping handling radioactive liquid is austenitic stainless steel. All piping joints and connections are welded, except where flanges are required to facilitate equipment removal for maintenance.

9.2.3 SYSTEM DESIGN EVALUATION

Availability and Reliability

A high degree of functional reliability is assured in this system by providing standby components and by assuring fail-safe response to the most probable mode of failure. Special provisions include duplicate heat tracing (with alarm protection) of lines, valves, and components normally containing concentrated boric acid.

The system has three high-pressure charging pumps, each capable of supplying the normal reactor coolant pump seal and makeup flow.

The electrical equipment of the Chemical and Volume Control System is arranged so that multiple items receive their power from various 480-V buses as described in Section 8.

The two boric acid transfer pumps are powered from separate 480-V buses. One charging pump and one boric acid transfer pump are capable of meeting cold shutdown requirements shortly after full-power operation. In cases of loss of a-c power, any charging pump and boric acid transfer pump can be energized from diesel generator power.

Control of Tritium

The Chemical and Volume Control System is used to control the concentration of tritium in the Reactor Coolant System. Essentially all of the tritium is in the chemical combination with oxygen as a form of water. Therefore, any leakage of coolant to the containment atmosphere carries tritium in the same proportion as it exists in the coolant. Thus, the level of tritium in the containment atmosphere, when it is sealed from outside air ventilation, is a function of tritium level in the reactor coolant, the cooling water temperature at the cooling coils, which determines the dew point temperature of the air, and the presence of leakage other than reactor coolant as a source of moisture in the containment air.

There are two major considerations with regard to the presence of tritium:

a. Possible plant personnel hazard during access to the containment. Leakage of reactor coolant during operation with a closed containment causes an accumulation of tritium in the containment atmosphere. It is desirable to limit the accumulation to allow containment access.

b. Possible public hazard due to release of tritium to the plant environment.

Neither of these considerations is limiting in this plant.

The concentration of tritium in the reactor coolant is maintained at a level, which precludes personnel hazard during access to the containment. This is achieved by discharging the condensate from the boric acid recovery process via the plant circulating water discharge.

The uncertainties associated with estimating the amounts of tritium generated are discussed in Appendix D.

Periodic determinations of tritium concentrations will be made by liquid scintillation counting of condensed water vapor from the containment.

Leakage Prevention

Quality control of the material and the installation of the Chemical and Volume Control System valves and piping, which are designated for radioactive service, is provided in order to essentially eliminate leakage to the atmosphere. The components designated for radioactive service are provided with welded connections to prevent leakage to the atmosphere. However, flanged connections are provided in each charging pump suction and discharge, on each boric acid pump suction and discharge, on the relief valves inlet and outlet, on three-way valves, on the flow meters and elsewhere where necessary for maintenance.

The positive displacement charging pumps stuffing boxes are provided with leakoffs to collect reactor coolant before it can leak to the atmosphere. All valves which are larger than 2 inches and which are designated for radioactive service at an operating fluid temperature above 212°F are provided with a stuffing box and lantern leakoff connections. Leakage to the atmosphere is essentially zero for these valves. All control valves are either provided with stuffing box and leakoff connections or are totally enclosed. Leakage to the atmosphere is essentially zero for these valves.

Diaphragm valves are provided where the operating pressure and the operating temperature permit the use of these valves. Leakage to the atmosphere is essentially zero for these valves.

Incident Control

The letdown line and the reactor coolant pumps seal water return lines penetrate the reactor containment. The letdown line contains two air-operated valves inside the reactor containment upstream of the regenerative heat exchanger. Three parallel air-operated orifice block valves inside the reactor containment and an air-operated valve outside the reactor containment are automatically closed by the containment isolation signal.

The reactor coolant pump's seal water return line contains one motor-operated isolation valve outside and one inside the reactor containment which are automatically closed by the containment isolation signal.

The two seal-water injection lines to the reactor coolant pumps and the charging line are inflow lines penetrating the reactor containment. Each line contains two check valves inside the reactor containment to provide isolation of the reactor containment if a break occurs in these lines outside the reactor containment.

Malfunction Analysis

To evaluate system safety, failures or malfunctions were assumed concurrent with a loss-of-coolant accident and the consequences analyzed and presented in Table 9.2-4. As a result of this evaluation, it is concluded that proper consideration has been given to plant safety in the design of the system.

If a rupture were to take place between the reactor coolant loop and the first isolation valve or check valve, this incident would lead to an uncontrolled loss of reactor coolant. The analysis of loss-of-coolant accidents is discussed in Section 14.

Should a rupture occur in the Chemical and Volume Control System outside the containment, or at any point beyond the first check valve or remotely operated isolation valve, actuation of the valve would limit the release of coolant and assure continued functioning of the normal means of heat dissipation from the core. For the general case of rupture outside the containment, the largest source of radioactive fluid subject to release is the contents of the volume control tank. The consequences of such a release are considered in Section 14.

When the reactor is sub-critical; i.e., during cold or hot shutdown, refueling, and approach to criticality, the relative reactivity status (neutron source multiplication) is continuously monitored and indicated by fission chambers counters and count rate indicators. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate, is slow enough to give ample time to start a corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical. See Section 14 for a more complete discussion of boron dilution accidents.

At least three separate and independent flow paths are available for reactor coolant boration; i.e., the charging line or through the two reactor coolant pump labyrinths. The malfunction or failure of one component will not result in the inability to borate the Reactor Coolant System. An alternate flow path is always available for emergency boration of the reactor coolant. As backup to the boration system, the operator can align the refueling water storage tank outlet to the suction of the charging pumps.

Concentrated boric acid can be injected into the reactor coolant system by means of the charging pumps through two flow paths:

a) normal charging line (30 gpm), or

b) seal water supply lines to the two reactor coolant pumps while bypassing seal injection filters (8 gpm per pump 3 gpm of which leaks back into the CVCS and 5 gpm of which passes into the RCS).

Each flow path is provided with a flow indicator.

Suction to the charging pumps can be delivered through three flow paths:

a) the blender and flow control valve,

b) a local manual valve path, or

c) an emergency boration path through the motor operated valve.

Each flow path is provided with a flow meter.

A letdown or charging line break would be indicated by excessive auto-makeup to the volume control tank. A break in the charging line or upstream of the letdown orifices in the letdown line would also result in an increase in charging pump speed and a possible hi-speed alarm, depending on the break size. If the break size is such that the charging/letdown system is unable to maintain pressurizer water level, the letdown line isolation valves near the reactor coolant loop would be tripped closed by a low level signal from the pressurizer level instrumentation (redundant valves and level channels are provided to insure letdown isolation in the event of a single active failure).

In the event that the letdown line must be isolated, except in an emergency, the charging line is also isolated to avoid thermal shocking of the charging line penetrations into the Reactor Coolant System. If the charging line must be isolated, the letdown line is also isolated to avoid flashing of the letdown stream as the pressure of the high temperature flow is reduced.

With the letdown and charging lines isolated, the Reactor Coolant System can be borated via the reactor coolant pump labyrinth seals by allowing the pressurizer water level to increase. If letdown of reactor coolant is necessary, the excess letdown line is capable of letting down a flow equivalent to the total labyrinth seal in-leakage from both reactor coolant pumps. As the system is cooled down, makeup water to maintain pressurizer level would be provided through the labyrinth seals. Therefore, the normal charging and letdown paths are not required to go to cold shutdown condition.

The minimum rate of injection of boric acid solution into the Reactor Coolant System is 10 gpm from the labyrinth seal leakage through each reactor coolant pump. With this injection rate and charging not in service, the time necessary to borate the system sufficiently for a cold shutdown is approximately five hours at EOL. Normal charging capability is 40 gpm (30 gpm through the charging line plus 10 gpm through the reactor coolant pump labyrinth seals), which can borate the reactor coolant system to the concentration necessary for a cold shutdown in approximately 1.5 hours. With the charging and letdown lines out of service, the length of time necessary to bring the reactor to cold shutdown conditions is increased by the three additional hours necessary to reach the appropriate boron concentration as discussed above.

Concentrated boric acid is normally injected into the Reactor Coolant System by means of the charging pumps which take suction from the boric acid tanks via the boric acid transfer pumps. Each operation is considered in turn:

1. Concentrated boric acid can be delivered to the suction of the charging pumps using the following paths:

    a. Through the blender and flow control valve. For this operation the operator may read the flow meter and the boric acid tank level indicators.

    b. Through a local manual valve path. For this operation the operator may read the flow meter and the boric acid tank level indicators.

    c. In the event that neither flow path "a" nor "b" is available, the operator would use the emergency boration path through the motor operated valve. For this emergency operation the operator may read the emergency path flow meter and the boric acid tank level.

2. The charging pumps can deliver concentrated boric acid into the Reactor Coolant System via the following paths:

    a. Normal charging line with flow meter.

    b. Seal water supply lines to the two reactor coolant pumps while bypassing the seal injection filters. If either path had to be used, local and Control Room flow indicators would indicate flow.

3. The Safety Injection pumps can also take a suction from the RWST and provide borated water to the RCS, when the RCS pressure is less than Safety Injection pump shutoff head. The quantity of boric acid stored in the RWST is sufficient to achieve Cold Shutdown at any time during core life.

On loss of seal injection water to the reactor coolant pump seals, seal water flow can be re-established by manually starting a standby-charging pump. Even if the seal water injection flow is not immediately re-established, the plant can continue to operate temporarily. The thermal barrier cooler has sufficient capacity to cool the reactor coolant flow which will pass through the thermal barrier cooler and seal package, as long as seal leakoff flow is > 2.5 gpm. If seal leakoff flow is ≤ 2.5 gpm when normal seal injection flow is lost, there will be one to two hours of slowly increasing temperatures before the pump's operating limits are reached.

With > 2.5 gpm seal leakoff flow from each pump's #1 seal, long term operation is possible, but in order to protect the reactor coolant pump seals from prolonged flow of unfiltered reactor coolant it is recommended that this condition be only temporary. The effect of continuous reactor coolant pump operation without injection water would be to possibly cause clogging of the #1 seal with the introduction of unfiltered water.

The thermal barrier-cooling coil is a complete backup to seal injection for cooling the reactor coolant pump bearings and seals and no overheating would result from continued operation without seal injection water, provided the pump's #1 seal leakoff flow rate is > 2.5 gpm.

Galvanic Corrosion

The only type of materials which are in contact with each other in borated water are stainless steels, Inconel, Stellite valve materials and Zircaloy fuel element cladding. These material have been shown (Reference 1) to exhibit only an insignificant degree of galvanic corrosion when coupled to each other.

For example, the galvanic corrosion of Inconel versus 304 stainless steel resulting from high temperature tests (575°F) in lithiated, boric acid solution was found to be less than 20.9 mg/dm5 for the test period of nine days. Further galvanic corrosion would be trivial since the cell currents at the conclusion of the tests were approaching polarization. Zircaloy versus 304 stainless steel was shown to polarize in 180°F lithiated, boric acid solution in less than eight days with a total galvanic attack of 3.0 mg/dm5. Stellite versus 304 stainless steel was polarized in seven days at 575°F in lithiated boric acid solution. The total galvanic corrosion for this couple was 0.97 mg/dm5.

As can be seen from the tests, the effects of galvanic corrosion are insignificant to systems containing borated water.

Fuel Element Failure Detection

Fuel element failure detection is achieved by monitoring the letdown flow, using channel R-9 of the Area Radiation Monitoring System (see Section 11.2). This channel consists of a fixed position, gamma sensitive GM tube detector. The radiation level is indicated locally outside the Letdown Heat Exchanger Room and remotely in the Control Room where it is recorded. A high radiation alarm is displayed on the radiation monitoring panels in the Control Room and locally. A remotely operated, long half-life radiation check source is provided. The source strength is sufficient to produce an approximately one decade above background meter indication. The range of the channel is 0.1 mr/hr to 100 r/hr.

Delay time for the monitor in detecting fuel element failure ranges from approximately one minute to approximately three minutes, depending on the letdown flow rate. At the maximum letdown flow rate (80 gpm), approximately 1.5 minutes will pass before the reactor coolant, contaminated by fission product release from the failed fuel element, will reach the monitor. At normal letdown rate (40 gpm), approximately three minutes will pass before the contaminated flow reaches the detection area.

The monitor will detect the release of failed fuel element fission products against a background of:

- N-16 source (assuming a sixty-second decay) -- 4.5E+1 mr/hr
- Nominal corrosion product sources -- 1.1E+1 mr/hr
- Previous fuel element defects -- Determined during operation

Fuel failure severity can be determined by relating any increase in radiation detected to a corresponding increase in either rod gap release of general fuel element defects:

- Percent fuel element defects -- 7.2E+1 mr/hr
- Rod gap release -- 3.2E+2 mr/hr

REFERENCES - SECTION 9.2

1. WCAP 1844, “The Galvanic Behavior of Materials in Reactor Coolants”, D. G. Sammarone, August 1961

9.0 AUXILIARY AND EMERGENCY SYSTEMS

The Auxiliary and Emergency Systems are supporting systems required to insure the safe operation or servicing of the Reactor Coolant System (described in Section 4).

In some cases the dependable operation of several systems is required to protect the Reactor Coolant System by controlling system conditions within specified operating limits. Certain systems are required to operate under emergency conditions.

The systems considered in this Section are:

- **Chemical and Volume Control System**

  This system provides for boric acid injection, chemical additions for corrosion control, reactor coolant cleanup and degasification, reactor coolant makeup, reprocessing of water letdown from the Reactor Coolant System, and reactor coolant pump seal water injection.

- **Auxiliary Coolant System**

  This system provides for transferring heat from reactor plant coolant to the Service Water System and consists of the following three systems:

  1. The Residual Heat Removal System removes the residual heat from the core and reduces the temperature of the Reactor Coolant System during the second phase of plant cooldown.

  2. The Spent Fuel Pool Cooling System removes the heat generated by spent fuel elements stored in the spent fuel pool.

  3. The Component Cooling System removes heat from the Reactor Coolant System, via the Residual Heat Removal System, during plant shutdown, cools the letdown flow to the Chemical and Volume Control System during power operation, and provides cooling to dissipate waste heat from various reactor plant components and the boric acid and waste evaporators.

- **Sampling System**

  This system provides the equipment necessary to obtain liquid and gaseous samples from the reactor plant systems.

- **Facility Service Systems**

  These systems include Fire Protection, Service Water, and Auxiliary Ventilation Systems.

- **Fuel Handling System**

  This system provides for handling fuel assemblies, Rod Cluster Control (RCC) assemblies and material irradiation specimens.

- **Equipment and System Decontamination Processes**

  These procedures provide for the decontamination of equipment, tools, and personnel.

9.1 GENERAL DESIGN CRITERIA

Criteria, which are specific to one of the auxiliary or emergency systems are listed and discussed in the appropriate system design basis subsection below. Criteria which apply primarily to other systems (and are discussed in other Sections) are also listed and cross-referenced below because details of closely related systems and equipment are given in this Section.

9.1.1 RELATED CRITERIA

Reactivity Control Systems Malfunction

Criterion: The Reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits (GDC 31).

As described in Section 7 and justified in Section 14, The Reactor Protection Systems are designed to limit reactivity transients to DNBR > 1.30 due to any single malfunction in the deboration controls.

Engineered Safety Features Performance Capability

Criterion: Engineered Safety Features such as the emergency core cooling system and the containment heat removal system shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public (GDC 41).

Each of the auxiliary cooling systems which serves an emergency function provides sufficient capability in the emergency mode to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the plant personnel and the public.

Containment Heat Removal Systems

Criterion: Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure this system shall perform its required function, assuming failure of any single active component (GDC 52).

Each of the auxiliary cooling systems, which serve as emergency function to prevent exceeding containment design pressure, provides sufficient capability in the emergency mode to accommodate any single failure of an active component and still perform its required function.

11.2 RADIATION PROTECTION

11.2.1 DESIGN BASIS

Monitoring Radioactivity Releases

Criterion: Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive (GDC 17).

The containment atmosphere, the Containment System vent, the Auxiliary Building vent, the Control Room Ventilation System, the spent fuel pool heat exchanger service water discharge, the RHR pump pit ventilation exhaust, the condenser air ejector exhaust, the containment fan-coil service water discharge, blowdown from the steam generators, the component cooling water, and the Waste Disposal System liquid effluent are monitored for radioactivity concentration during normal operations, anticipated transients, and accident conditions. High radiation in any of these is indicated and alarmed in the Control Room.

All gaseous effluent from possible sources of accidental radioactive release external to the Reactor Containment (e.g., the spent fuel pool and waste handling equipment) is exhausted from an Auxiliary Building vent, which is monitored. All accidental spills of liquids are contained within the Auxiliary Building and collected in a sump.

For any leakage from the Reactor Containment under accident conditions, the Shield Building Ventilation System provides dilution, holdup and filtration capability to minimize the dose contribution. The Auxiliary Building Special Ventilation System provides filtration capabilities for the containment leakage, which bypasses the Shield Building Ventilation System.

The Plant Radiation Monitoring System supplemented by portable survey equipment provides adequate monitoring of radioactivity release. An outline of the procedures and equipment to be used in the event of an accident is presented in Section 11.2.3. The environmental monitoring program is described in Section 2.8.

Monitoring Fuel and Waste Storage Areas

Criterion: Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels (GDC 18).

Monitoring and alarm instrumentation is provided for fuel and waste storage and handling areas to detect inadequate cooling and excessive radiation levels. Radiation monitors are provided to maintain surveillance over the release of radioactive gases and liquids, and the permanent record of activity releases is provided by radiochemical and analysis of known quantities of waste.

The Spent Fuel Pool Cooling System loop flow is monitored to ensure proper operation, as described in Section 9.

A controlled ventilation system removes gaseous radioactivity from the atmosphere of the fuel storage and waste treatment areas of the Auxiliary Building and discharges it to the atmosphere via the Auxiliary Building vent. Radiation monitors are in continuous service in these areas to actuate high radiation alarms, as described in Section 11.2.3.

Fuel and Waste Storage Radiation Shielding

Criterion: Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities (GDC 68).

Auxiliary shielding for the Waste Disposal System and its storage components are designed to limit radiation to levels not exceeding 1 mR/hr in normally occupied areas, to levels not exceeding 2.5 mR/hr in periodically occupied areas and to levels not exceeding 15 mR/hr in controlled occupancy areas.

Gamma radiation is continuously monitored in the Auxiliary Building. High level signals are alarmed locally and annunciated in the Control Room.

Protection Against Radioactivity Release from Spent Fuel and Waste Storage Areas

Criterion: Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity (GDC 69).

All waste handling and storage facilities are contained and equipment is designed so that accidental releases directly to the atmosphere are monitored and will not exceed the applicable guidelines, as discussed in Sections 11.1.2, 14.2.2 and 14.2.3.

11.2.2 PRIMARY AND SECONDARY SHIELDING

Design Basis

Radiation shielding is designed for operation at maximum radiation levels at the site boundary to below those levels allowed for continuous non-occupational exposure. The plant is capable of continued safe operation with 1% fuel element defects.

In addition, the shielding and containment provided ensure that in the unlikely event of a Design Basis Accident, the subsequent off-site radiation exposures will be below the guidelines of 10 CFR 50.67.

Sufficient shielding exists to limit essential equipment exposure and to provide adequate access to vital areas necessary to aid in the mitigation of, or recovery from, a Design Basis Accident. Plant shielding was inspected by a NRC representative and found to satisfy the requirements of Item II.B.2 of NUREG-0737 (see NRC SER in reference 1). Operating personnel at the plant are protected by adequate shielding, monitoring, and operating procedures. Each area in the plant is classed according to the dose rate allowable in the area, based on the expected frequency and duration of occupancy. All plant areas capable of personnel occupancy are classified as one of the five zones of radiation level listed in Table 11.2-1. The Radiation Control areas are shown in Figures 11.2-1 through 11.2-4. Typical Zone 0 areas are the Turbine Building and turbine plant service areas. Typical Zone I areas are the offices and Control Room. Zone II areas include the local control spaces in the Auxiliary Building and the operating floor of the Containment during reactor shutdown. Areas designated as Zone III include the sample room, valve galleries, fuel-handling areas, and intermittently occupied work areas. Typical Zone IV areas are the shielded equipment compartments enclosing the gas decay tanks and volume control tanks in the Auxiliary Building, waste-container storage area, and the Reactor Coolant loop compartments after shutdown.

All radiation and high radiation areas are appropriately marked and isolated in accordance with 10 CFR 20 and other applicable regulations.

The shielding is divided into four categories according to function.

These functions include the primary shielding, the secondary shielding, the fuel handling shielding, and auxiliary shielding.

Primary Shielding

The primary shielding is designed to:

- Reduce the neutron fluxes incident on the reactor vessel to limit the radiation-induced increase in transition temperature.
- Attenuate the neutron flux sufficiently to prevent excessive activation of plant components.
- Limit the gamma flux in the reactor vessel and the primary concrete shielding to avoid excessive temperature gradients or dehydration of the primary concrete shield.
- Reduce the residual radiation from the core, reactor internals and reactor vessel to levels which will permit access to the region between the primary and secondary shields after plant shutdown.
- Reduce the radiation leakage contribution to obtain optimum division of the shielding between the primary and secondary shields.

Secondary Shielding

The main function of the secondary shielding is to attenuate the radiation originating in the reactor and the reactor coolant. The major source in the reactor coolant is the Nitrogen-16 activity, which is produced by neutron activation of oxygen, during passage of the coolant through the core. The secondary shielding limits the full power radiation levels outside the containment building to less than 1 mR/hr. The secondary shield also ensures that the design radiation levels are not exceeded following a Design Basis Accident.

Fuel Handling Shielding

The fuel handling shielding permits the safe removal and transfer of spent fuel assemblies and rod cluster control assemblies (RCCAs) from the reactor vessel to the spent fuel pool. It is designed to attenuate radiation from spent fuel, RCCAs, and reactor vessel internals to less than 2.5 mR/hr at the refueling cavity water surface and less than 1.0 mR/hr in the Auxiliary Building.

Auxiliary Shielding

The function of the shielding is to protect personnel working near various system components in the Chemical and Volume Control System, the Residual Heat Removal System, the Waste Disposal System and the Sampling System. The shielding provided for the Auxiliary Building is designed to limit radiation levels to less than 1 mR/hr in normally occupied areas, and at or below 2.5 mR/hr in periodically occupied areas. Additional shielding has been provided and equipment has been relocated to minimize exposures to personnel and equipment where necessary.

Shielding Design

Primary Shielding

The primary shielding consists of the reactor internals, the reactor vessel wall, and a concrete structure surrounding the reactor vessel.

The primary shielding immediately surrounding the reactor pressure vessel consists of a reinforced concrete structure extending from the base of the containment to a height of 69.0 feet. The lower portion of the shield is a minimum thickness of 7.0 feet of concrete and is an integral part of the main structural concrete support for the reactor vessel. It extends upward to the operating floor, forming a portion of the refueling cavity. This cavity is approximately rectangular in shape, and has concrete sidewalls, which are 5 feet 5 inches thick adjacent to areas in which fuel is transported.

The primary concrete shielding is air cooled to prevent overheating and dehydration from the heat generated by radiation absorption in the concrete. Eight "windows" have been provided in the primary shield for insertion of the out-of-core nuclear instrumentation. Cooling for the primary shield concrete, nuclear instrumentation, and vessel supports is provided by circulating 12,000 cfm of containment air between the reactor vessel wall and the surrounding concrete structure. The primary shield neutron fluxes and design parameters are listed in Table 11.2-2.

Secondary Shielding

The secondary shield surrounds the Containment, the reactor coolant loops and the primary shield. It consists of interior walls within the Containment, the operating floor, and the Shield Building. The Shield Building also serves as the accident shield.

The total thickness of concrete provided for this function in the area above grade is 5 feet 8 inches. Of this, approximately 3 feet 2 inches is concrete wall inside the Containment immediately surrounding the reactor coolant loops and 2 feet 6 inches is in the Shield Building wall. This shielding reduces the radiation intensity at the outside surface of the Shield Building to a negligible level during normal plant operation.

The Shield Building consists of the 2 foot 6 inch reinforced concrete cylinder capped by a shallow, reinforced concrete dome 2 feet thick. Supplemental shielding has been provided for the Containment penetrations where required. Section 5 contains a detailed discussion of the Shield Building. The secondary shielding design parameters are listed in Table 11.2-3. The equipment access hatch is shielded by a 2 foot 6 inch thick concrete shadow shield. The control room is protected with concrete sidewalls 2 feet thick, and a concrete roof 2 feet thick.

The accident shielding design parameters are listed in Table 11.2-4.

Fuel Handling Shielding

The refueling cavity is formed by the upper portions of the primary shield concrete and other sidewalls of varying thicknesses. A portion of the cavity is used for storing the upper and lower internals packages. These are shielded with concrete walls 5 feet thick. The remaining walls vary from 4 feet to 6 feet thick, and provide the shielding required for handling spent fuel.

The refueling cavity, flooded with borated water to a height of 40 feet 2 inches during refueling operations, provides a temporary water shield above the components being withdrawn from the reactor pressure vessel. The water height during refueling is approximately 24 feet above the reactor pressure vessel flange. This height ensures that a minimum of 10 feet of water will be above the top of a withdrawn fuel assembly. Under these conditions, the radiation level is less than 50 mR/hr at the water surface.

The spent fuel assemblies and RCCAs are remotely removed from the reactor containment through the horizontal spent fuel transfer tube and placed in the spent fuel pool. Concrete, 5 feet thick, shields the spent fuel transfer tube. This shielding is designed to protect personnel from radiation during the time a spent fuel assembly is passing through the main concrete support of the Containment and the transfer tube.

Radial shielding, during fuel transfer is provided by the water and concrete walls of the fuel transfer canal. An equivalent of 6 feet of concrete is provided to insure a radiation level of 1.0 mR/hr in the Auxiliary Building areas adjacent to the spent fuel pool.

Spent fuel is stored in the spent fuel pool, which is located adjacent to the Containment. Radial shielding for the spent fuel is provided by 5-foot thick concrete walls. The pool is flooded with borated water to a level such that the normal water height above the stored fuel assemblies is approximately 25 feet.

The fuel handling shielding design parameters are listed in Table 11.2-5.

Auxiliary Shielding

The auxiliary shield consists of concrete walls around certain components and piping which process reactor coolant. In some cases, the concrete block walls are removable to allow personnel access to equipment during maintenance periods. Each equipment compartment is individually shielded so those compartments may be entered without having to shut down the adjacent system for any reason.

The primary shield material provided throughout the Auxiliary Building is concrete. The principal auxiliary shielding provided is tabulated in Table 11.2-6.

11.2.3 RADIATION MONITORING SYSTEM

The Radiation Monitoring System provides continuous radiological surveillance of plant system and working areas. The system performs the following basic functions:

- Warns operating personnel of radiological health hazards, such as abnormal radiation fields.
- Provides warning of plant malfunctions, which could lead to plant damage and/or radiological hazards.
- Prevents or minimizes inadvertent releases of radioactivity to the environment via automatic action capability.
- Provides monitoring of controlled radiological plant releases.

Radiation detection instruments are located in areas of the plant, which house equipment containing or processing radioactive fluid. These instruments continually detect, compute, and record operating radiation levels. If the radiation level should rise above the set point for any channel, an alarm is initiated in the Control Room or Radiation Protection Office. Some channels also alarm locally. In stipulated cases, the alarm signal also provides the necessary signal for automatic process controls (e.g., valve closure, damper isolation, etc.). The Radiation Monitoring System operates in conjunction with regular and special radiation surveys and with chemical and radiochemical analyses performed by the plant staff. Adequate information and warning is thereby provided for the continued safe operation of the plant and assurance that personnel exposure does not exceed the limits of 10 CFR 20.

Two high range detectors are located in the containment. These two-containment area monitoring channels are the only components in the Radiation Monitoring System designed to operate following a major loss-of-coolant accident.

The components of the Radiation Monitoring System are designed according to the following environmental conditions:

- Temperature - an ambient temperature range as specified in Table 11.2-8.
- Humidity - 0 to 95% relative humidity.
- Pressure - Components in the Auxiliary Building and Control Room are designed for normal atmospheric pressure. Area Monitoring System components inside the Containment are designed to withstand containment test pressure.
- Radiation - Process and area radiation monitors are of a non-saturating design so that they "peg" full scale if exposed to radiation levels at over full-scale intensities. Critical process monitors are located in areas where the normal and post-accident background radiation levels will not affect their usefulness.

The Radiation Monitoring System consists of two types of components, the Process monitors and the Area monitors.

- The Process Radiation Monitoring System refers to those radiation monitors capable of analyzing fluid (air or water) flow for indication of increasing radiation levels.
- The Area Radiation Monitoring System monitors the direct radiation in various areas of the plant or indication of increasing radiation levels.

The Environmental Radiation Monitoring Program, as described in the Radiological Environmental Monitoring Manual (REMM), monitors radiation in various areas surrounding the plant. (This is described in Section 2.8.)

Main Process Radiation Monitoring System

The Main Process Radiation Monitoring System is designed to provide information to plant personnel on:

- Radioactivity levels present in fluid (air and water) systems.
- Leakage across boundaries of closed systems.
- Radioactivity concentrations in liquid and gaseous flow paths that lead to release from the plant.

In conjunction with the design functions spelled out above, the system is capable of initiating automatic actions designed to prevent or minimize any inadvertent/uncontrolled release of radioactivity to the environment.

The Main Process Monitoring System consists of 13 channels of monitoring equipment, 9 of, which are equipped with some level of automatic action upon receipt of a high radiation alarm. Seven of the 13 channels perform engineered safety related functions. The Main Process Monitoring System consists of the following:

Process Monitors

- R-11 Containment System vent (Air Particulate)
- R-12 Containment System vent (Radioactivity Gas)
- R-13 Auxiliary Building vent A
- R-14 Auxiliary Building vent B
- R-15 Condenser air ejector
- R-16 Containment fan coil water
- R-17 Component Cooling System (Liquid Effluent)
- R-18 Waste Disposal System (Liquid Effluent)
- R-19 Steam gen. blowdown (Liquid Sample)
- R-20 Service Water System
- R-21 Containment System vent (Activity)
- R-22 Residual heat removal pump pit
- R-23 Control Room vent

The channels are capable of operational verification via the use of check sources that are either incorporated into the detector housing or externally mounted. In addition, there are alarms provided for channel failure and high radiation conditions. The alarms are indicated both at the individual radiation monitor meter panel and on the main annunciator panel, as well as via the annunciator audible alarm. The radiation level is indicated by a meter and can be recorded on multipoint recorders. Tables 11.2-7 and 11.2-8 contain the channel data pertinent to each detector.

Certain process radiation monitors also utilize flow alarms, which alert the operator or technician to an abnormal flow situation in the process stream. Non-interruptible power for the system is provided via inverters off the 125 V d-c supply.

Each channel contains a completely integrated modular assembly, which includes the following:

a. Level Amplifier

   Amplifies and discriminates the detector output pulse to provide a discriminated and shaped pulse output to the log level amplifier.

b. Log Level Amplifier

   Accepts the shaped pulse of the level amplifier output, performs a log integration (converts total pulse rate to a logarithmic analog signal). The analog signal is then converted to a digital signal that provides output for suitable indication and recording.

c. Power Supplies

Three separate duel power supplies are used to power the control room equipment 1) Area monitor, 2) Train A Process Monitors and 3) Train B Process Monitors. The field equipment is powered by its associated control room supply.

d. Test-Calibration Circuitry

These circuits provide a predetermined value that pulses an LED into the NAI crystal that generates a signal to perform a channel test, and a solenoid-operated radiation check source to verify the channel's operation.

e. Radiation Level Meter

This meter, mounted on the assembly drawer, has a scale calibrated logarithmically in counts per minute in the ranges from 1E+1 to 1E+7.

f. Indicating Lights/Annunciators

These lights indicate high, alert and normal radiation alarm levels and a channel failure. An annunciator on the main control board is actuated on high and alert radiation signals and check source active, channel failure, high/low flow (process skids) and power supply failure.

g. Bistable Circuits

Several bistable circuits are provided, to alarm on high and alert radiation (actuation point may be set at any level over the range of the instruments), check source active, high/low flow and channel failure.

h. A remotely operated long-half-life radiation check source is furnished in each channel. The energy emissions are similar to the radiation energies being monitored. The source strength is sufficient to indicate approximately one decade above background.

The Process Radiation Monitoring System consists of the following radiation monitoring channels:

Containment or Containment System Vent - Air Particulate Monitor (R-11)

This monitor is provided to measure air particulate gamma radioactivity in the Containment with the ability to alternately monitor the Containment purge exhaust. This application ensures that the release rate during purging is maintained below specified limits.

High radiation level for the channels initiates closure of the Containment purge supply and exhaust duct valves.

This monitor has a measuring range of 1.06E-12 to 1.06E-6 μCi/cc.

This channel takes a continuous air sample from either the Containment atmosphere, or the purge exhaust. The sample is drawn from the Containment discharge ductwork through a closed, sealed system monitored by a scintillation counter - filter paper detector assembly. The filter paper collects particulate matter greater than 1 micron in size, on its constantly moving surface, and is viewed by a Beta Plastic-scintillation detector. In the case of the purge exhaust an isokinetic nozzle is used to obtain a representative sample. The sample is returned either to the Containment purge exhaust or the Auxiliary Building exhaust, after it passes through the series connected (R-12) gas monitor.

The detector assembly is in a completely enclosed housing. The pulse signal is transmitted to the Radiation Monitoring System panels in the Control Room. Lead shielding is provided to reduce the background level to where it does not interfere with the detector's sensitivity. The filter paper mechanism, an electro-mechanical assembly which controls the filter paper movement, is provided as an integral part of the detector unit.

Containment or Containment System Vent - Radiogas Monitor (R-12)

This monitor is provided to measure gaseous gamma radioactivity in the Containment, and to ensure that the radiation release rate during purging is maintained below specified limits. High gas radiation level initiates closure of the Containment purge supply and exhaust duct valves.

This monitor has a measuring range of 4.31E-7 to 4.31E-1 μCi/cc.

A continuous air sample is taken from the Containment atmosphere or the purge exhaust after it passes through the air particulate monitor, and draws the sample through a closed, sealed system to the gas monitor assembly. The sample is constantly mixed in the fixed, shielded volume, where it is counted by a scintillation detector sensitive to both gamma and beta activity. The sample is then returned to the Containment, the Containment System vent or the Auxiliary Building exhaust.

The detector assembly is in a completely enclosed housing mounted in a constant volume gas container. Lead shielding is provided to reduce the background level to a point where it does not interfere with the detector's sensitivity. The detector output is transmitted to the Radiation Monitoring System Panels in the Control Room.

The Containment air particulate and radioactivity gas monitors (R-11 and R-12) have assemblies common to both channels. They are described as follows:

a. The flow control assembly includes a pump unit and selector valves that provide a representative sample (or a "clean" sample) to the detector.

b. The pump unit consists of:
   - A pump to obtain the air sample.

- A flowmeter to indicate the flow rate.
- A mass flow controller control valve to provide flow adjustment.
- A flow alarm assembly to provide low and high flow alarm signals

c. Selector valves are provided to direct the desired sample to the detector for monitoring and to block flow when the channel is in maintenance or "purging" condition.

d. A pressure sensor is provided to protect the system from high-pressure transients. This unit automatically closes the inlet and outlet valves upon a high-pressure condition.

e. Detector purging is accomplished with a valve control arrangement whereby the normal sample flow is blocked and the detector purged with a "clean" sample. This facilitates detector calibration by establishing the background level and aids in verifying sample activity level.

f. The flow control panel in the Control Room radiation monitoring racks permits remote operation of the flow control assembly. By operating a sample selector switch on the control panel, either the Containment or the Containment System vent sample may be monitored.

Alarm lights are actuated by the following:

- Flow alarm assembly (low or high flow)
- The pressure sensor assembly (high pressure)
- The filter paper sensor (paper drive malfunction)
- The pump power control switch (pump motor on)

Containment System Vent Air Activity Monitor (R-21)

The Containment System vent air activity monitor is designed primarily as a backup to the R-11/R-12 detector systems. Should R-11/R-12 be out of service, R-21 allows Containment air sampling and radioactivity analysis to be performed. This monitor system continuously monitors the Containment vent for iodine, particulate, and gas activity. The detector system consists of a sample line and pump. The vent air is drawn through a fixed particulate filter, a charcoal filter, and an off-line gas monitor in series. The off-line gas monitor is a scintillation detector sensitive to both gamma and beta activity.

The sample tank design provides a cyclonic airflow around the detector axis to preclude stagnancy within the sensing volume. The detector output is transmitted to the Radiation Monitoring System panels in the Control Room.

The charcoal filter for this channel is for measuring the accumulation of iodine isotope activity. The particulate filter is for the collection of particulate activity such as Cs-134 and Cs-137. Analysis of these filters is accomplished in the Count Room. These filters additionally serve to protect the sensitive sample volume chamber of the radiogas detector.

A high radiation alarm provides a closure signal to the Containment purge supply and exhaust valves. This monitor has a measuring range of 4.31E-7 to 4.31E-1 μCi/cc.

During routine fuel handling within Containment, ventilation air discharge passes through particulate filters. This discharge is monitored by two redundant Radiation Monitoring System channels (R12 and R21). An alarm by either of these channels will initiate containment ventilation isolation. Manual realignment is subsequently required to place the system back into operation utilizing the charcoal filters.

During these operations there is direct communication between the refueling group and the Control Room. If a fuel assembly were damaged during handling, the physical configuration of the duct system requires a 65-foot free air transport of the radioactivity to the duct entrance before release from Containment. This assures a reasonable interval between the time of initial release and discharge to the environment. In view of this, the most probable mechanism for Containment isolation would be manual activation by the Control Room upon request from the refueling group, rather than through the aforementioned automatic Containment Isolation System.

Considering the least probable case of containment isolation initiated by the Radiation Monitoring System, the large transport distance would preclude highly concentrated puffs. The response time for system isolation by radiation alarm is about four seconds, which includes the effects of signal delay. This limits the potential for a significant release before isolation.

Auxiliary Building Vent Monitors (R-13, R-14)

The Auxiliary Building vent monitors are used to monitor the Auxiliary Building vent flowpath on a continuous basis. The detectors are used to measure airborne radioactivity in the air as it is discharged out the stack. An off-line sampler is used to monitor and sample the Auxiliary Building vent stack. Upon receipt of a high radiation alarm, the system performs the following functions:

a. Shuts down normal Auxiliary Building ventilation.

b. Activates the Special Zone Auxiliary Building ventilation.

c. Initiates isolation of all normal ducting to the Auxiliary Building vent stack.

d. Closes the waste gas decay tank gas release valve.

e. Reroutes R11/12 sample exhaust flow from Auxiliary Bldg. vent to Containment.

f. Isolates the 2" containment depressurization line and stops the 2" containment supply blower.

g. Automatically diverts the Spent Fuel Pool Ventilation System exhaust through its charcoal filter banks.

h. Automatically isolates the Waste Gas Analyzer via redundant isolation valves MG(R)-560, MG(R)-561, MG(R)-562 and MG(R)-563.

An impedance matching circuit is used to match the signal source to the coaxial cable and transmit an alarm signal to the Control Room, where it is recorded on multipoint recorders and alarms on high activity.

Additionally, a sampler consisting of an isokinetic nozzle inserted in the vent to provide a representative sample to series connected particulate and charcoal collection filters is included. The charcoal filter is used for the accumulation of iodine isotope activity. The particulate filter is used for the collection of particulate activity such as Cs-134 and Cs-137.

Remote indication and annunciation are provided on the Waste Disposal System control board. A high level alarm shuts down normal ventilation fans, except Spent Fuel Pool Ventilation System fans; activates the Auxiliary Building Special Ventilation System; diverts exhaust from the Spent Fuel Pool Ventilation System through charcoal filter banks; initiates isolation of all ducting to the Auxiliary Building vent, with the exception of the Special Zone Ventilation discharge and the Spent Fuel Pool Ventilation discharge; and closes the gas release valve of the gas decay tanks.

This monitor has a measuring range of 4.31E-7 to 4.31E-1 μCi/cc.

Condenser Air Ejector Gas Monitor (R-15)

The condenser air ejector gas monitor is used to monitor the condenser air ejector discharge flow path on a continuous basis. The condenser air ejector discharges to the Auxiliary Building vent stack for HEPA filtration prior to release. The detector consists of a gamma scintillation tube placed in-line in the discharge flow path. The detector is designed to measure the gamma activity of the non-condensable gases removed by the air ejector. A high alarm on R-15 initiates automatic action to ensure that the three-way control valve located in the air ejector discharge header is aligned to the Auxiliary Building vent stack. Since the plant is normally run in this mode, a high alarm will not normally affect this valve. The high alarm also initiates closure of the steam generator blowdown sample isolation valves and the steam generator blowdown isolation valves.

A gamma sensitive scintillation detector is inserted into a drywell in an in-line fixed volume container, which includes adequate shielding to reduce the background radiation to where it does not interfere with the detector's maximum sensitivity. This monitor has a measuring range of 2E-7 to 5E-3 μCi/cc.

Control Room Ventilation Monitor (R-23)

This channel continuously monitors the Control Room environment for an indication of airborne activity entering through the ventilation system. The detector is a beta-sensitive plastic scintillator that is mounted in the air supply duct.

Readout is in the Control Room on multipoint recorders and at a rate meter station with high-low alarm setting. High alarm circuits actuate the necessary dampers and fans to isolate the Control Room environment and recirculate the air through a PAC filtering system. Redundant air intakes are provided to allow for any makeup requirement.

This monitor has a measuring range of 4.31E-7 to 4.31E-1 μCi/cc.

Residual Heat Removal Pump Pit Monitor (R-22)

This monitor provides continuous monitoring of the exhaust air from the RHR pump pits for indication of pump leakage while circulating liquid containing high gaseous activity.

The detector is a beta-sensitive plastic scintillator mounted directly in an off-line sampler. The sample volume chamber design provides a cyclonic airflow through the detector to preclude stagnation within the sensing volume. The detector output is transmitted to the Radiation Monitoring System panels in the Control Room. The detector has a measuring range of 4.31E-7 to 4.31E-1 μCi/cc.

A high radiation alarm alerts the operator so that a timely transfer to the standby pump may be made. No automatic action is initiated by a high alarm, but operator action is assumed to occur if an RHR pump is leaking.

Containment Fan Coil Water Monitor (R-16)

The containment fan coil unit water monitor is designed to monitor the fan coil cooling water (service water) discharge for gamma activity. The fan coil cooling water discharge returns to the lake via the Circulating Water System. The detector system consists of a gamma-sensitive scintillation detector mounted in an in-line sample chamber. The sample chamber receives a continuous sample from the fan coil discharge headers. One sample line is connected to the discharge header from the 1A and 1B fan coil units, with a similar arrangement for the 1C and 1D units. A high alarm requires operator action to sequentially isolate each of the four fan coil units in order to determine the faulted unit.

The range of this monitor is 4E-8 to 5E-2 μCi/ml.

Component Cooling System Liquid Monitor (R-17)

This channel continuously monitors the Component Cooling System for radiation indicative of a leak of reactor coolant from the Reactor Coolant System and/or the Residual Heat Removal

System to the Auxiliary Coolant System. The detector system consists of an in-line T-type monitor located in the pumps discharge header downstream of the 1A and 1B component cooling water heat exchangers.

The range of this monitor is 4E-8 to 5E-2 μCi/ml.

Waste Disposal System Liquid Effluent Monitor (R-18)

This channel continuously monitors all Waste Disposal System liquid releases from the plant. Automatic valve closure action is initiated by this monitor to prevent further release after a high radiation level is indicated and alarmed. The detector assembly consists of a scintillation detector located in-line in the discharge header of the 1A and 1B waste condensate pumps. Remote indication and annunciation are provided on the Waste Disposal System local control board and in the Control Room.

The range of this monitor is 4E-8 to 5E-2 μCi/ml.

Steam Generator Blowdown System Liquid Sample Monitor (R-19)

This channel monitors the liquid phase of the secondary side of the steam generator for radiation, which would indicate a primary-to-secondary system leak, providing backup information to the condenser air ejector gas monitor. Samples from the bottom of each of the two steam generators are mixed in a common header and the common sample is continuously monitored by a scintillation counter and sample volume chamber assembly. Upon indication of a high radiation level, each steam generator is individually sampled in order to determine the source. This sequence is achieved by manually selecting the desired steam generator to be monitored.

The steam generator blowdown and air ejector radiation monitors are interconnected, such that either monitor isolates the blowdown and reroutes the air ejector exhaust.

A high radiation signal closes the isolation valves in the blowdown lines and sample lines.

The range of this monitor is 4E-8 to 5E-2 μCi/ml.

Service Water Monitor (R-20)

This channel continuously monitors the service water return path from the spent fuel pool and component cooling heat exchangers. A scintillation counter and sample volume chamber assembly monitors these effluent discharges. An increase above ambient radiation levels would indicate a leak across one of these heat exchangers and selective component isolation would then be used to locate the malfunction.

System readout is in the Control Room on multipoint recorders, a rate meter station in the Control Room with a high-low alarm setting, and a rate meter station at the detector location. A high level alarm is annunciated in the Control Room.

The range of this monitor is 4E-8 to 5E-2 μCi/ml.

In channels R-16, R-17, R-18, R-19 and R-20 a photo-multiplier tube scintillation crystal (Na-I) combination, mounted in a hermetically sealed unit, is used for liquid effluent radiation actuation. Lead shielding is provided to reduce the background level so it does not interfere with detector's sensitivity. The inline fixed volume container is an integral part of the detector unit. In channel R-17, the detector is inserted into a dry well in the Component Cooling System piping.

Auxiliary Process Radiation Monitoring System

The Auxiliary Process Monitoring System consists of two sets of components, which have been installed since plant construction. The motivating force behind many of the additional monitors was the result of lessons learned from TMI-2. These components have been installed for the purpose of an expanded, more reliable radiation monitoring capability (see NRC Order and NRC SERs in references 2, 3, and 4).

The first set of components is a group of eight high and low radiation monitors. These monitors are area-type monitors, which have been installed on the outside of process flow piping. A pair of monitors, one high range and one low range, are located on the outside of the containment vent ducting, the Auxiliary Building vent ducting, main steam 1A piping, and main steam 1B piping. Each detector has a local readout and Control Room readout, with recorders also located in the Control Room. These radiation monitors are as follows:

Channel	Area Monitor
R-31	Steam line 1A (low range)
R-32	Steam line 1A (hi range)
R-33	Steam line 1B (low range)
R-34	Steam line 1B (hi range)
R-35	Auxiliary building vent (low range)
R-36	Auxiliary building vent (hi range)
R-37	Containment building vent (low range)
R-38	Containment building vent (hi range)

The second set of components are a pair of SPING-4 monitors. The SPING-4 is a self-contained microprocessor-based radiation detection system used to monitor for particulate, iodine, and noble gas activity in the air. The two SPING-4 Process Monitors are installed off-line in the containment vent stack and the Auxiliary Building vent stack. All SPING system detectors are tied to the central control consoles for monitoring of each SPING-4 stack process monitor, which contains nine individual detectors. The two SPING-4 Process Monitors and the nine individual detectors are as follows:

Channel	Area Monitor
01	Auxiliary Building vent ducting (SPING-4)

Address Code	Detector
01-01	Beta particulate
01-02	Alpha particulate
01-03	Iodine
01-04	Iodine background
01-05	Lo-range gas
01-06	Gamma area monitor
01-07	Mid-range gas
01-08	Gas background
01-09	Hi-range gas

Channel	Area Monitor
02	Containment building vent ducting (SPING-4)

Address Code	Detector
02-01	Beta particulate
02-02	Alpha particulate
02-03	Iodine
02-04	Iodine background
02-05	Lo-range gas
02-06	Gamma area monitor
02-07	Mid-range gas
02-08	Gas background
02-09	Hi-range gas

Information from the SPING-4 process monitors is available at either of the central control consoles, which are located in the Radiation Protection Office and the Radiological Analysis Facility in the Technical Support Center.

Area Radiation Monitoring System

There are three area radiation monitoring systems providing continuous radiological surveillance of critical plant systems and work areas. Detectors are located in Containment, the Auxiliary Building and the Technical Support Center. The system provides operating personnel with early warning of certain plant malfunctions which might lead to a radiological health hazard or plant damage.

The General Area Radiation Monitoring System consists of eight channels, which monitor radiation levels to various areas of the plant. These are as follows:

Channel	Area Monitor
R-1	Control Room
R-2	Containment (by personnel hatch)
R-4	Charging pump room

Channel	Area Monitor
R-5	Fuel handling area
R-6	Sampling room
R-7	In-Core instrument seal table area
R-9	Reactor coolant letdown line
R-10	New fuel pit (criticality monitor)

Each channel consists of a fixed position gamma sensitive GM tube or ion chamber (R-5, R-10) detector. The detector output is amplified and the log count rate is determined by the integral amplifier at the detector. The radiation level is indicated locally at the detector, and in the Control Room where it is recorded.

High radiation alarms are displayed on the radiation monitoring panels in the Control Room, and at the detector location. The Control Room annunciator provides a single window, which alarms for any channel detecting high radiation. Verification of which channel has alarmed is made at the Radiation Monitoring System panels. A remotely operated, long half-life radiation check source is provided in each channel. The source strength is sufficient to produce an approximately one decade above background meter indication. Tables 11.2-7 and 11.2-8 contain the channel data pertinent to each detector.

The computer-indicator module amplifies the radiation level signal, as computed by the low-level amplifier, for indication and recording. The module also provides controls for actuation of the channel check source.

A meter is mounted on the front of each computer indicator module and is calibrated logarithmically from 0.1 mR/hr to 100 R/hr for channels R-1 through R-10.

Radiation Monitoring System panel alarms consist of an indicator light for high radiation and a light to annunciate detector failure or loss of signal. The remote meter and alarm assembly at the detector contains a red indicator light that is actuated on high radiation. The criticality monitor will actuate a special evacuation alarm in the required areas.

The second area Radiation Monitoring System consists of seven channels (R-25 through R-30 and R-39) which were installed after six years of operation as a result of needs identified through experience. The original equipment supplier was Eberline. One detector, R-30, is an ion chamber. The remainder are all GM tube-type detectors. Local indication is available near the detectors. A local alarm is available only on channel R-30.

Remote indication is not available and trend data are not recorded for the following channels:

Channel	Area Monitor
R-25	Rad-waste drum area
R-26	SGBT ion exchange filter room
R-27	Spent resin storage tank room
R-28	Compacted drum storage area

Channel	Area Monitor
R-29	SGBT monitor tank room
R-30	Reactor cavity sump C
R-39	Sludge intercept filter area

The third Auxiliary Area Radiation Monitoring System consists of 23 channels. This system was installed as a result of lessons learned after TMI and NRC requirements. The original equipment suppliers were Eberline and General Atomic. The detectors are ion chambers and energy compensated GM tubes. Local readout is available at field mounted data acquisition modules (DAM) and two central control consoles. An additional, five portable cart mounted beta air monitors can be tied into the system and can be read out at either central control console. High radiation alarms are displayed at the two central control consoles. Historical data is stored at the data acquisition modules via a microcomputer. Two recorders in the Control Room are provided for the wide range containment monitors (R-40 and R-41). These two monitors are designed to operate in a post-LOCA environment. The Auxiliary Area Radiation System consists of the following:

Address Code	Area Monitor
03-01	Waste disposal
03-02	Post accident sampling room
03-03	Component cooling heat exchanger
04-01	Machine Shop
04-02	Monitor room
04-03	Make-up demineralizer
04-04	I&C Storage Area
05-01	RHR pump pit
05-02	Radwaste compactor
05-03	Auxiliary building loading dock
06-01	I & C Shop
06-02	Shield building vent filters
06-03	Control Room A/C vent filters
06-04	Containment ventilation exhaust filters
06-05	Zone SV exhaust filters
07-01	RAF count room
07-02	Technical Support Center
07-03	Technical Support Center Stairwell
08-01	Sulfuric acid storage tank
08-02	Containment spray pumps
08-03	Heating boiler

Instrument Number	Area Monitors
2906401	1B containment hi level radiation monitor (R-41)
2906501	1A containment hi level radiation monitor (R-40)

Main Steam Line Primary to Secondary Leakage Monitors (R-42, R-43)

These two channels continuously monitor the N-16 (nitrogen) concentration in both of the main steam lines using detectors, which are placed just outside of containment. In 2001, R-42 was installed near main steam 1A and R-43 was installed near main steam 1B. No automatic actions or automatic controls are associated with either of these channels. These channels will provide an early indication of primary to secondary leakage similar to R-15 and R-19. Installation of these channels will meet the EPRI PWR Primary-To-Secondary Leak Guidelines (TR-104788-R2).

Radiation Monitoring System-Control Room Panel

All of the control room system equipment for one unit is centralized in three panels. High reliability and ease of maintenance are emphasized in the design of this system. Sliding channel drawers are used for rapid replacement of units, assemblies and entire channels. It is possible to completely remove the various chassis from the panels.

Recording

Data from each of the Control Room monitors is recorded by an electrical storage device.

Monitor Ranges

The ranges of the radiation monitors are given in Table 11.2-7 Calibration curves showing count rate versus radiation activity (μCi/cc) are provided for all isotopes to be measured.

Operating Conditions

Table 11.2-8 indicates the detector operating condition during normal operation. Where fluid temperature is too high for the monitor, a cooling device is included. The different operating temperature ranges are within the design limits of the sensors.

Radiation Protection Program

Personnel Monitoring

A permanent record of accumulated total effective dose equivalent (TEDE) exposure received by individuals is maintained per 10 CFR 20 requirements. TEDE is the sum of the deep-dose equivalent (DDE) and the committed effective dose equivalent (CEDE). DDE exposures are provided by the interpretation of TLD chips. Electronic dosimeters, or in some cases self-reading dosimeters, provide day-by-day indication of DDE exposure.

All persons who are required to be monitored are issued beta-gamma TLD badges and are required to wear such badges at all times while within the radiologically controlled areas of the plant. Personnel who are issued TLD badges pick them up prior to entry into the radiologically controlled area and deposit them after leaving the radiologically controlled area at the Radiation Protection Office.

Special or additional TLD badges are issued at the discretion of Radiation Protection personnel.

The TLD badges are processed on a routine basis at specified intervals. Badges are processed immediately whenever it appears that an over exposure may have occurred, or any time deemed necessary by Radiation Protection personnel.

An electronic dosimeter or a self-reading dosimeter is issued for entry into the Radiologically Controlled Area in addition to a TLD badge for individuals whose work conditions make a day-to-day indication of exposure desirable. Electronic dosimeters require the Radiological worker to sign in on the assigned Radiation Work Permit (RWP) using control stations outside of the Radiation Protection Office (RPO). If self-reading pocket dosimeters are used they are read, recorded and re-zeroed regularly and are issued for entry into the RCA when deemed necessary by Radiation Protection. Electronic dosimeters are reset by signing out on the control stations located outside the RPO. Dosimeter and TLD records furnish the exposure data for the administrative control of radiation exposure.

Personnel Bio-Assay Program

The Personnel Bio-Assay Program at the Kewaunee Nuclear Power Plant consists of whole-body counting and Passive Monitoring with a portal monitor. Whole body counts are made for select personnel routinely exposed to radioactive materials.

Personnel Protection Equipment

Personnel entering the radiologically controlled area are required to wear protective clothing as required by the radiation work permit. The nature of the work to be done and the conditions of the area are the governing factors in the selection of protective clothing to be worn by individuals.

Personnel Respiratory Protection

The Kewaunee Nuclear Power Plant is designed to minimize concentrations of airborne radioactivity due to inadvertent leaks, spills or other causes by filtered ventilation systems and isolation of equipment in compartments. Further, a radiation protection program is provided to minimize airborne concentrations by detecting and controlling potential sources of airborne radioactivity. The normal concentrations present in areas occupied by personnel are much less than the derived air concentrations (DACs) and the use of respiratory protection equipment is normally not necessary.

Usage of respiratory protection devices will be evaluated under unusual situations arising from plant operations to limit CEDE (i.e. TEDE) below 10CFR20 requirements. When it is not practicable to apply process or other engineering controls to control the concentrations of radioactive material in the air, consistent with maintaining the TEDE ALARA, the necessary protective devices will be specified and monitored by radiation protection personnel.

Three general types of respiratory protective equipment are utilized in the respiratory protection program. The type used for a particular circumstance is determined by the concentration in the air and the protection factor needed to prevent personnel from breathing or being exposed to airborne radioactivity in excess of that specified by 10 CFR 20, Section 20.1201.

The three general types of respiratory protective equipment intended for use are:

a. Air-Purifying Respirators

   These units are utilized for protection against airborne particulates only.

b. Atmosphere Supplying, Air-Line Respirators

   These units provide protection from radioactive particulates, gases and vapors. These respirators use the plant's Service Air System to supply a filtered and regulated source of breathing air to either a full-face piece or hood.

c. Atmosphere Supplying, Self-Contained Breathing Apparatus (SCBA)

   These respirators utilize back mounted compressed air cylinders to supply a full-face piece in the pressure-demand mode of operation. The SCBA provides protection against radioactive particulates, gases and vapors and may also be used in atmospheres considered Immediately Dangerous to Life or Health (IDLH).

A Cascade System consisting of large cylinders of compressed air exists for refilling the SCBA cylinders on an as-needed basis.

After each use, non-disposable respirators are cleaned and sanitized before being reissued. The Radiation Protection Group cleans, maintains and controls the use of all respiratory protection equipment. Decontaminating and sanitizing agents are utilized which prevent damage to sealing surfaces and transparent face pieces. Following air drying, the respirators are checked for radioactive contamination and re-cleaned or disposed of if necessary. Each respirator is then visually inspected for mechanical degradation, and stored in a clean, sealed plastic bag.

Routine maintenance on respiratory equipment is performed, as the need becomes apparent through visual inspection following cleaning. For SCBA respirators, the ability to provide a sufficient, regulated air supply is periodically tested. For airline respirators, the quality of the breathing air is also periodically tested by analyzing samples of the Service Air System.

Each individual whose work requires the use of respiratory protection is trained in proper methods of donning and use of the various types of equipment. Initial training is done by training group personnel. At this time, individual problems, such as interference with sealing due to eyeglasses, will become apparent. Periodic retraining is done by the training group based on an individual's need for respiratory equipment use.

In accordance with 29 CFR 1910.134, a Licensed Health Care Professional performs evaluations of medical fitness for individuals requiring the use of respiratory protective equipment. Quantitative mask fit tests are administered annually.

If prolonged use of respiratory equipment is necessary, personnel are relieved at reasonable intervals such that usage of the equipment does not endanger personnel or discourage observance of proper work and safety procedures.

Action Levels

Areas are posted as Airborne Radioactivity Areas whenever an individual present could exceed, during the hours in a week, an intake of 0.6% of an ALI or 12 DAC-hrs. ALI and DAC values are given in Table 1 of Appendix B of 10 CFR 20.

The use of respirator protection devices will be evaluated pursuant to 10 CFR 20.1702 consistent with maintaining the TEDE ALARA. When respirators are deemed necessary, 10 CFR 20.1703 shall govern their proper usage.

Facilities and Access Provisions

The plant site is divided into various areas depending on radiation and/or contamination levels present.

The Clean Area includes all areas in the plant, which are not designated as the Radiologically Controlled Area and have radiation and contamination levels within the limits set for the Clean Area.

The Radiologically Controlled Area includes all areas in the plant in which radiation and/or contamination levels or the potential of such levels exist above those limits as stated for the Clean Areas. All entrances to the Radiologically Controlled Areas are posted as such and limited to emergency access only. Normal entry to and exit from the Radiologically Controlled Area is through the designated Radiation Protection Office (RPO) only.

Exclusion Areas include all areas within the Radiologically Controlled Areas found to have levels of radiation and/or contamination above those specified for Radiologically Controlled Areas. These are Radiation, High Radiation, Very High Radiation, Airborne Radioactivity, Contaminated, and Radioactive Materials Areas. Each area is posted with its respective sign and requirements.

The Unrestricted Area includes all property owned by WPS immediately adjacent to and surrounding the Restricted Area. Normal environmental radiation and control procedures apply here.

The general arrangement of the service facilities is designed to provide adequate personnel decontamination and change areas.

Locker rooms are used to store items of personal clothing not required or allowed in the Radiologically Controlled Area. A supply of various protective clothing items for personnel is maintained within the RCA.

Personnel contamination monitors are located outside the Radiation Protection Office (RPO). All personnel are required to survey themselves on leaving the Radiologically Controlled Area.

A decontamination shower and wash sink are located adjacent to the RPO. Decontamination facilities are described in Section 9.7.3.

The fuel storage area has facilities to handle the decontamination of large items of equipment. The decontamination area contains service facilities.

All visitors to the RCA are required to wear electronic dosimeters, or in some cases, self-reading dosimeters, TLD badges, or be provided with an escort having personnel monitoring devices.

Administrative and physical security measures are employed to prevent unauthorized entry of personnel to any High Radiation Area. These measures include the following:

Personnel are restrained from entry by locks, interlocks, or similar devices that alert the Control Room of entries into High Radiation Areas (giving exposure rates greater than 1000 mREM/hr). Administrative control requires the issuance of a Radiation Work Permit (RWP) prior to entry to any High Radiation Area.

a. Any individual or group of individuals entering a High Radiation Area is provided with a radiation-monitoring device, which continuously indicates the radiation, dose rate in the area.

b. Personnel are required to wear protective clothing as designated by Radiation Protection personnel for entry into a contaminated area.

Normal access into the Radiologically Controlled Area is through electronic dosimeter (ED) activated turnstiles. Workers are required to sign onto an RWP using the sign-on terminals. After the system activates the ED, the workers proceed through the turnstiles. The ED must be turned on to activate the turnstile. Radiation protection personnel can provide guidance and review applicable RWP requirements.

Instrumentation

Laboratory facilities are provided for the Radiation Protection Group. These facilities include both laboratory and calibration rooms. A Counting Room is equipped to analyze routine air sample and contamination swipe surveys. The Radiation Protection Office (RPO) also serves as a central location for portable radiation survey instruments, respiratory protection equipment and contamination control supplies.

The survey equipment is supplemented with radioanalytical instrumentation located in the Counting Room and Radioanalytical Facility. Portal monitors are located in the Security Building as a final check on all personnel leaving the plant.

The types of portable radiation survey instruments available for routine monitoring functions are adequate for the performance of the surveys required by 10 CFR Part 20.

Survey and radioanalytical instruments are periodically calibrated. Calibration and maintenance records are recorded for each instrument.

11.2.4 EVALUATION

The whole body gamma dose in the Control Room under accident conditions is calculated assuming TID-14844 releases of the following sources to reactor containment:

- 100% of the noble gases
- 50% of the halogens
- 1% of the nonvolatile fission product inventory.

The above sources tabulated in Table 11.2-11 are assumed to be homogeneously distributed within the free volume of the Containment. The source intensity in Mev per photon as a function of time after the accident is determined by considering decay only; no credit is taken for filtration or wash down.

The direct dose (deep dose equivalent) in the Control Room due to the activity dispersed within the Containment is calculated by a digital computer program, which is based on a point kernel attenuation model. The source region is divided into a number of incremented source volumes and the associated attenuation, gamma ray buildup, and distance through regions between each source point and the control room are computed. The summation of all point source contributions gives the total direct dose (deep dose equivalent) in the Control Room.

In addition to the direct dose (deep dose equivalent), the contribution of scattered radiation was also estimated. These estimates indicate that scattered radiation levels will contribute less than 10% of the direct dose (deep dose equivalent). Scattered radiation levels include both scattering from air (sky-shine) and scattering from large surfaces in the vicinity of the Control Room.

Without any credit for intervening walls and structures, the integrated four-week deep dose equivalent (forty hours per week) to Control Room operators would be less than 2.5 REM. This includes contributions from the Control Room.

Further study (Reference 7) has shown that the maximum thirty-day doses would be:

- Whole Body Dose (γ) = 1.9 REM
- Beta Skin Dose (β) = 57 REM
- Thyroid Dose = 27 REM*

* Although not credited in the above study, in an effort to prevent a post-accident IN VIVO dose to the thyroid glands of operating personnel, iodide can be made available to personnel early in the accident scenario. This medication, when taken six hours prior to an exposure to radioiodine, gives a minimum protective effect of 97% with respect to the localization of radioiodine in the human thyroid. The medication can be taken safely by most humans at 130 mg/day for approximately ten days.

These dose estimates were determined using input assumptions found in Section 14.3.5 and Reference 7.

While the halogen release is lower than the original estimates, it is consistent with current industry findings based on the post-TMI-2 investigations (Reference 6).

All components necessary for the operation of the external recirculation loop following a loss-of-coolant accident are capable of remote manual operation from the Control Room and can be powered by the diesel generators so that it should not be necessary to enter the Auxiliary Building in the vicinity of the recirculation loops.

To determine the possible exposure that an operator could receive under accident conditions while operating a manual backup item (e.g., valve), it is estimated rather conservatively that it will require fifteen minutes to operate the valve. In addition, it is assumed that an additional fifteen minutes is required to get to and from the manual equipment. The total integrated whole body deep dose equivalent that an operator would receive performing the above operation would be about 8 REM. This exposure is calculated at one-half hour following the accident and assumes that the equipment being operated or serviced is adjacent to the Shield Building. The basis for such a calculation relating to the RHR System are given in Figures 11.2-5 and 11.2-6. Exposures in the vicinity of equipment located within the Auxiliary Building would be much less due to the shielding afforded by intervening walls and structures in the Auxiliary Building.

Hazards due to the radioactive halogen gas exposure to operators in the Control Room during the course of a loss-of-coolant accident is minimized by providing a Control Room air conditioning system that filters all incoming air and is capable of internal recirculation within the Control Room to minimize the amount of outside air drawn into the Control Room. In the event of high radiation level in the Control Room ventilating air, makeup air from outside is shut off, and absolute and charcoal filters are placed in service on the recirculating air stream. The Control Room ventilating fans are supplied by emergency power.

If maintenance on the loop is required, such operations would be limited in duration. The radiation levels adjacent to equipment containing the sump water and fission products could be as high as 200 to 300 REM per hour shortly after the initiation of recirculation. Any emergency maintenance operations described above could be carried out using portable breathing equipment to limit the inhalation hazard.

11.2.5 MINIMUM OPERATING CONDITIONS

All liquid waste releases are continuously monitored for gross activity during discharge to ensure that the activity concentration limits are below those specified in 10 CFR 20 for unrestricted areas.

11.2.6 TESTS AND INSPECTIONS

Complete radiation surveys were made throughout the Containment and Auxiliary Building during initial phases of plant start up. Survey data was taken and compared with levels at power levels of 0.01%, 10% and 100% of rated full power. Survey data were reviewed for conformance with design levels before increasing to the next power range. The 100% power readings are repeated after each refueling outage to ensure no degradation of shielding has occurred.

The radiation monitors are tested in accordance with the surveillance frequencies specified in the Technical Specifications and Offsite Dose Calculation Manual (ODCM).

REFERENCES - SECTION 11.2

1. NRC Safety Evaluation Report, SA Varga (NRC) to CW Giesler (WPS), Letter No. K-83-12 dated January 12, 1983

2. NRC Order, SA Varga (NRC) to CW Giesler (WPS), Letter No. K-83-59 dated March 14, 1983

3. NRC Safety Evaluation Report, MB Fairtile (NRC) to DC Hintz (WPS), Letter No. K-85-14 dated January 9, 1985

4. NRC Safety Evaluation Report, SA Varga (NRC) to DC Hintz (WPS), Letter No. K-85-7 dated January 2, 1985

5. **DELETED**

6. "Iodine-131 Behavior During the TMI-2 Accident" NSAC/30, Nuclear Safety Analysis Center, September 1981

7. Kewaunee Updated Control Room Habitability Evaluation Report, Letter from CR Steinhardt (WPS) to NRC Document Control Desk, dated February 28, 1989

14.3 REACTOR COOLANT SYSTEM PIPE RUPTURES (LOSS OF COOLANT ACCIDENT)

General

Condition III - Infrequent Faults

By definition, Condition III occurrences are faults which may happen very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the Reactor Coolant System or of containment barriers.

The time sequence of events for the small breaks is shown in Table 14.3-1a and Table 14.3-1b presents the results of the analysis.

Condition IV - Limiting Faults

Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These are the most drastic occurrences which must be designed against, and they represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR 50.67. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the Emergency Core Cooling System and the containment.

The analysis of total effective dose equivalent (TEDE) doses, resulting from events leading to fission product release, appears in Section 14.3.5. The fission product inventories, which form a basis for these calculations are presented in Appendix D. Sections 14.3.4, 14.3.5 and Appendix H also include the discussions of systems interdependency contributing to limiting fission product leakage from the containment following a Condition IV occurrence.

The time sequence of events for a large break is shown in Table 14.3.2-8 and Figure 14.3.2-1, and Table 14.3.2-9 presents the results of these analyses.

14.3.1 LOSS OF REACTOR COOLANT FROM SMALL RUPTURED PIPES OR FROM CRACKS IN LARGE PIPES WHICH ACTUATES EMERGENCY CORE COOLING SYSTEM

Identification of Causes and Accident Description

A loss-of-coolant accident is defined as a rupture of the Reactor Coolant System piping or of any line connected to the system. See Section 4.1.3 for a more detailed description of the loss of reactor coolant accident boundary limits. Ruptures of small cross sections will cause expulsion of the coolant at a rate which can be accommodated by the charging pumps which would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant, which would be released to the containment, contains the fission products existing in it.

Should a larger break occur, depressurization of the Reactor Coolant System causes fluid to flow to the Reactor Coolant System from the pressurizer resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the pressurizer low-pressure trip setpoint is reached. The Safety Injection System is actuated when the appropriate setpoint is reached. The consequences of the accident are limited in two ways:

1) Reactor trip and borated water injection complement void formation in causing rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay,

2) Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperature.

Before the break occurs the plant is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals and the vessel continues to be transferred to the Reactor Coolant System. The heat transfer between the Reactor Coolant System and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary side, system pressure increases and steam dump may occur. Makeup to the secondary side is provided by the auxiliary feedwater pumps. The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates emergency feedwater flow by starting auxiliary feedwater pumps. As discussed in Section 6.6, manual initiation of auxiliary feedwater is acceptable at low power levels. The secondary flow aids in the reduction of Reactor Coolant System pressure. When the RCS depressurizes to 700 psig, the accumulators begin to inject water into the reactor coolant loops. Reactor coolant pump trip is assumed to be coincident with the reactor trip and effects of pump coast-down are included in the blowdown analyses.

Analysis of Effects and Consequences

Method of Analysis

The requirements of an acceptable ECCS Evaluation Model are presented in Appendix K of 10 CFR 50 (Reference 1). The requirements of Appendix K regarding specific model features were met by selecting models, which provide a significant overall conservatism in the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS system. Decay heat generated throughout the transient is also conservatively calculated as required by Appendix K of 10 CFR 50. The small break Loss-of-Coolant Accident analysis is documented in KNPP Steam Generator Replacement and Tave Operating Window Program Licensing Report, November 2000.

Small Break LOCA Analysis Using NOTRUMP

The Westinghouse NOTRUMP Small Break Evaluation Model consists of the NOTRUMP and SBLOCTA computer codes. NOTRUMP is used to model the system hydraulics and SBLOCTA calculates the fuel rod cladding heatup.

The postulated small break LOCA is predominately a gravity dominated accident in which the slow draining of the RCS is accompanied by the formation of distinct mixture levels throughout the RCS. These mixture levels vary with time and are dependent upon the transient two-phase transport of mass and energy, which takes place within the RCS during the course of the accident. Consequently, the degree of accuracy with which a system model is capable of simulating the RCS response to a small break LOCA is dependent upon the model's capability to accurately model the RCS transient mass and energy distribution.

For postulated LOCAs due to small breaks the NOTRUMP computer code is used to calculate the transient depressurization of the RCS as well as to describe the mass and enthalpy of flow through the break. The NOTRUMP computer code is a state-of-the-art one-dimensional general network code incorporating a number of advanced features. Among these are calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes and regime dependent heat transfer correlation. The Westinghouse NOTRUMP Small Break Evaluation Model was developed to determine the RCS response to design basis small break LOCAs, and to address NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accident in Westinghouse-Designed Operating Plants".

NOTRUMP (Reference 2 and 3) is a general one-dimensional nodal network computer code which describes the spatial detail of the RCS with a network of fluid nodes (representing various system fluid volumes), flow links (representing various fluid flow paths), metal nodes (representing various metal masses), and heat transfer links (representing various heat transfer paths between metal structures and surrounding fluid). The use of NOTRUMP in the analysis involves, among other things, the representation of the reactor core as heated control volumes with an associated phase separation model to permit a transient mixture height calculation. The broken loop and intact loop are each modeled explicitly. Transient behavior of the system is determined from the governing conservation equations of mass, energy and momentum. The multi-node capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, enables a proper calculation of behavior of loop seal during a postulated small break LOCA.

Peak clad temperature calculations are performed with the SBLOCTA code (Reference 4), using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. The code evaluates the fuel cladding and the coolant temperatures during the hypothetical small break LOCA. Each of the fuel rods modeled by SBLOCTA is analyzed using finite-difference conduction equations in both the axial and radial directions. It calculates the effect of cladding swell and burst and considers the exothermic reaction between zircaloy and water. A top skewed axial power shape is chosen for the hot rod because the power is concentrated in the upper regions of the core. Such a distribution is limiting for small-break LOCAs because it minimizes coolant swell, while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations. The small break LOCA analysis assumes the core continues to operate at full rated power until the control rods are completely inserted.

Small break LOCA calculations are based on minimum safeguards assumptions designed to minimize pumped ECCS flow to the core. These calculations include loss of a train of ECCS and high head SI pump degradation of pump head by 10%. For a small break LOCA with an equivalent break diameter less than the inner diameter of the high head SI line, pumped high head SI flow is delivered to both the intact and broken loop at the RCS backpressure (Figure 14.3-1a). Justification for this assumption is provided in Reference 6. The effect of flow from the RHR pumps is not considered since their shutoff head is lower than RCS pressure during the portion of the transients considered here. For a small break LOCA with an equivalent break diameter greater than or equal to the inner diameter of the high head SI line, pumped high head SI flow is delivered only to the intact loop with one line spilling to containment back pressure (Figure 14.3-1b). This is assumed since the modeled break may include the severance of the high head SI line or an area of a severed high head SI line. Flow from the RHR pumps into the upper plenum is assumed since transient RCS pressure is below the shutoff head (Figure 14.3-1b). However, RHR flow was not fully credited since the NOTRUMP condensation models for the upper plenum have not been licensed by the NRC. The low head SI flow shown in Figure 14.3-1b was reduced by 50%, which conservatively compensates for the effects of condensation in the upper plenum.

Delivery of the SI flow to the RCS was assumed to be delayed 30 seconds after the generation of a SI signal. This delay includes the time required for diesel startup and loading of the SI pumps onto the emergency buses and for the pump to come to full speed in order to deliver full flow. The assumed delay time is sufficient to account for degraded grid conditions. Finally, the new and approved SI condensation model (Reference 6) was used for all analysis cases.

Results

Reactor Coolant System Pipe Breaks

This section, presents results of the limiting break size in terms of highest peak clad temperature. The worst break size (small break) is a 3-inch diameter break. The depressurization transient for this break is shown in Figure 14.3-2a. The extent to which the core is uncovered is shown in Figure 14.3-2b.

During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained. The resultant heat transfer cools the fuel rod and clad to very near the coolant temperatures as long as the core remains covered by a two-phase mixture.

The maximum hot spot clad temperature calculated during the transient is 1030°F. This analysis assumes the most limiting temperature conditions, fuel types (High-Tavg Siemens 14x14 fresh heavy fuel) and includes the effects of fuel densification as described in Reference 5. The peak clad temperature transients are shown in Figure 14.3-2c for the worst break size, i.e., the break with the highest peak clad temperature. When the mixture level drops below the top of the core, the steam flow computed by NOTRUMP provides cooling to the upper portion of the core. The core heat transfer coefficients for this phase of the transient are given in Figure 14.3-2g. The hot spot fluid temperature for the worst break is shown in Figure 14.3-2f.

The reactor shutdown time (5.0-sec) is equal to the reactor trip signal time (2.0 sec) plus 3.0 sec for rod insertion. During this rod insertion period, the reactor is conservatively assumed to operate at rated power.

Conclusions

Analyses presented in this section show that the high head portion of the Emergency Core Cooling System, together with accumulators, provide sufficient core flooding to keep the calculated peak clad temperatures below required limits of 10 CFR 50.46. Hence, adequate protection is afforded by the Emergency Core Cooling System in the event of a small break loss-of-coolant accident.

Following the TMI accident, Westinghouse performed generic studies of small break loss-of-coolant accidents. Results of these studies indicated that peak clad temperatures greater than 2200°F may occur if the reactor coolant pumps are tripped after a significant loss of reactor coolant inventory. To prevent such a loss, the operators are instructed to trip the pumps early in the accident.

Additional Break Sizes

Additional break sizes were analyzed. Figures 14.3-3a and 14.3-4a present the RCS pressure transient for the 2- and 4-inch breaks, respectively and Figures 14.3-3b and 14.3-4b present the volume history (mixture height) plots for both breaks. The peak clad temperatures for both cases are less than the peak-clad temperature of the 3-inch break. The peak clad temperatures for the 2-inch break case is given in Figure 14.3-3c.

The time sequence of events for small breaks analyzed is shown in Table 14.3-1a, and Table 14.3-1b presents the results for these analyses.

14.3.2 MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURES (LOSS-OF-COOLANT ACCIDENT)

The analysis specified by 10 CFR 50.46 (Reference 1), "Acceptance Criteria for Emergency

Core Cooling Systems for Light Water Power Reactors", is presented in this section. The results of the Best-Estimate large break loss-of-coolant accident (LOCA) analysis are summarized in Table 14.3.2-8, and show compliance with the acceptance criteria.

For the purpose of ECCS analyses, Westinghouse (W) defines a large break loss-of-coolant accident (LOCA) as a rupture 1.0 ft^2 or larger of the reactor coolant system piping including the double ended rupture of the largest pipe in the reactor coolant system or of any line connected to that system. The boundary considered for loss of coolant accidents as related to connecting piping is defined in Section 4.1.3.

Should a major break occur, rapid depressurization of the Reactor Coolant System (RCS) to a pressure nearly equal to the containment pressure occurs in approximately 35 seconds, with a nearly complete loss of system inventory. Rapid voiding in the core shuts down reactor power. A safety injection system signal is actuated when the low pressurizer pressure setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

1. Borated water injection complements void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. An average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical. However, no credit is taken for the insertion of control rods to shut down the reactor in the large break analysis.

2. Injection of borated water provides heat transfer from the core and prevents excessive cladding temperatures.

Before the break occurs, the reactor is assumed to be in a full power equilibrium condition, i.e., the heat generated in the core is being removed through the steam generator secondary system. At the beginning of the blowdown phase, the entire RCS contains sub-cooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. During blowdown, heat from fission product decay, hot internals and the vessel, continues to be transferred to the reactor coolant. After the break develops, the time to departure from nucleate boiling is calculated. Thereafter, the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes voided, both transition boiling and forced convection are considered as the dominant core heat transfer mechanisms. Heat transfer due to radiation is also considered.

The heat transfer between the RCS and the secondary system may be in either direction, depending on the relative temperatures. In the case of the large break LOCA, the primary pressure rapidly decreases below the secondary system pressure and the steam generators are an additional heat source. In the Kewaunee Nuclear Plant Large Break LOCA analysis using the WCOBRA/TRAC UPI methodology, the steam generator secondary is conservatively assumed to be isolated (main feedwater and steam line) at the initiation of the event to maximize the secondary side heat load.

Performance Criteria for Emergency Core Cooling System

The reactor is designed to withstand thermal effects caused by a loss-of-coolant accident including the double-ended severance of the largest reactor cooling system cold leg pipe. The reactor core and internals together with the Emergency Core Cooling System (ECCS) are designed so that the reactor can be safely shut-down and the essential heat transfer geometry of the core preserved following the accident. Long-term coolability is maintained.

When the RCS depressurizes to approximately 750 psig, the accumulators begin to inject borated water into the reactor coolant loops. Borated water from the accumulator in the faulted loop is assumed to spill to containment and be unavailable for core cooling for breaks in the cold leg of the RCS. Flow from the accumulator in the intact loop may not reach the core during depressurization of the RCS due to the fluid dynamics present during the ECCS bypass period. ECCS bypass results from the momentum of the fluid flow up the downcomer due to a break in the cold leg, which entrains ECCS flow out toward the break. Bypass of the ECCS diminishes as mechanisms responsible for the bypassing are calculated to be no longer effective.

The blowdown phase of the transient ends when the liquid level in the lower plenum reaches its minimum. After the end of the blowdown, refill of the reactor vessel lower plenum begins. Refill is completed when emergency core cooling water has filled the lower plenum of the reactor vessel, which is bounded by the bottom of the active fuel region of the fuel rods (called bottom of core (BOC) recovery time).

The reflood phase of the transient is defined as the time period lasting from BOC recovery until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the latter stage of blowdown and on into the beginning of reflood, the intact loop accumulator tank rapidly discharges borated cooling water into the RCS. Although a portion injected prior to end of bypass is lost out the cold leg break, the accumulator eventually contributes to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The high head safety injection (HHSI) pump aids in the filling of the downcomer and core and subsequently supply water to help maintain a full downcomer and complete the reflooding process. The low head safety injection (LHSI), which injects into the upper plenum (hence, upper plenum injection - UPI) also aids the reflooding process by providing water to the core through the vessel upper plenum.

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been reduced to long-term steady state levels associated with dissipation of residual heat generation. After the water level of the refueling water storage tank (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching from the injection mode to the sump recirculation mode of ECCS operation. Spilled borated water is drawn from the engineered safety features (ESF) containment sumps by the LHSI pumps (also called the Residual Heat Removal pumps, or RHR pumps) and returned to the upper plenum and RCS cold legs. Figure 14.3.2-1 contains a schematic of the bounding sequence of events for the Kewaunee large break LOCA transient.

For the Best-Estimate large break LOCA analysis, one ECCS train, including one high head safety injection (HHSI) pump and one RHR (low-head) pump, starts and delivers flow through the injection lines. One branch of the HHSI injection line spills to the containment backpressure; the other branch connects to the intact loop cold leg accumulator line. The RHR injection line connects directly into the upper plenum. Both emergency diesel generators (EDGs) are assumed to start in the modeling of the containment fan coolers and spray pumps. Modeling full containment heat removal systems operation is required by Branch Technical Position CSB 6-1 (Reference 14) and is conservative for the large break LOCA.

To minimize delivery to the reactor, the HHSI branch line chosen to spill is selected as the one with the minimum resistance. In addition, the pump performance curves are degraded, with the high head degraded by 15% of design head and the low head degraded by 10% of design head.

Large Break LOCA Analytical Model

In 1988, as a result of the improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs, the NRC staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models", so that a realistic evaluation model may be used to analyze the performance of the ECCS during a hypothetical LOCA (Reference 7). Under the amended rules, best-estimate thermal-hydraulic models may be used in place of models with Appendix K features. The rule change also requires, as part of the analysis, an assessment of the uncertainty of the best-estimate calculations. It further requires that this analysis uncertainty be included when comparing the results of the calculations to the prescribed acceptance limits. Further guidance for the use of best-estimate codes was provided in Regulatory Guide 1.157 (Reference 8).

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology (Reference 9). This method outlined an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis.

A LOCA evaluation methodology for three- and four-loop PWR plants based on the revised 10 CFR 50.46 rules was developed by Westinghouse with the support of EPRI and Consolidated Edison and was approved by the NRC (Reference 10). The methodology is documented in WCAP-12945, "Code Qualification Document (CQD) for Best Estimate LOCA Analysis" (Reference 11). Extension of this methodology to plants equipped with residual heat removal (RHR) injection into the upper plenum was approved in May 1999 (Reference 15) and is documented in Reference 12.

The thermal-hydraulic computer code which was reviewed and approved for the calculation of fluid and thermal conditions in the PWR during a large break LOCA is WCOBRA/TRAC

Version MOD7A, Rev. 1 (Reference 11).

WCOBRA/TRAC combines two-fluid, three-field, multi-dimensional fluid equations used in the vessel with one-dimensional drift-flux equations used in the loops to allow a complete and detailed simulation of a PWR. This best-estimate computer code contains the following features:

- Ability to model transient three-dimensional flows in different geometries inside the vessel
- Ability to model thermal and mechanical non-equilibrium between phases
- Ability to mechanistically represent interfacial heat, mass, and momentum transfer in different flow regimes
- Ability to represent important reactor components such as fuel rods, steam generators, reactor coolant pumps, etc.

The reactor vessel is modeled with the three-dimensional, three-field fluid model, while the loop, major loop components, and safety injection points are modeled with the one-dimensional fluid model.

The basic building block for the vessel is the channel, a vertical stack of single mesh cells. Several channels can be connected together by gaps to model a region of the reactor vessel. Regions that occupy the same level form a section of the vessel. Vessel sections are connected axially to complete the vessel mesh by specifying channel connections between sections. Heat transfer surfaces and solid structures that interact significantly with the fluid can be modeled with rods and unheated conductors. The fuel parameters are generated using the Westinghouse fuel performance code (PAD 4.0, Reference 6).

One-dimensional components are connected to the vessel. Special purpose components exist to model specific components such as the steam generator and pump.

A typical calculation using WCOBRA/TRAC begins with the establishment of a steady-state initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are discussed in the next section.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient through blowdown, refill, and reflood follows continuously, using the same computer code (WCOBRA/TRAC) and the same modeling assumptions. Containment pressure is modeled with the BREAK component using a time dependent pressure table. Containment pressure is calculated using the COCO code (Reference 5) and mass and energy releases from the WCOBRA/TRAC calculation. The parameters used in the containment analysis to determine this pressure curve are presented in Tables 14.3.2-1 through 14.3.2-3.

The methods used in the application of WCOBRA/TRAC to the large break LOCA are described in References 10 through 12. A detailed assessment of the computer code WCOBRA/TRAC was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the code's ability to predict key physical phenomena in a PWR large break LOCA. Modeling of a PWR introduces additional uncertainties which are identified and quantified in the plant-specific analysis (Reference 13). The final step of the best-estimate methodology is to combine all the uncertainties related to the code and plant parameters and estimate the PCT at the 95th percentile ($PCT^{95\%}$). The steps taken to derive the PCT uncertainty estimate are summarized below:

1. Plant Model Development
   In this step, a WCOBRA/TRAC model of the Kewaunee Nuclear Power Plant (KNPP) is developed. A high level of noding detail is used, in order to provide an accurate simulation of the transient. However, specific guidelines are followed to assure that the model is consistent with models used in the code validation. This results in a high level of consistency among plant models, except for specific areas dictated by hardware differences such as in the upper plenum of the reactor vessel or the ECCS injection configuration.

2. Determination of Plant Operating Conditions
   In this step, the expected or desired range of the plant operating conditions to which the analysis applies is established. The parameters considered are based on a "key LOCA parameters" list that was developed as part of the methodology. A set of these parameters, at mostly nominal values, is chosen for input as initial conditions to the plant model. A split break in the cold leg (a longitudinal break along the side of the pipe) is modeled initially, as was determined to be limiting for a typical two-loop plant (Reference 12). A transient is run utilizing these parameters and is known as the "initial transient". Next, several confirmatory runs are made, which vary a subset of the key LOCA parameters over their expected operating range in one-at-a-time sensitivities. The results of these calculations for KNPP are discussed in Section 4 of Reference 13. The most limiting input conditions, based on these confirmatory runs, are then combined into a single transient, which is then called the "reference transient".

3. PWR Sensitivity Calculations
   A series of PWR transients are performed in which the initial fluid conditions and boundary conditions are ranged around the nominal conditions used in the reference transient. The results of these calculations for KNPP form the basis for the determination of the initial condition bias and uncertainty discussed in Section 5 of Reference 13.

   Next, a series of transients are performed which vary the power distribution, taking into account all possible power distributions during normal plant operation. The results of these calculations for KNPP form the basis for the determination of the power distribution bias and uncertainty (response surface) discussed in Section 6 of Reference 13.

   Finally, a series of transients are performed which vary parameters that affect the overall system response ("global" parameters) and local fuel rod response ("local" parameters). The results of these calculations for KNPP form the basis for the determination of the model bias and uncertainty (response surface) discussed in Section 7 of Reference 13.

4. Response Surface Calculations
   The results from the power distribution and global model WCOBRA/TRAC runs performed in Step 3 are fit by regression analyses into equations known as response surfaces. The results of the initial conditions run matrix are used to generate a PCT uncertainty distribution.

5. Uncertainty Evaluation
   The total PCT uncertainty from the initial conditions, power distribution, and model calculations is derived using the approved methodology (References 12). The uncertainty calculations assume certain plant operating ranges which may be varied depending on the results obtained. These uncertainties are then combined to determine the initial estimate of the total PCT uncertainty distribution for the split and limiting guillotine breaks. The results of these initial estimates of the total PCT uncertainty are compared to determine the limiting break type. If the guillotine break is limiting, an additional set of guillotine transients are performed which vary overall system response ("global" parameters) and local fuel rod response ("local" parameters). The results of these calculations form the basis for the determination of the model bias and uncertainty for guillotine breaks discussed in Section 8 of Reference 13. Finally, an additional series of runs is made to quantify the bias and uncertainty due to assuming that the above three uncertainty categories are independent. The final PCT uncertainty distribution is then calculated for the limiting break type, and the 95th percentile PCT ($PCT^{95\%}$) is determined, as described later under Uncertainty Evaluation.

6. Plant Operating Range
   The plant operating range over which the uncertainty evaluation applies is defined. Depending on the results obtained in the above uncertainty evaluation, this range may be the desired range established in step 2, or may be narrower for some parameters to gain additional margin.

There are three major uncertainty categories or elements:

- Initial condition bias and uncertainty
- Power distribution bias and uncertainty
- Model bias and uncertainty

Conceptually, these elements may be assumed to affect the reference transient PCT as shown below:

$$PCT_i = PCT_{REF,i} + \Delta PCT_{IC,i} + \Delta PCT_{PD,i} + \Delta PCT_{MOD,i} \qquad (14.3.2\text{-}1)$$

where,

$PCT_{REF,i}$ = Reference transient PCT: The reference transient PCT is calculated using WCOBRA/TRAC at the nominal conditions identified in Table 14.3.2-4, for the blowdown and reflood periods.

$\Delta PCT_{IC,i}$ = Initial condition bias and uncertainty: This bias is the difference between the reference transient PCT, which assumes several nominal or average initial conditions, and the average PCT taking into account all possible values of the initial conditions. This bias takes into account plant variations which have a relatively small effect on PCT. The elements which make up this bias and its uncertainty are plant-specific.

$\Delta PCT_{PD,i}$ = Power distribution bias and uncertainty: This bias is the difference between the reference transient PCT, which assumes a nominal power distribution, and the average PCT taking into account all possible power distributions during normal plant operation. Elements which contribute to the uncertainty of this bias are calculational uncertainties, and variations due to transient operation of the reactor.

$\Delta PCT_{MOD,i}$ = Model bias and uncertainty: This component accounts for uncertainties in the ability of the WCOBRA/TRAC code to accurately predict important phenomena which affect the overall system response ("global" parameters) and the local fuel rod response ("local" parameters). The code and model bias is the difference between the reference transient PCT, which assumes nominal values for the global and local parameters, and the average PCT taking into account all possible values of global and local parameters.

The separability of the bias and uncertainty components in the manner described above is an approximation, since the parameters in each element may be affected by parameters in other elements. The bias and uncertainty associated with this assumption is quantified as part of the overall uncertainty methodology and included in the final estimates of $PCT^{95\%}$.

Large Break LOCA Analysis Results

A series of WCOBRA/TRAC calculations were performed using the Kewaunee Nuclear Power Plant (KNPP) input model, to determine the effect of variations in several key LOCA parameters on peak cladding temperature (PCT). From these studies, an assessment was made of the parameters that had a significant effect as will be described in the following sections.

LOCA Transient Description

The plant-specific analysis performed for the Kewaunee Nuclear Power Plant indicated that the split break is more limiting than the double-ended cold leg guillotine (DECLG) break. The plant conditions used in the split break reference transient are listed in

Table 14.3.2-4. The results of the initial transient and the confirmatory calculations performed to determine the final reference transient are listed in Table 14.3.2-5. Note that the initial transient and confirmatory calculations were performed at a slightly lower power (1.4% lower) and a slightly larger T_{avg} operating window than the final reference transient. This was done to incorporate a mid-analysis request to reduce the calorimetric uncertainty from 2.0% to 0.6%. Neither change is considered to be a significant perturbation to the plant initial operating conditions and will not affect the relative outcome of the confirmatory and break spectrum calculations. Table 14.3.2-4 reflects the final reference transient conditions at the higher power. Since many of these parameters are at their bounded values, the calculated results are a conservative representation of the response to a large break LOCA. The following is a description of the final reference transient.

The LOCA transient can be conveniently divided into a number of time periods in which specific phenomena are occurring. For a typical large break, the blowdown period can be divided into the critical heat flux (CHF) phase, the upward core flow phase, and the downward core flow phase. These are followed by the refill, reflood, and long term core cooling phases. The important phenomena occurring during each of these phases in the reference transient are discussed below.

The containment back pressure curve used in all of the calculations is calculated using the COCO code (Reference 5) and mass and energy releases from the WCOBRA/TRAC transient at the lower power. The parameters used in the containment analysis to determine this pressure are listed in Tables 14.3.2-1 through 14.3.2-3. The mass and energy releases from the lower powered transient are shown in Table 14.3.2-6. These mass and energy releases were used to calculate the final containment pressure curve (Figure 14.3.2-1) used in the reference transient shown on Table 14.3.2-5 and all of the subsequent WCOBRA/TRAC calculations. This containment pressure was assessed to be a lower bound to pressure calculated using the mass and energy releases from the final reference transient at the final uprated power.

Critical Heat Flux (CHF) Phase (0-5 seconds)

The reactor coolant pumps are assumed to trip coincident with the break opening. Shortly after the break is assumed to open, the vessel depressurizes rapidly and the core flow decreases as subcooled liquid flows out of the vessel into the broken cold leg. The fuel rods go through departure from nucleate boiling (DNB) and the cladding rapidly heats up (Figure 14.3.2-3) while the core power shuts down due to voiding in the core. Control rod insertion is not modeled. The hot water in the core and upper plenum flashes to steam. The water in the upper head flashes and is forced down through the guide tubes. The brrak flow becomes saturated and is substantially reduced (Figure 14.3.2-4).

At approximately 4 seconds, the pressure in the pressurizer has fallen to the point where the safety injection signals are initiated.

Upward Core Flow Phase (4-8 seconds)

The colder water in the downcomer and lower plenum flashes, and the mixture swells. Since the intact loop pump is assumed to trip at the initiation of the break, it begins to coast down and does not serve to enhance upflow cooling by pushing fluid into the core. The upflow phase is short-lived for this reason. However, there is sufficient upflow cooling to begin significantly reducing the heat up in the fuel rods. As the lower plenum fluid depletes, upflow through the core ends (Figure 14.3.2-5).

Downward Core Flow Phase (8-30 seconds)

The break flow begins to dominate and pulls flow down through the core. Figure 14.3.2-5 shows the total core flow at the core midplane. The blowdown PCT of 1654°F occurs as the downflow increases in intensity and continues to decrease while downflow is sustained. At approximately 11 seconds, the pressure in the cold leg falls to the point where accumulators begin injecting cold water into the cold legs (Figure 14.3.2-6). Because the break flow is still high, much of the accumulator emergency core cooling system (ECCS) water entering the downcomer is bypassed out of the break. As the system pressure continues to decrease, the break flow, and consequently the core flow, is reduced. The break flow further reduces and the accumulator water begins to fill the downcomer and lower plenum. The core flow is nearly stagnant during this period and the hot assembly experiences a near adiabatic heat up.

Refill Phase (30-40 seconds)

The high head safety injection (HHSI) pump begins to inject (Figure 14.3.2-7) into the cold leg at approximately 34 seconds assuming a delay time of 30 seconds after the SI signal is initiated when a loss of offsite power is assumed. Since the break flow has significantly reduced by this time, much of the ECCS entering the downcomer via the cold leg is retained in the downcomer and refills the lower plenum. The low head safety injection (LHSI) pump is assumed to begin injecting (Figure (14.3.2-8) cold ECCS water into the upper plenum at approximately 39 seconds, assuming a delay of 35 seconds for the loss of offsite power case, after the SI signal has been actuated. The water enters the vessel at the hot leg nozzle centerline elevation and falls down to the upper core plate through the outer global channels. The liquid drains down through the low power region via the open hole channel of the counter-current flow limiting (CCFL) region. The hot assembly experiences a nearly adiabatic heatup as the lower plenum fills with ECCS water (Figures 14.3.2-3 and 14.3.2-10).

Reflood (40-250 seconds)

At approximately 40 seconds, the intact loop accumulator is empty of water, and begins injecting nitrogen into the cold leg (Figure 14.3.2-6). The insurge in the downcomer forces the downcomer liquid into the lower plenum and core regions (Figures 14.3.2-9 through 14.3.2-11). During this time, core cooling is increased, and the hot assembly clad temperature decreases slightly.

The clad temperature in the hot assembly returns to nearly adiabatic heatup for about

30 seconds, until the core again begins to refill. The LHSI liquid flows down through the low power region and crossflows into the average assemblies near the bottom of the core. This water quenches the bottom of the core, which produces vapor that flows up through the average and hot assemblies, providing bottom-up cooling. The reflood PCT of 1763°F occurs at approximately 70 seconds.

Long Term Core Cooling

At the end of the WCOBRA/TRAC calculation, the core and downcomer levels are increasing as the pumped safety injection flow exceeds the break flow. The core and downcomer levels would be expected to continue to rise, until the downcomer mixture level approaches the loop elevation. At that point, the break flow would increase, until it roughly matches the injection flowrate. The core would continue to be cooled until the entire core is eventually quenched.

Confirmatory Sensitivity Studies

A number of sensitivity calculations were carried out to investigate the effect of the key LOCA parameters, and to develop the required data for the uncertainty evaluation. In the sensitivity studies performed, LOCA parameters were varied one at a time. For each sensitivity study, a comparison between the base case and the sensitivity case transient results was made.

The results of the sensitivity studies are summarized in Tables 14.3.2-5 and 14.3.2-7. A full report on the results for all sensitivity study results is included in Section 4 of Reference 13. The results of these analyses lead to the following conclusions:

1. The limiting break type is a cold leg split break, and the limiting split break area is 0.7 times the area of a cold leg pipe ($C_D = 0.7$), which is 2.88 ft^2. This split break size is then modeled in the reference transient, as well as in the subsequent calculations used in the determination of uncertainties.

2. Modeling the pressurizer on the broken loop results in a higher PCT than modeling the pressurizer on the intact loop.

3. Modeling loss-of-offsite power (LOOP) results in a higher PCT than when the reactor coolant pumps are assumed to continue to run (no-LOOP).

4. Maximum steam generator tube plugging (10%) results in the highest PCT.

5. Modeling the minimum value of vessel average temperature ($T_{avg} = 556.3$°F) results in the highest PCT.

6. Modeling the maximum power fraction ($P_{LOW} = 0.6$) in the low power/periphery channel of the core results in the highest PCT.

Initial Conditions Sensitivity Studies

Several calculations were performed to evaluate the effect of change in the initial conditions on the calculated LOCA transient. These calculations analyzed key initial plant conditions over their expected range of operation. These studies included effects of ranging RCS conditions (pressure and temperature), safety injection temperature, and accumulator conditions (pressure, temperature, volume, and line resistance). The results of these studies are presented in Section 5 of Reference 13.

The calculated results were used to develop initial condition uncertainty distributions for the blowdown and reflood peaks. These distributions are then used in the uncertainty evaluation to predict the PCT uncertainty component resulting from initial conditions uncertainty ($\Delta PCT_{IC,i}$).

Power Distribution Sensitivity Studies

Several calculations were performed to evaluate the effect of power distribution on the calculated LOCA transient. The power distribution attributes which were analyzed are the peak linear heat rate relative to the core average, the maximum relative rod power, the relative power in the bottom third of the core (P_{BOT}), and the relative power in the middle third of the core (P_{MID}). The choice of these variables and their ranges are based on the expected range of plant operation. The ranges for each of these variables can be superimposed upon a scatter plot of all possible power shapes for a typical KNPP fuel cycle (including 18-month fuel cycles). The box surrounding the power shapes encompasses the range on P_{BOT} and P_{MID} that was analyzed with this power distribution run matrix, as shown in Section 6 of Reference 13.

The power distribution parameters used for the reference transient are biased to yield a relatively high PCT. The reference transient uses the maximum $F_{\Delta H}$, a skewed to the top power distribution, and a F_Q at the midpoint of the sample range.

A run matrix was developed in order to vary the power distribution attributes singly and in combination. The calculated results are presented in Section 6 of Reference 13. The sensitivity results indicated that power distributions with peak powers shifted towards the middle of the core produced higher PCTs as a result of some steam cooling in the top of the core for UPI plants.

The calculated results were used to develop response surfaces, as described in Step 4 of Section 14.3.2.2, which could be used to predict the change in PCT for various changes in the power distributions for the blowdown and reflood peaks. These were then used in the uncertainty evaluation, to predict the PCT uncertainty component resulting from uncertainties in power distribution parameters, ($\Delta PCT_{PD,i}$).

Global Model Sensitivity Studies

Several calculations were performed to evaluate the effect of break flow path resistance and upper plenum drain distribution on the PCT for the split break with the limiting break area ($C_D = 0.7$). As in the power distribution study, these parameters were varied singly and in combination in order to obtain a data base which could be used for response surface generation. The run matrix and ranges of the break flow parameters are described in Reference 12. The limiting guillotine break was also identified using the methodology described in Reference 12. The plant specific calculated results are presented in Section 7 of Reference 13. The results of these studies indicated that the split break calculation resulted in much higher PCTs than the guillotine break calculations. Therefore, no further guillotine calculations needed to be performed.

The calculated results were used to develop response surfaces as described in Section 14.3.2.2, which could be used to predict the change in PCT for various changes in the flow conditions. These were then used in the uncertainty evaluation to predict the PCT uncertainty component resulting from uncertainties in global model parameters ($\Delta PCT_{MOD,i}$).

Uncertainty Evaluation and Results

The PCT equation was presented in Section 14.3.2.2. Each element of uncertainty is initially considered to be independent of the other. Each bias component is considered a random variable, whose uncertainty and distribution is obtained directly, or is obtained from the uncertainty of the parameters of which the bias is a function. For example, $\Delta PCT_{PD,i}$ is a function of F_Q, $F_{\Delta H}$, P_{BOT}, and P_{MID}. Its distribution is obtained by sampling the plant F_Q, $F_{\Delta H}$, P_{BOT}, and P_{MID} distributions and using a response surface to calculate $\Delta PCT_{PD,i}$. Since ΔPCT_i is the sum of these biases, it also becomes a random variable. Separate initial PCT frequency distributions are constructed as follows for the split break and the limiting guillotine break size:

1. Generate a random value of each ΔPCT element.

2. Calculate the resulting PCT using Equation 14.3.2-1.

3. Repeat the process many times to generate a histogram of PCTs.

A final verification step is performed in which additional calculations (known as "superposition" calculations) are made with WCOBRA/TRAC, simultaneously varying several parameters which were previously assumed independent (for example, power distributions and models). Predictions using Equation 14.3.2-1 are compared to this data, and additional biases and uncertainties are applied.

The estimate of the PCT at 95% probability is determined by finding that PCT below which 95% of the calculated PCTs reside. This estimate is the licensing basis PCT, under the revised ECCS rule.

The results for the Kewaunee Nuclear Power Plant are given in Table 14.3.2-8, which shows the reflood 95th percentile PCT ($PCT^{95\%}$) of 2084°F. As expected, the difference between the 95% value and the average value increases with increasing time, as more parameter uncertainties come into play.

Evaluations

The transition from Siemens Standard/Heavy Fuel Assemblies to Westinghouse 14x14 Vantage+ fuel with Performance+ features (422V+) fuel has been evaluated for the effects of hydraulic mismatch and differences in fuel designs. The Reference Transient for the KNPP was used to determine the transition core effects.

Two additional calculations were performed for this assessment. For one calculation, the hot assembly was modeled with the fresh 422V+ fuel, surrounded by Siemens Heavy fuel, once burned. For the second calculation, the hot assembly was modeled with Siemens Heavy fuel (once burned), surrounded by fresh 422V+ assemblies. In both calculations, the low power/peripheral region was modeled with Siemens Heavy fuel (once burned). The results of the assessment indicate that the Best-Estimate analysis with a full core of 422V+ fuel for the KNPP bounds the transition core cycles.

Large Break LOCA Conclusions

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The demonstration that these limits are met for the Kewaunee Nuclear Power Plant is as follows:

1. There is a high level of probability that the peak cladding temperature (PCT) shall not exceed 2200°F. The results presented in Table 14.3.2-8 indicate that this regulatory limit has been met with a reflood $PCT^{95\%}$ of 2084°F.

2. The maximum calculated local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. The approved Best-Estimate LOCA methodology assesses this requirement using a plant-specific transient which has a PCT in excess of the estimated 95 percentile PCT ($PCT^{95\%}$). Based on this conservative calculation, a maximum local oxidation of 8.44% is calculated, which meets the regulatory limit.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel were to react. The total amount of hydrogen generated, based on this conservative assessment is 0.0074 times the maximum theoretical amount, which meets the regulatory limit.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling. This requirement is met by demonstrating that the PCT does not exceed 2200°F, the maximum local oxidation does not exceed 17%, and the seismic and LOCA forces are not sufficient to distort the fuel assemblies to the extent that the core cannot be cooled. The BE UPI methodology (References 11 and 12) specifies that the effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crush

extends to in-board assemblies. Fuel assembly structural analyses performed for Kewaunee indicate that this condition does not occur. Therefore, this regulatory limit is met.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core. The conditions at the end of the WCOBRA/TRAC calculations indicates that the transition to long term cooling is underway even before the entire core is quenched.

SER Requirements

The SER requirements for three- and four-loop plants (Reference 11) have been met for this KNPP analysis. The BE UPI Evaluation Model has additional requirements to verify that the plant conditions fall within the range of conditions represented by the test simulations used for assessment of phenomena unique to upper plenum injection plants (Reference 15). Table 14.3.2-9 compares the plant conditions for KNPP to the test conditions utilized in the BE UPI methodology (References 12). From this table, it is clear that KNPP conditions fall within the range of test conditions. Thus, the BE UPI SER requirements have been met for Kewaunee.

Plant Operating Range

The expected PCT and its uncertainty developed above is valid for a range of plant operating conditions. In contrast to current Appendix K calculations, many parameters in the base case calculation are at nominal values. The range of variation of the operating parameters has been accounted for in the estimated PCT uncertainty. Table 14.3.2-10 and Figures 14.3.2-12 through 14.3.2-14 summarize the operating ranges for the Kewaunee Nuclear Power Plant. If operation is maintained within these ranges, the LOCA analysis developed in Reference 13 is considered to be valid.

14.3.3 CORE AND INTERNALS INTEGRITY ANALYSIS

The response of the reactor core and vessel internals under excitation produced by a simultaneous complete severance of a reactor coolant pipe and seismic excitation of typical two loop plant internals has been determined. A detailed description of the analysis applicable to the Kewaunee Nuclear Power Plant design appears in WCAP 7822, (Reference 1) Indian Point Unit 2 Reactor Internals Mechanical Analysis for Blowdown Excitation (Westinghouse Proprietary).

See Steam Generator Replacement and Tavg Operating Window Program Licensing Report (Reference 7), Section 5.2 for Updates to Internals Qualifications.

Reactor Internals Response Under Blowdown and Seismic Excitation

A loss-of-coolant accident may result from a rupture of reactor coolant piping. During the blowdown of the coolant, critical components of the core are subjected to vertical and horizontal excitation as a result of rarefaction waves propagating inside the reactor vessel.

For these large breaks, the reduction in water density greatly reduces the reactivity of the core, thereby shutting down the core whether the rods are tripped or not. (The subsequent refilling of the core by the Emergency Core Cooling System uses borated water to maintain the core in a sub-critical state.) Therefore, the main requirement is to assure effectiveness of the Emergency Core Cooling System. Insertion of the control rods, although not needed, gives further assurance of ability to shut the plant down and keep it in a safe shutdown condition.

The pressure waves generated within the reactor are highly dependent on the location and nature of the postulated pipe failure. In general, the more rapid the severance of the pipe, the more severe the imposed loadings on the components. A one-millisecond severance time is taken as the limiting case.

In the case of the hot leg break, the vertical hydraulic forces produce an initial upward lift of the core. A rarefaction wave propagates through the reactor hot leg nozzle into the interior of the upper core barrel. Since the wave has not reached the flow annulus on the outside of the barrel, the upper barrel is subjected to an impulsive compressive wave. Thus, dynamic instability (buckling) or large deflection of the upper core barrel or both is the possible response of the barrel during hot leg blowdown. In addition to the above effects, the hot leg break results in transverse loading on the upper core components as the fluid exits the hot leg nozzle.

In the case of the cold leg break, a rarefaction wave propagates along a reactor inlet pipe arriving first at the core barrel at the inlet nozzle of the broken loop. The upper barrel is then subjected to a nonaxisymmetric expansion radial impulse, which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel. After the cold leg break, the initial steady-state hydraulic lift forces (upward) decrease rapidly (within a few milliseconds) and then increase in the downward direction. These cause the reactor core and lower support structure to move initially downward.

If a simultaneous seismic event with the intensity of the design basis earthquake (DBE) is postulated with the loss-of-coolant accident, the imposed loading on the internals component maybe additive in certain cases, and therefore, the combined loading must be considered. In general, however, the loading imposed by the earthquake is small compared to the blowdown loading.

Acceptance Criteria for Results of Analyses

The criteria for acceptability in regard to mechanical integrity analysis is that adequate core cooling and core shutdown must be assured. This implies that the deformation of the reactor internals must be sufficiently small so that the geometry remains substantially intact. Consequently, the limitations established on the internals are concerned principally with the maximum allowable deflections and/or stability of the parts, in addition to a stress criterion, to assure integrity of the components.

Allowable Deflection and Stability Criteria

Upper Barrel - The upper barrel deformation has the following limits:

a) To insure a shutdown and cooldown of the core during blowdown, the basic requirement is a limitation on the outward deflection of the barrel at the locations of the inlet nozzles connected to the unbroken lines. A large outward deflection of the barrel in front of the inlet nozzles, accompanied with permanent strains, could close the inlet area and stop the cooling water coming from the accumulators. (The remaining distance between the barrel and the vessel inlet nozzle after the accident must be such that the inlet flow area be approximately the same as that of the accumulator pipes). Consequently, a permanent barrel deflection in front of the unbroken inlet nozzles larger than a certain limit called the "no-loss of function" limit, could impair the efficiency of the Emergency Core Cooling System.

b) To assure rod insertion and to avoid disturbing the Control Rod Cluster guide structure, the barrel should not interfere with the guide tubes. This condition also requires a stability check to assure that the barrel will not buckle under the accident loads.

Control Rod Cluster Guide Tubes - The guide tubes in the upper core support package house the control rods. The deflection limits were established from tests.

Fuel Assembly - The limitations for this case are related to the stability of the thimbles in the upper end. The upper end of the thimbles shall not experience stresses above the allowable dynamic compressive stresses. Any buckling of the upper end of the thimbles due to axial compression could distort the guideline and thereby affect the free fall of the control rod.

Upper Package – The maximum allowable local deformation of the upper core plate where a guide tube is located is 0.100 inch. This deformation will cause the plate to contact the guide tube since the clearance between plate and guide tube is 0.100 inches. This limit will prevent the guide tubes from undergoing compression. For a plate local deformation of 0.150 inches, the guide tube will be compressed and deformed transversely to the upper limit previously established; consequently, the value of 0.150 inches is adopted as the no loss-of-function local deformation, with an allowable limit of 0.100 inches.

Allowable Stress Criteria

For this faulted condition, the allowable stress criteria is given by Figure 14.3-21a. This figure defines various criteria based upon their corresponding method of analysis.

To account for multi-axial stresses, the Von Mises Theory is also considered.

Method of Analysis

Blowdown Model

BLODWN-2 is a digital computer program developed for the purpose of calculating local fluid pressure, flow, and density transients that occur in PWR coolant systems during a loss-of-coolant accident (Reference 2). This program applies to the sub-cooled, transition, and saturated two-phase blowdown regimes. BLODWN-2 is based on the method of characteristics wherein the resulting set of ordinary differential equations, obtained from the laws of conservation of mass, momentum, and energy, are solved numerically using a fixed mesh in both space and time.

Although spatially one-dimensional conservation laws are employed, the code can be applied to describe three-dimensional system geometries by use of the equivalent piping networks. Such piping networks may contain any number of pipes or channels of various diameters, dead ends, branches (with up to six pipes connected to each branch), contractions, expansions, orifices, pumps, and free surfaces (such as in the pressurizer). System losses such as friction, contraction, expansion, etc., are considered.

BLODWN-2 predictions have been compared with numerous test data as reported in WCAP-7401 (Reference 4). It is shown that the BLODWN-2 digital computer program correlates well with both the sub-cooled and the saturated blowdown regimes.

FORCE Model for Blowdown

BLODWN-2 evaluates the pressure and velocity transients for a maximum of 2400 locations throughout the system. These pressure and velocity transients are stored as a permanent tape file and are made available to the program FORCE which utilizes a detailed geometric description in evaluating the loadings on the reactor internals.

Each reactor component for which FORCE calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated summing the effects of:

1) The pressure differential across the element.

2) Flow stagnation on, and unrecovered orifice losses across the element.

3) Friction losses along the element.

Input to the code, in addition to the BLODWN-2 pressure and velocity transients includes the effective area of each element on which the force acts, due to the pressure differential across the element, a coefficient to account for flow stagnation, unrecovered orifice losses, and the total area of the element along which the shear forces act.

The mechanical analysis has been performed using conservative assumptions in order to obtain results with extra margin. Some of the most significant are:

a) The mechanical and hydraulic analysis has been performed separately without including the effect of the water-solid interaction. Peak pressures obtained from the hydraulic analysis will be attenuated by the deformation of the structures.

b) When applying the hydraulic forces, no credit is taken for the stiffening effect of the fluid environment, which will reduce the deflections and stresses in the structure.

c) The multi-mass model described below is considered to have a sufficient number of degrees of freedom to represent the most important modes of vibration in the vertical direction. This model is conservative in the sense that further mass-spring resolution of the system would lead to further attenuation of the shock effects obtained with the present model.

Method of Blowdown Re-Analysis

Re-analysis performed in support of increased full power primary coolant temperature range and new fuel products (such as Siemens-Designed 14x14 fuel) have made use of the MULTIFLEX (References 5 and 6) computer code, rather than BLODWN-2 described above. These analyses use the FORCE2 computer code (described in Reference 6) to post process MULTIFLEX hydraulic transient results into vertical forces as described above for the FORCE code. Lateral forces are computed using the LATFORC code (described in Reference 6). Additional details of these re-analysis are found in Section 6.5 of the Steam Generator Replacement and Tavg Operating Window Program Licensing Report (Reference 7). As described in Section 5.2 of Reference 7, the core and internals integrity calculations found the re-analysis loads to remain bounded by the calculations described above.

Vertical Excitation Model for Blowdown

For the vertical excitation, the reactor internals are represented by a multi-mass system connected with springs and dashpots simulating the elastic response and the viscous damping of the components. Also incorporated in the multi-mass system is a representation of the motion of the fuel elements relative to the fuel assembly grids. The fuel elements in the fuel assemblies are kept in position by friction forces originating from the preloaded fuel assembly grid fingers. Coulomb-type friction is assumed in the event that sliding between the rods and the grid fingers occurs. Figure 14.3-21 shows the spring-mass system used to represent the internals. In order to obtain an accurate simulation of the reactor internals response, the effects of internal damping, clearances between various internals, snubbing action caused by solid impact, Coulomb friction induced by fuel rods motion relative to the grids, and pre-loads in holddown springs have been incorporated in the analytical model. The reactor vessel is regarded as a fixed base while the internals undergo relative displacement with respect to their initial position. The modeling is conducted in such a way that uniform masses are lumped into easily identifiable discrete masses while elastic elements are represented by springs. Table 14.3-3 lists the various masses, springs, etc.

The appropriate dynamic differential equations for the multi-mass model describing the aforementioned phenomena are formulated and the results obtained using a digital computer program which computes the response of the multi-mass model when excited by a set of time dependent forcing functions. The appropriate forcing functions are applied simultaneously and independently to each of the masses in the system. The results from the program give the forces, displacements and deflections as functions of time for all the reactor internals components (lumped masses). Reactor internals response to both hot and cold leg pipe ruptures are analyzed. The forcing functions used in the study are obtained from hydraulic analyses of the pressure and flow distribution around the entire reactor coolant system as caused by double-ended severance of a reactor coolant system pipe.

Vertical Excitation Model for Earthquake

As shown in WCAP-7822 (Reference 1) the reactor internals are modeled as a single degree-of-freedom system for vertical earthquake analysis. The maximum acceleration at the vessel support is increased by amplification due to the building-soil interaction.

Transverse Excitation Model for Blowdown

Various reactor internal components are subjected to transverse excitation during blowdown. Specifically, the barrel, guide tubes, and upper support columns are analyzed to determine their response to this excitation.

Core Barrel

For the hydraulic analysis of the pressure transients during hot leg blowdown, the maximum pressure drop across the barrel is a uniform radial compressive impulse. The barrel is then analyzed for dynamic buckling using these conditions and the following conservative assumptions:

a) The effect of the fluid environment is neglected (water stiffening is not considered);

b) The shell is treated as simply supported.

During cold leg blowdown, the upper barrel is subjected to a nonaxisymmetric expansion radial impulse, which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel.

The analysis of transverse barrel response to cold leg blowdown is performed as follows:

1) The upper core barrel is treated as a simply supported cylindrical shell of constant thickness between the upper flange weldment and the lower core barrel weldment without taking credit for the supports at the barrel mid-span offered by the outlet nozzles. This assumption leads to conservative deflection estimates of the upper core barrel.

2) The upper core barrel is analyzed as a shell with four variable sections to model the support flange, upper barrel, reduced weld section, and a portion of the lower core barrel.

3) The barrel with the core and thermal shield is analyzed as a beam fixed at the top and elastically supported at the lower radial support and the dynamic response is obtained.

Guide Tubes - The dynamic loads on RCC guide tubes are more severe for a loss-of-coolant accident caused by hot leg rupture than for an accident by cold leg rupture since the cold leg break leads to much smaller changes in the transverse coolant flow over the RCC guide tubes. Thus, the analysis is performed only for a hot leg blowdown.

The guide tubes in closest proximity to the ruptured outlet nozzle are the most severely loaded. The transverse guide tube forces during the hot leg blowdown decrease with increasing distance from the ruptured nozzle location.

A detailed structural analysis of the RCC guide tubes was performed to establish the equivalent cross-section properties and elastic and support conditions. An analytical model was verified both dynamically and statically by subjecting the control rod cluster guide tube to a concentrated force applied at the transition plate. In addition, the guide tube was loaded experimentally using a triangular distribution to conservatively approximate the hydraulic loading. The experimental results consisted of a load deflection curve for the RCC guide tube plus verification of the deflection criteria to assure RCC insertion.

The response of the guide tubes to the transient loading due to blowdown may be found by utilizing the equivalent single freedom system for the guide tube using experimental results for equivalent stiffness and natural frequency.

The time dependence of the hydraulic transient loading has the form of a step function with constant slope front with a rise time to peak force of the same order of the guide tube fundamental period in water. The dynamic application factor in determining the response is a function of the ramp impulse rise time divided by the period of the structure.

Upper Support Columns - Upper support columns located close to the broken nozzle during hot leg break will be subjected to transverse loads due to cross flow.

The loads applied to the columns were computed with a similar method to the one for the guide tubes, i.e., taking into consideration the increase in flow across the column during the accident. The columns were studied as beams with variable section and the resulting stresses were obtained using the reduced section modulus at the slotted portions.

Transverse Excitation Model for Earthquake

The reactor building with the reactor vessel support, the reactor vessel, and the reactor internals are included in this analysis. The mathematical model of the building, attached to ground, is identical to that used to evaluate the building structure. The reactor internals are mathematically modeled by beams, concentrated masses and linear springs.

All masses, water and metal are included in the mathematical model. All beam elements have the component weight or mass distribution uniformly, e.g., the fuel assembly-mass and barrel mass. Additionally, wherever components are attached uniformly their mass is included as an additional uniform mass, e.g., baffles and formers acting on the core barrel. The water near and about the beam elements is also included as a distributed mass. Horizontal components are considered as a concentrated mass acting on the barrel. This concentrated mass also includes components attached to the horizontal members, since these are the media through which the reaction is transmitted. The water near and about these separated components is considered as being additive at these concentrated mass points.

The concentrated masses attached to the barrel represent the following:

- the upper core support structure, including the upper vessel head and one-half the upper internals;
- the upper core plate, including one-half the thermal shield and the other one-half of the upper internals;
- the lower core plate, including one-half of the lower core support columns;
- the lower one-half of the thermal shield, and
- the lower core support, including the lower instrumentation and the remaining half of the lower core support columns.

The modulus of elasticity is chosen at its hot value for the three major materials found in the vessel, internals, and fuel assemblies. In considering shear deformation, the appropriate cross-sectional area is selected along with a value for Poisson's ratio. The fuel assembly moment of inertia is derived from experimental results by static and dynamic tests performed on fuel assembly modes. These tests provide stiffness values for use in this analysis.

The fuel assemblies are assumed to act together and are represented by a single beam. The following assumptions are made in regard to connection restraints. The vessel is pinned to the vessel support and part of the containment building. The barrel is clamped to the vessel at the barrel flange and spring-connected to the vessel at the lower core barrel radial support. This spring corresponds to the radial support stiffness for two opposite supports acting together. The beam representing the fuel assemblies is pinned to the barrel at the locations of the upper and lower core plates.

The response spectrum method has been used in the calculation. After computing the transverse natural frequency and obtaining the normal modes of the complete structure, the maximum response is obtained from the superposition of the usual mode response with the conservative assumptions that all the modes are in phase and that all the peaks occur simultaneously.

Conclusions - Mechanical Analysis

The results of the analysis applicable to the Kewaunee design are presented in Table 14.3-3a and Table 14.3-3b. These tables summarize the maximum deflections and stresses for blowdown, seismic, and blowdown plus seismic loadings.

The stresses due to the DBE (vertical and horizontal components) were combined in the most unfavorable manner with the blowdown stresses in order to obtain the largest principal stress and deflection.

These results indicate that the maximum deflections and stress in the critical structures are below the established allowable limits. For the transverse excitation, it is shown that the upper barrel does not buckle during a hot leg break and that it has an allowable stress distribution during a cold leg break. Section 5.9, under Primary Piping, discusses the restraints which were added to restrain the reactions of jet forces in the primary loop piping caused by pipe rupture, to limit the LOCA loads.

Even though control rod insertion is not required for plant shutdown, this analysis shows that most of the guide tubes will deform within the limits established experimentally to assure control rod insertion with the exceptions shown in Table 14.3-3a. It can be seen in the Table that 31 of the 33 guide tubes are below the NLF limit. For those guide tubes deflected above the NLF limit, it must be assumed that the rods will not drop. However, the conclusion reached is that the core will shutdown in an orderly fashion due to the formation of voids, and this orderly shutdown will be aided by the great majority of rods that do drop.

14.3.4 CONTAINMENT INTEGRITY EVALUATION

14.3.4.1 Long Term LOCA Mass and Energy Releases

The uncontrolled release of pressurized high-temperature reactor coolant, termed a loss-of-coolant accident (LOCA), will result in release of steam and water into the containment. This, in turn, will result in increases in the local subcompartment pressures, and an increase in the global containment pressure and temperature. Therefore, there are both long- and short-term issues reviewed relative to a postulated LOCA that must be considered at the conditions for the Kewaunee Nuclear Power Plant (KNPP) with Model 54F steam generators.

The long-term LOCA mass and energy releases are analyzed to approximately 10^6 seconds and are utilized as input to the containment integrity analysis. This demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a hypothetical large-break LOCA. The containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure. For this program, Westinghouse generated the mass and energy releases using the March 1979 model, described in Reference 1. The Nuclear Regulatory Commission (NRC) review and approval letter is included with Reference 1. The following sections discuss the long-term LOCA mass and energy releases. The results of this analysis were provided for use in the containment integrity analysis.

Input Parameters and Assumptions

The mass and energy release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. For example, the RCS operating temperatures are chosen to bound the highest average coolant temperature range of all operating cases and a temperature uncertainty allowance of (+4.0°F) is then added. Nominal parameters are used in certain instances. For example, the RCS pressure in this analysis is based on a nominal value of 2250 psia plus an uncertainty allowance (+30 psi). All input parameters are chosen consistent with accepted analysis methodology.

Some of the most critical items are the RCS initial conditions, core decay heat, safety injection flow, and primary and secondary metal mass and steam generator heat release modeling. Specific assumptions concerning each of these items are discussed in the following paragraphs. Tables 14.3.4-1 through 14.3.4-3 present key data assumed in the analysis.

The core rated power of 1683 MWt adjusted for calorimetric error (i.e., 102% of 1650 MWt) was used in the analysis. As previously noted, the use of RCS operating temperatures to bound the highest average coolant temperature range were used as bounding analysis conditions. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures that are at the maximum levels attained in steady-state operation. Additionally, an allowance to account for instrument error and deadband is reflected in the initial RCS temperatures. The selection of 2250 psia as the limiting pressure is considered to affect the blowdown phase results only, since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally the RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Thus, 2250 psia plus uncertainty was selected for the initial pressure as the limiting case for the long-term mass and energy release calculations.

The selection of the fuel design features for the long-term mass and energy release calculation is based on the need to conservatively maximize the energy stored in the fuel at the beginning of the postulated accident (i.e., to maximize the core stored energy). The core stored energy that was selected to bound the Siemens Power Corporation (SPC) fuel product that is currently in Kewaunee was 6.4 full power seconds (FPS), which is equivalent to the value for standard Westinghouse 14 x 14 fuel at beginning-of-life (BOL) conditions. The margins in the core stored energy include +15% in order to address the thermal fuel model and associated manufacturing uncertainties and the time in the fuel cycle for maximum fuel densification. Thus, the analysis very conservatively accounts for the stored energy in the core.

Margin in RCS volume of 3% (which is composed of 1.6% allowance for thermal expansion and 1.4% allowance for uncertainty) was modeled.

A uniform steam generator tube plugging level of 0% was modeled. This assumption maximizes the reactor coolant volume and fluid release by virtue of consideration of the RCS fluid in all steam generator tubes. During the post-blowdown period, the steam generators are active heat sources since significant energy remains in the secondary metal and secondary mass that has the potential to be transferred to the primary side. The 0% tube plugging assumption maximizes the heat transfer area and, therefore, the transfer of secondary heat across the steam generator tubes. Additionally, this assumption reduces the reactor coolant loop resistance, which reduces the ΔP upstream of the break for the pump suction breaks and increases break flow. Thus, the analysis conservatively accounts for the level of steam generator tube plugging.

The secondary-to-primary heat transfer is maximized by assuming conservative heat transfer coefficients. This conservative energy transfer is ensured by maximizing the initial internal energy of the inventory in the steam generator secondary side. This internal energy is based on full-power operation plus uncertainties.

Regarding safety injection flow, the mass and energy release calculation considered configurations/failures to conservatively bound respective alignments. The cases include: (a) a minimum safeguards case (1 high-head safety injection (HHSI) and 1 low-head safety injection (LHSI) pumps) (see Table 14.3.4-2); and (b) a maximum safeguards case, (2 HHSI and 2 LHSI pumps) (see Table 14.3.4-3). In addition, the containment backpressure is assumed to be equal to the containment design pressure. This assumption was shown in Reference 1 to be conservative for the generation of mass and energy releases.

In summary, the following assumptions were employed to ensure that the mass and energy releases are conservatively calculated, thereby maximizing energy release to containment:

1. Maximum expected operating temperature of the RCS (100% full-power conditions)

2. Allowance for RCS temperature uncertainty (+4.0°F)

3. Margin in RCS volume of 3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% allowance for uncertainty)

4. Core rated power of 1650 MWt

5. Allowance for calorimetric error (+2% of power)

6. Conservative heat transfer coefficients (i.e., steam generator primary/secondary heat transfer, and RCS metal heat transfer)

7. Allowance in core stored energy for effect of fuel densification

8. A margin in core stored energy (+15% to account for manufacturing tolerances)

9. An allowance for RCS initial pressure uncertainty (+30 psi)

10. A maximum containment backpressure equal to design pressure (46.0 psig)

11. Steam generator tube plugging leveling (0% uniform)
    - Maximizes reactor coolant volume and fluid release
    - Maximizes heat transfer area across the steam generator tubes
    - Reduces coolant loop resistance, which reduces the ΔP upstream of the break for the pump suction breaks and increases break flow

Thus, based on the previously discussed conditions and assumptions, an analysis of Kewaunee was made for the release of mass and energy from the RCS in the event of a LOCA at 1683 MWt.

Description of Analyses

The evaluation model used for the long-term LOCA mass and energy release calculations is the March 1979 model described in Reference 1.

This report section presents the long-term LOCA mass and energy releases generated in support of the Kewaunee SGR Program. These mass and energy releases are then subsequently used in the containment integrity analysis.

The mass and energy release rates described form the basis of further computations to evaluate the containment following the postulated accident. Discussed in this section are the long-term LOCA mass and energy releases for the hypothetical double-ended pump suction (DEPS) rupture with minimum safeguards and maximum safeguards and double-ended hot leg (DEHL) rupture break cases. The mass and energy releases for these three cases are shown in Tables 14.3.4-4 through 14.3.4-18. These three LOCA cases are used for the long-term containment integrity analyses in Section 14.3.4.2.

LOCA Mass and Energy Release Phases

The containment system receives mass and energy releases following a postulated rupture in the RCS. These releases continue over a time period, which, for the LOCA mass and energy analysis, is typically divided into four phases.

1. Blowdown – the period of time from accident initiation (when the reactor is at steady-state operation) to the time that the RCS and containment reach an equilibrium state.

2. Refill – the period of time when the lower plenum is being filled by accumulator and emergency core cooling system (ECCS) water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.

3. Reflood – begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Post-reflood (Froth) – describes the period following the reflood phase. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators prior to exiting the break as steam. After the broken loop steam generator cools, the break flow becomes two phase.

Computer Codes

The Reference 1 mass and energy release evaluation model is comprised of mass and energy release versions of the following codes: SATAN VI, WREFLOOD, FROTH, and EPITOME. These codes were used to calculate the long-term LOCA mass and energy releases for Kewaunee.

SATAN VI calculates blowdown, the first portion of the thermal-hydraulic transient following break initiation, including pressure, enthalpy, density, mass and energy flow rates, and energy transfer between primary and secondary systems as a function of time.

The WREFLOOD code addresses the portion of the LOCA transient where the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and when water supplied by the ECCS refills the reactor vessel and provides cooling to the core. The most important feature of WREFLOOD is the steam/water mixing model.

FROTH models the post-reflood portion of the transient. The FROTH code is used for the steam generator heat addition calculation from the broken and intact loop steam generators.

EPITOME continues the FROTH post-reflood portion of the transient from the time at which the secondary equilibrates to containment design pressure to the end of the transient. It also compiles a summary of data on the entire transient, including formal instantaneous mass and energy release tables and mass and energy balance tables with data at critical times.

Break Size and Location

Generic studies have been performed with respect to the effect of postulated break size on the LOCA mass and energy releases. The double-ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the RCS loop can be postulated for a pipe rupture for mass and energy release purposes:

- Hot leg (between vessel and steam generator)
- Cold leg (between pump and vessel)
- Pump suction (between steam generator and pump)

The break locations analyzed for this program are the DEPS rupture (10.46 ft^2) and the DEHL rupture (9.154 ft^2). Break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA for the DEPS cases. For the DEHL case, the releases were calculated only for the blowdown. The following information provides a discussion on each break location.

The DEHL rupture has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be the highest for this break location, the amount of energy released from the steam generator secondary is minimal because the majority of the fluid that exits the core vents directly to containment bypassing the steam generators. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break. For the hot leg break, generic studies have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the containment pressure would continually decrease). Therefore, only the mass and energy releases for the hot leg break blowdown phase are calculated and presented in this section of the report.

The cold leg break location has also been found in previous studies to be much less limiting in terms of the overall containment energy releases. The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that for the pump suction break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the cold leg break is bounded by other breaks and no further evaluation is necessary.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the RCS in calculating the releases to containment. Thus, only the DEHL and DEPS cases are used to analyze long-term LOCA containment integrity.

Application of Single-Failure Criterion

An analysis of the effects of the single-failure criterion has been performed on the mass and energy release rates for each break analyzed. An inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the safety injection system. This is not an issue for the blowdown period, which is limited by the DEHL break.

Two cases have been analyzed to assess the effects of a single failure. The first case assumes minimum safeguards safety injection (SI) flow based on the postulated single failure of an emergency diesel generator. This results in the loss of one train of safeguards equipment. The other case assumes maximum safeguards SI flow based on no postulated failures that would impact the amount of ECCS flow. The analysis of the cases described provides confidence that the effect of credible single failures is bounded.

Acceptance Criteria for Analyses

A large break loss-of-coolant accident is classified as an American Nuclear Society (ANS) Condition IV event, an infrequent fault. To satisfy the NRC acceptance criteria presented in the Standard Review Plan, Section 6.2.1.3, the relevant requirements are the following:

- 10 CFR 50, Appendix A
- 10 CFR 50, Appendix K, paragraph I.A

To meet these requirements, the following must be addressed:

- Sources of energy
- Break size and location
- Calculation of each phase of the accident

Blowdown Mass and Energy Release Data

The SATAN-VI code is used for computing the blowdown transient. The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is incorporated into the analysis. The methodology for the use of this model is described in Reference 1.

Table 14.3.4-4 presents the calculated mass and energy release for the blowdown phase of the DEHL break for Kewaunee. For the hot leg break mass and energy release tables, break path 1 refers to the mass and energy exiting from the reactor vessel side of the break; break path 2 refers to the mass and energy exiting from the steam generator side of the break.

Table 14.3.4-7 presents the calculated mass and energy releases for the blowdown phase of the DEPS break. For the pump suction breaks, break path 1 in the mass and energy release tables refers to the mass and energy exiting from the steam generator side of the break. Break path 2 refers to the mass and energy exiting from the pump side of the break.

Reflood Mass and Energy Release Data

The WREFLOOD code is used for computing the reflood transient. The WREFLOOD code consists of two basic hydraulic models — one for the contents of the reactor vessel and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena such as pumped safety injection and accumulators, reactor coolant pump performance, and steam generator release are included as auxiliary equations that interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flow rates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break, i.e., the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and ECCS injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the usage and application of the Reference 1 mass and energy release evaluation model in recent analyses, e.g., D. C. Cook Docket (Reference 3). Even though the Reference 1 model credits steam/water mixing only in the intact loop and not in the broken loop, the justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 3). Moreover, this assumption is supported by test data and is further discussed below.

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two-phase interaction with condensation of steam by cold ECCS water. The second is a single-phase mixing of condensate and ECCS water. Since the steam release is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data have been reviewed for validation of the containment integrity reflood steam/water mixing model. This data was generated in

1/3-scale tests (Reference 4), which are the largest scale data available and thus most clearly simulates the flow regimes and gravitational effects that would occur in a pressurized water reactor (PWR). These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

A group of 1/3-scale tests corresponds directly to containment integrity reflood conditions. The injection flow rates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 1. For all of these tests, the data clearly indicate the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is, therefore, wholly supported by the 1/3-scale steam/water mixing data.

Additionally, the following justification is also noted. The post-blowdown limiting break for the containment integrity peak pressure analysis is the pump suction double-ended rupture break. For this break, there are two flow paths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam that is not condensed by ECCS injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECCS injection water as it passes through the broken loop cold leg, complete mixing occurs and a portion of it is condensed. It is this portion of steam that is condensed that is taken credit for in this analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECCS injection nozzle. A description of the test and test results are contained in References 1 and 3.

Tables 14.3.4-8 and 14.3.4-13 present the calculated mass and energy releases for the reflood phase of the pump suction double-ended rupture, minimum safeguards, and maximum safeguards cases, respectively.

The transient response of the principal parameters during reflood are given in Tables 14.3.4-9 and 14.3.4-14 for the DEPS cases.

Post-Reflood Mass and Energy Release Data

The FROTH code (Reference 2) is used for computing the post-reflood transient. The FROTH code calculates the heat release rates resulting from a two-phase mixture present in the steam generator tubes. The mass and energy releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken loop and intact loop steam generators. During this phase of the transient, the RCS has equilibrated with the containment pressure. However, the steam generators contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to core decay heat. For a pump suction break, a two-phase fluid exits the core, flows through the hot legs, and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two phase. During the

FROTH calculation, ECCS injection is addressed for both the injection phase and the recirculation phase. The FROTH code calculation stops when the secondary side equilibrates to the saturation temperature (T_{sat}) at the containment design pressure, after this point the EPITOME code completes the steam generator depressurization.

The methodology for the use of this model is described in Reference 1. The mass and energy release rates are calculated by FROTH and EPITOME until the time of containment depressurization. After containment depressurization (14.7 psia), the mass and energy release available to containment is generated directly from core boil-off/decay heat.

Tables 14.3.4-10 and 14.3.4-15 present the two-phase post-reflood mass and energy release data for the pump suction double-ended break cases.

Decay Heat Model

On November 2, 1978, the Nuclear Power Plant Standards Committee (NUPPSCO) of the ANS approved ANS Standard 5.1 (Reference 5) for the determination of decay heat. This standard was used in the mass and energy release model for Kewaunee. Table 14.3.4-18 lists the decay heat curve used in the Kewaunee steam generator replacement mass and energy release analysis.

Significant assumptions in the generation of the decay heat curve for use in the LOCA mass and energy releases analysis include the following:

1. The decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

2. The decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.

3. The fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Reference 5.

5. The fuel has been assumed to be at full power for 10^8 seconds.

6. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

7. Two sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

Based upon NRC staff review, (Safety Evaluation Report [SER] of the March 1979 evaluation model [Reference 1]), use of the ANS Standard-5.1, November 1979 decay heat model was approved for the calculation of mass and energy releases to the containment following a LOCA.

Steam Generator Equilibration and Depressurization

Steam generator equilibration and depressurization is the process by which secondary-side energy is removed from the steam generators in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature is the saturation temperature (T_{sat}) at the containment design pressure. After the FROTH calculations, the EPITOME code continues the FROTH calculation for steam generator cooldown removing steam generator secondary energy at different rates (i.e., first- and second-stage rates). The first-stage rate is applied until the steam generator reaches T_{sat} at the user specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the actual containment pressure. Then the second-stage rate is used until the final depressurization, when the secondary reaches the reference temperature of T_{sat} at 14.7 psia, or 212°F. The heat removal of the broken loop and intact loop steam generators are calculated separately.

During the FROTH calculations, steam generator heat removal rates are calculated using the secondary-side temperature, primary-side temperature and a secondary-side heat transfer coefficient determined using a modified McAdam's correlation. Steam generator energy is removed during the FROTH transient until the secondary-side temperature reaches saturation temperature at the containment design pressure. The constant heat removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The steam generator energy available to be released during the first stage interval is determined by calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user-specified intermediate equilibration pressure, assuming saturated conditions. This energy is then divided by the first-stage energy removal rate, resulting in an intermediate equilibration time. At this time, the rate of energy release drops substantially to the second-stage rate. The second-stage rate is determined as the fraction of the difference in secondary energy available between the intermediate equilibration and final depressurization at 212°F, and the time difference from the time of the intermediate equilibration to the user-specified time of the final depressurization at 212°F. With current methodology, all of the secondary energy remaining after the intermediate equilibration is conservatively assumed to be released by imposing a mandatory cooldown and subsequent depressurization down to atmospheric pressure at 3600 seconds, i.e., 14.7 psia and 212°F (the mass and energy balance tables have this point labeled as "Available Energy").

Sources of Mass and Energy

The sources of mass considered in the LOCA mass and energy release analysis are given in Tables 14.3.4-5, 14.3.4-11, and 14.3.4-16. These sources are the RCS, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA mass and energy release analysis are given in Tables 14.3.4-6, 14.3.4-12, and 14.3.4-17. The energy sources are the following:

- Reactor coolant system water
- Accumulator water (both inject)
- Pumped safety injection water
- Decay heat
- Core-stored energy
- Reactor coolant system metal (includes steam generator tubes)
- Steam generator metal (includes transition cone, shell, wrapper, and other internals)
- Steam generator secondary energy (includes fluid mass and steam mass)
- Secondary transfer of energy (feedwater into and steam out of the steam generator secondary)

The analysis used the following energy reference points:

- Available energy: 212°F; 14.7 psia [energy available that could be released]
- Total energy content: 32°F; 14.7 psia [total internal energy of the RCS]

The mass and energy inventories are presented at the following times, as appropriate:

- Time zero (initial conditions)
- End of blowdown time
- End of refill time
- End of reflood time
- Time of broken loop steam generator equilibration to pressure setpoint
- Time of intact loop steam generator equilibration to pressure setpoint
- Time of full depressurization (3600 seconds)

In the mass and energy release data presented, no Zirc-water reaction heat was considered because the cladding temperature does not rise high enough for the rate of the Zirc-water reaction heat to proceed.

14.3.4.2 Long-Term LOCA Containment Response (COCO) Analysis

Accident Description

The Kewaunee Nuclear Power Plant (KNPP) containment system is designed so that for all loss-of-coolant accident (LOCA) break sizes, up to and including the double-ended severance of a reactor coolant pipe, the containment peak pressure remains below the design pressure. This section details the containment response subsequent to a hypothetical LOCA. The containment response analysis uses the long-term mass and energy release data from Section 14.3.4.1.

The containment response analysis demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a LOCA inside containment. The impact of LOCA mass and energy releases on the containment pressure is addressed to ensure that the containment pressure remains below its design pressure at the licensed core power conditions. In support of equipment design and licensing criteria (e.g., qualified operating life), with respect to post-accident environmental conditions, long-term containment pressure and temperature transients are generated to conservatively bound the potential post-LOCA containment conditions.

Input Parameters and Assumptions

An analysis of containment response to the rupture of the reactor coolant system (RCS) must start with knowledge of the initial conditions in the containment. The pressure, temperature, and humidity of the containment atmosphere prior to the postulated accident are specified in the analysis and are shown in Table 14.3.4-19.

Also, values for the initial temperature of the service water (SW) and refueling water storage tank (RWST) are assumed, along with containment spray (CS) pump flow rate and containment fan coil unit (CFCU) heat removal performance. All of these values are chosen conservatively, as shown in Table 14.3.4-19. Long-term sump recirculation is addressed via residual heat removal system (RHR) heat exchanger (HX) performance. The primary function of the RHR system is to remove heat from the core by way of low-head safety injection. Table 14.3.4-19 provides the RHR system parameters assumed in the analysis.

A series of analyses, using different break sizes and locations, was performed for the LOCA containment response. Section 14.3.4.1 documented the mass and energy releases for the minimum and maximum safeguards cases for a double-ended pump suction (DEPS) break and the releases from the blowdown of a double-end hot leg (DEHL) break.

For the maximum safeguards DEPS case, a failure of a containment spray pump was assumed as the single failure. This leaves one containment spray pump and four CFCUs available as active heat removal systems. Table 14.3.4-20 provides the performance data for one spray pump in operation. (Note: For the maximum safeguards case, a limiting assumption was made concerning the modeling of the recirculation system, i.e., heat exchangers. Minimum safeguards data were conservatively used to model the RHR HXs, i.e., one RHR HX was credited for residual heat removal. Emergency safeguards equipment data are given in Table 14.3.4-19.)

The minimum safeguards case was based upon a diesel train failure. This leaves one containment spray pump and two CFCUs available as active heat removal systems. Due to the duration of the DEHL transient (i.e., blowdown only), no containment safeguards equipment is modeled.

The calculations for all of the DEPS cases were performed for 1 million seconds (approximately 11.6 days). The DEHL cases were terminated soon after the end of the blowdown. The sequence of events for each of these cases is shown in Tables 14.3.4-21 through 14.3.4-23.

The following are the major assumptions made in the analysis:

1. The mass and energy released to the containment are described in Section 14.3.4.1 for LOCA.

2. Homogeneous mixing is assumed. The steam-air mixture and the water phases each have uniform properties. More specifically, thermal equilibrium between the air and the steam is assumed. However, this does not imply thermal equilibrium between the steam-air mixture and the water phase.

3. Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties.

4. For the blowdown portion of the LOCA analysis, the discharge flow separates into steam and water phases at the breakpoint. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment. For the post-blowdown portion of the LOCA analysis, steam and water releases are input separately.

5. The saturation temperature at the partial pressure of the steam is used for heat transfer to the heat sinks and the fan coolers.

Description of COCO Model

Calculation of containment pressure and temperature is accomplished by use of the digital computer code COCO (Reference 6). COCO is a mathematical model of a generalized containment. The proper selection of various options in the code allows the creation of a specific model for particular containment design. The values used in the specific model for different aspects of the containment are derived from plant-specific input data. The COCO code has been used and found acceptable to calculate containment pressure transients for many dry containment plants, most recently including Vogtle Units 1 and 2, Turkey Point Units 3 and 4, Salem Units 1 and 2, Diablo Canyon Units 1 and 2, Indian Point Unit 2, and Indian Point 3. Transient phenomena within the RCS affect containment conditions by means of convective mass and energy transport through the pipe break.

For analytical rigor and convenience, the containment air-steam-water mixture is separated into a water (pool) phase and a steam-air phase. Sufficient relationships to describe the transient are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate boundary conditions. As thermodynamic equations of state and conditions may vary during the transient, the equations have been derived for all possible cases of superheated or saturated steam and subcooled or saturated water. Switching between states is handled automatically by the code.

Passive Heat Removal

The significant heat removal source during the early portion of the transient is the containment structural heat sinks. Provision is made in the containment pressure response analysis for heat transfer through, and heat storage in, both interior and exterior walls. Every wall is divided into a large number of nodes. For each node, a conservation of energy equation expressed in finite-difference form accounts for heat conduction into and out of the node and temperature rise of the node. Table 14.3.4-24 is the summary of the containment structural heat sinks used in the analysis. The thermal properties of each heat sink material are shown in Table 14.3.4-25.

The heat transfer coefficient to the containment structure for the early part of the event is calculated based primarily on the work of Tagami (Reference 7). From this work, it was determined that the value of the heat transfer coefficient can be assumed to increase parabolically to a peak value. In COCO, the value then decreases exponentially to a stagnant heat transfer coefficient which is a function of steam-to-air-weight ratio.

The h for stagnant conditions is based upon Tagami's steady-state results.

Tagami presents a plot of the maximum value of the heat transfer coefficient, h, as function of "coolant energy transfer speed," defined as follows:

$$h = \frac{\textit{total coolant energy transferred into containment}}{(\textit{containment volume})(\textit{time interval to peak pressure})}$$

From this, the maximum heat transfer coefficient of steel is calculated:

$$h_{max} = 75\left(\frac{E}{t_p V}\right)^{0.60} \qquad \text{(Equation 1)}$$

where:

h_{max} = maximum value of h (Btu/hr ft^2 °F).
t_p = time from start of accident to end of blowdown for LOCA and steam line isolation for secondary breaks (sec).
V = containment net free volume (ft^3).
E = total coolant energy discharge from time zero to t_p(Btu).
75 = material coefficient for steel.

(Note: Paint is accounted for by the thermal conductivity of the material (paint) on the heat sink structure, not by an adjustment on the heat transfer coefficient.)

The basis for the equations is a Westinghouse curve fit to the Tagami data.

The parabolic increase of the heat transfer coefficient to the peak value is calculated by COCO according to the following equation:

$$h_s = h_{max}\left(\frac{t}{t_p}\right)^{0.5}, 0 \leq t \leq t_p \qquad \text{(Equation 2)}$$

where:

h_s = heat transfer coefficient between steel and air/steam mixture (Btu/hr ft^2 °F).
t = time from start of event (sec).

For concrete, the heat transfer coefficient is taken as 40 percent of the value calculated for steel during the blowdown phase.

The exponential decrease of the heat transfer coefficient to the stagnant heat transfer coefficient is given by:

$$h_s = h_{stag} + \left(h_{max} - h_{stag}\right)e^{-0.05(t-t_p)} \qquad t > t_p \qquad \text{(Equation 3)}$$

where:

h_{stag} = $2 + 50X, 0 < X < 1.4$.
h_{stag} = h for stagnant conditions (Btu/hr ft^2 °F).
X = steam-to-air weight ratio in containment.

Active Heat Removal

For a large break, the engineered safety features are quickly brought into operation. Because of the brief period of time required to depressurize the RCS or the main steam system, the containment safeguards are not a major influence on the blowdown peak pressure. However, they reduce the containment pressure after the blowdown and maintain a low, long-term pressure and a low, long-term temperature.

Refueling Water Storage Tank, Injection

During the injection phase of post-accident operation, the low-head safety injection pumps water from the RWST tank into the reactor vessel. Since this water enters the vessel at refueling water storage tank temperature, which is less than the temperature of the water in the vessel, it is modeled as absorbing heat from the core until the saturation temperature is reached. Safety injection and containment spray can be operated for a limited time, depending on the RWST capacity.

Residual Heat Removal, Sump Recirculation

After the supply of refueling water is exhausted, the recirculation system is operated to provide long-term cooling of the core. In this operation, water is drawn from the sump, cooled in an RHR exchanger, then pumped back into the reactor vessel to remove core residual heat and energy stored in the vessel metal. The heat is removed from the RHR HX by the component cooling water (CCW). The RHR HXs and CCW HXs are coupled in a closed-loop system, where the ultimate heat sink is the service water cooling to the CCW HX.

Containment Spray

Containment spray is an active removal mechanism that is used for rapid pressure reduction and for containment iodine removal. During the injection phase of operation, the CS pumps draw water from the RWST and spray it into the containment through nozzles mounted high above the operating deck. As the spray droplets fall, they absorb heat from the containment atmosphere. Since the water comes from the RWST, the entire heat capacity of the spray from the RWST temperature to the temperature of the containment atmosphere is available for energy absorption. During the recirculation phase, credit is taken for the sprays.

When a spray droplet enters the hot, saturated, steam-air containment environment, the vapor pressure of the water at its surface is much less than the partial pressure of the steam in the atmosphere. Hence, there will be diffusion of steam to the drop surface and condensation on the droplet. This mass flow will carry energy to the droplet. Simultaneously, the temperature difference between the atmosphere and the droplet will cause the droplet temperature and vapor pressure to rise. The vapor pressure of the droplet will eventually become equal to the partial pressure of the steam, and the condensation will cease. The temperature of the droplet will essentially equal the temperature of the steam-air mixture.

The equations describing the temperature rise of a falling droplet are as follows.

$$\frac{d}{dt}(Mu) = mh_g + q \qquad \text{(Equation 4 – Heat transfer)}$$

where,

M	=	droplet mass
u	=	internal energy
m	=	diffusion rate
h_g	=	steam enthalpy
q	=	heat flow rate
t	=	time

$$\frac{d}{dt}(M) = m \qquad \text{(Equation 5 – Mass transfer)}$$

where,

q	=	$h_cA * (T_s - T)$
m	=	$k_gA * (P_s - Pv)$
A	=	area
h_c	=	coefficient of heat transfer

k_g = coefficient of mass transfer
T = droplet temperature
T_s = steam temperature
P_s = steam partial pressure
P_v = droplet vapor pressure

The coefficients of heat transfer (h_c) and mass transfer (k_g) are calculated from the Nusselt number for heat transfer, Nu, and the Nusselt number for mass transfer, Nu′. Both Nu and Nu′ may be calculated from the equations of Ranz and Marshall (Reference 8).

$$Nu = 2 + 0.6(\mathrm{Re})^{1/2}(\mathrm{Pr})^{1/3} \qquad \text{(Equation 6)}$$

where,

Nu = Nusselt number for heat transfer
Pr = Prandtl number
Re = Reynolds number

$$Nu' = 2 + 0.6(\mathrm{Re})^{1/2}(Sc)^{1/3} \qquad \text{(Equation 7)}$$

where,

Nu′ = Nusselt number for mass transfer
Sc = Schmidt number

Thus, Equations 4 and 5 can be integrated numerically to find the internal energy and mass of the droplet as a function of time as it falls through the atmosphere. Analysis shows that the temperature of the (mass) mean droplet produced by the spray nozzles rises to a value within 99% of the bulk containment temperature in less than 2 seconds. Detailed calculations of the heatup of spray droplets in post-accident containment atmospheres by Parsly (Reference 9) show that droplets of all sizes encountered in the containment spray reach equilibrium in a fraction of their residence time in a typical pressurized water reactor containment. These results confirm the assumption that the containment spray will be 100% effective in removing heat from the atmosphere.

Containment Fan Coil Unit

The CFCUs are another means of heat removal. Each CFCU has a fan that draws in the containment atmosphere from the volume adjacent to the CFCU. Since the CFCUs do not use water from the RWST, the mode of operation remains the same both before and after the low-head safety injection change to the recirculation mode. The steam/air mixture is routed through the enclosed CFCU unit, past essential service water cooling coils. The fan then discharges the air through ducting containing a check damper. The discharged air is directed out emergency discharge dampers immediately adjacent to the CFCU. See Table 14.3.4-26 for CFCUs heat removal capability assumed for the containment response analyses.

Acceptance Criteria

The containment response for design-basis containment integrity is an American Nuclear Society (ANS) Condition IV event, an infrequent fault. The relevant requirements to satisfy Nuclear Regulatory Commission (NRC) acceptance criteria are as follows.

A. GDC 10 and GDC 49: To satisfy the requirements of GDC 10 and 49, the peak calculated containment pressure should be less than the containment design pressure of 46 psig

B. GDC 52: To satisfy the requirements of GDC 52, the calculated pressure at 24 hours should be less than 50% of the peak calculated value. (This is related to the criteria for doses at 24 hours.)

Analysis Results

The containment pressure, steam temperature and water (sump) temperature profiles from each of the LOCA cases are shown in Figures 14.3.4-1 and 14.3.4-6. Figures 14.3.4-1 and 14.3.4-2 show the DEHL break transient and Figures 14.3.4-3 through 14.3.4-8 show the minimum safeguards and maximum safeguards DEPS break cases. Table 14.3.4-27 summarizes the LOCA containment response results for the three cases studied.

Double-Ended Hot Leg Break

This analysis assumes a loss of offsite power coincident with a double-ended rupture of the RCS piping between the reactor vessel outlet nozzle and the steam generator inlet (i.e., a break in the RCS hot leg). The associated single-failure assumption is the failure of a diesel to start, resulting in one train of low-head safety injection and containment safeguards equipment being available. This combination results in a minimum set of safeguards being available. Further, loss of offsite power delays the actuation times of the safeguards equipment due to the required diesel startup time after receipt of the safety injection signal.

The postulated RCS break results in a rapid release of mass and energy to the containment with a resulting rapid rise in both the containment pressure and temperature. This rapid rise in containment pressure results in the generation of a containment high signal at 0.27 seconds and a containment high-high signal at 3.0 seconds. The containment pressure continues to rise rapidly in response to the release of mass and energy until the end of blowdown at 19.0 seconds, with the pressure reaching a value of 44.5 psig at 18.2 seconds. The end of blowdown marks a time when the initial inventory in the RCS has been exhausted and a process of filling the RCS downcomer in preparation for reflood has begun. Since the reflood for a hot leg break is very fast due to the low resistance to steam venting posed by the broken hot leg, Westinghouse terminates hot leg break mass and energy release transients at end of blowdown. The basis for this is discussed in

Section 14.3.4.1. Figures 14.3.4-1 and 14.3.4-2 show the pressure and steam temperature transients.

Double-Ended Pump Suction Break with Minimum Safeguards

This analysis assumes a loss of offsite power coincidence with a double-ended rupture of the RCS piping between the steam generator outlet and the RCS pump inlet (suction). The associated single-failure assumption is the failure of a diesel to start, resulting in one train of safety injection pumps and containment safeguards equipment being available. This combination results in a minimum set of safeguards being available. Further, loss of offsite power delays the actuation times of the safeguards equipment due to the required diesel startup time after receipt of the safety injection signal.

The postulated RCS break results in a rapid release of mass and energy to the containment with a resulting rapid rise in both the containment pressure and temperature. This rapid rise in containment pressure results in the generation of a containment high signal at 0.28 seconds and a containment high-high signal at 2.7 seconds. The containment pressure continues to rise rapidly in response to the release of mass and energy until nearly the end of the blowdown phase. The end of blowdown phase at 14 seconds marks a time when the initial inventory in the RCS has been exhausted and a slow process of filling the RCS downcomer in preparation for reflood has begun. The accumulators, which began injecting at approximately 7 seconds, continue to inject until approximately 38 seconds when the nitrogen cover gas begins to be released to the containment. This increases the containment pressure to its overall peak of 43.0 psig during the early portion of the reflood phase at 58.8 seconds. Since the mass and energy release during the reflood period is low and the containment fan coil units and containment sprays initiate, the containment pressure decreases throughout the remainder of the reflood phase. At this juncture, by design of the model described in Section 6.3, energy removal from the steam generator secondaries begins at a very high rate. This results in a rise in containment pressure while the energy is being removed from the faulted loop steam generator, bringing the faulted loop steam generator secondary pressure into equilibrium with the containment pressure. The result of this steam generator secondary energy release is a containment pressure of approximately 39.5 psig at 900 seconds. After this event, the mass and energy released is reduced due to so much energy removal from the steam generators having been accomplished and the containment pressure slowly falls out to the time when the injection sprays are terminated at 3802 seconds. The containment pressure is relatively stable until the recirculation begins at 3992 seconds when it again begins to decrease.

At this time, the low-head safety injection is realigned for recirculation resulting in an increase in the safety injection temperature due to delivery from the hot sump. The containment pressure continues to decrease due to lower decay heat, steam generator energy release, and continued CFCU cooling. This trend continues to the end of the transient at 1.0E+06 seconds. Figures 14.3.4-3, 14.3.4-4, and 14.3.4-5 show the pressure, steam temperature, and sump temperature transients. Table 14.3.4-3 shows the detailed sequence of events.

Double-Ended Pump Suction Break with Maximum Safeguards

The DEPS break with maximum safeguards has a transient history very similar to the minimum safeguards case. Table 14.3.4-4 provides the key sequence of events and Table 14.3.4-9 shows that a peak pressure of 42.5 psig at 58.6 seconds was calculated.

Conclusions

The LOCA containment response analyses have been performed as part of the SGR Program for Kewaunee. The analyses included long-term pressure and temperature profiles for each case. As illustrated above, all cases resulted in a peak containment pressure that was less than 46 psig. In addition, all long-term cases were well below 50 percent of the peak value within 24 hours. Based on the results, all applicable criteria for Kewaunee have been met.

14.3.5 OFF-SITE DOSE CONSEQUENCES

Introduction

The NRC has established guidelines in 10CFR 50.67 for radiation doses resulting from accidental releases of radioactivity from a reactor plant. This section shows the capability of the Kewaunee Nuclear Power Plant to stay within the dose criteria set forth in 10 CFR 50.67 following the design basis accident with conservative assumptions including assumed conditions of release consistent with those of NRC Regulatory Guide 1.183.

The Kewaunee Nuclear Power Plant Containment System is described in detail in Section 5. One feature of particular importance to the environmental consequences of a loss-of-coolant accident is the presence of two barriers, in series, to fission product leakage: the Reactor Containment Vessel and the Shield Building.

Reactor Containment Vessel leakage is collected within an annular volume between these barriers before release; the annulus is, therefore, effective as a means of holding leakage for decay and providing additional dilution prior to release. Release from the Shield Building to the environment is through absolute and charcoal filters provided in the Shield Building Ventilation (SBV) system. For reference in the evaluation of environmental consequences, a schematic diagram of this system is shown in Figure 14.3-35.

Shield Building Ventilation System fans establishes a negative pressure with respect to the atmosphere in the annulus within six minutes after the accident. The amount of filtered annulus air released to the environment is just sufficient to maintain the negative annulus pressure and compensate for in-leakage. The balance of the filtered annulus air is recirculated to the Shield Building to provide for further decay and filtration.

A limited amount of containment leakage could potentially bypass the Shield Building annulus through certain lines that terminate in the Auxiliary Building. This leakage will be collected and processed by the Auxiliary Building Special Ventilation System. An even smaller amount of containment leakage may bypass both the Shield Building and the Auxiliary Building and go directly to the environment. Both of these pathways have been evaluated.

Cause of Activity Release

The postulated cause of radioactivity release to the environment analyzed in this section is an extremely improbable double-ended rupture of a 29-inch inside diameter pipe in the reactor coolant loop. Following the assumptions of NRC Regulatory Guide 1.183, it is assumed that the Design Basis Accident will release to the Reactor Containment Vessel the following portions of the core activity:

- 100% of the noble gases (Xe, Kr) (5%in the gap and 95% in the fuel)
- 40%of the iodines (5% in the gap and 35% in the fuel)
- 30% of the alkali metals (Cs, Rb) (5% in the gap and 25% in the fuel)
- 5% of the tellurium metals (Te, Sb)
- 2% of the barium and strontium
- 0.25% of the noble metals (Ru, Rh, Mo, Tc)
- 0.05% of the cerium group (Ce, Pu, Np)
- 0.02% of the lanthanides (La, Zr, Nd, Nb, Pr, Y, Cm, Am)

The release of activity to containment occurs over a 1.8-hour interval. The gap activity is released in the first 30 minutes, and the fraction of the core activity that is released does so over the next 1.3 hours. A gap fraction of 5% is assumed for iodines, noble gases, and alkali metals. Gap activity of the other nuclides is not considered. With the exception of the iodines and noble gases, all activity released to containment is modeled as particulates. The iodine in containment is modeled as 4.85% elemental, 0.15% organic, and 95% particulate. A homogeneous mixture of this activity within the containment atmosphere is assumed to occur instantaneously. Because of the multiple redundancy in engineered safety features, such a release is considered incredible.

Sequence of Events Following a LOCA

As discussed previously, the Shield Building Ventilation System is designed to provide three functions during the course of the loss-of-coolant accident:

- Provide a negative pressure region to control and limit environmental leakage;
- Enhance mixing and dilution of any Containment Vessel leakage to the annulus;
- Provide holdup and long-term filtration of annulus air.

Immediately following the accident, the Shield Building pressure increases due to heat transferred from the containment shell. Operation of one of the Shield Building Ventilation System's two redundant fans establishes a negative pressure within ten minutes. During this period no credit is taken for the filtered exhaust of air by the Shield Building Ventilation System. Instead it is assumed that the Shield Building does not exist. From 0 to 10 minutes, 90% of the containment leakage is assumed to be released directly to the atmosphere without holdup or filtering. The remaining 10% goes to the Auxiliary Building Special Ventilation Zone where it is filtered before release to the atmosphere.

At 10 minutes into the accident, the Shield Building Ventilation System is assumed to be fully effective in controlling leakage into the Shield Building. It is assumed that from 10 minutes on, 89% of the containment leakage is processed by the Shield Building Ventilation System. Of the remaining 11%, 10% is assumed to go to the Auxiliary Building Special Ventilation Zone where it is subject to processing by the Auxiliary Building Special Ventilation System and 1% is assumed to be released directly to the atmosphere. These leakage assumptions are regarded as conservative upper limits as stated in the Technical Specifications for Kewaunee. A filter efficiency of 90% is applied to the removal of elemental and organic iodine and a filter efficiency of 99% is applied to the removal of particulates.

Figure 14.3-36 shows results of a shield building ventilation performance test. Also shown is a curve enveloping all data points with a considerable margin. This envelope is the basis for the conservative exhaust rates used in calculating offsite doses following a LOCA.

As described in Section 5, the shield building ventilation fans take a filtered suction from the annulus and their discharge is apportioned between atmospheric discharge and annulus recirculation flow. The atmospheric discharge (or SBV exhaust flow as shown in Figure 14.3-36) is dependent on the amount of annulus in leakage at the vacuum setpoint chosen for the Shield Building Ventilation System Control. The SBV exhaust flow will reach 2700 cfm at 20 minutes post-LOCA. Using the envelope in Figure 14.3-36, this exhaust flow will maintain a negative 1" wc (water column) in the annulus. It is recognized that exhaust flow rates must be higher early in the accident due to annulus heating. This is shown in Figure 14.3-37A. To simplify the analysis a constant exhaust rate of 6000±10% cfm has been used from 10 to 30 minutes and 3100 cfm from 30 minutes to 30 days. This simplification can be made due to the size of the envelope in Figures 14.3-36 and 14.3-37A.

The Containment Vessel Internal Spray System is described in Section 6.4. The primary purpose of the spray system is to spray cool water into the containment atmosphere in the event of a LOCA and thereby ensuring that containment pressure does not exceed its design value. However, the spray system also has the property of removing iodine and particulates from the containment vessel atmosphere. The iodine removal coefficients due to containment spray are given in Table 14.3-8. During spray operation, no credit is taken for sedimentation removal of particulates, although it would take place. Recirculation sprays are not credited. Credit is taken for sedimentation removal of particulates after spray termination. The analysis credits a sedimentation coefficient of $0.1h^{-1}$. Table 14.3-8 also provides a summary of the other parameters used in the analysis, which was documented in Reference 1.

Method of Analysis

The evaluation of the environmental consequences of a loss-of coolant accident consists of determining the radiation dose resulting from inhalation of radioiodine discharged from the

Shield Building and of determining the dose due to direct gamma radiation from the radioactive cloud created by the discharge of Containment Vessel leakage from the Shield Building.

The evaluation of the environmental consequences of a loss-of-coolant is based on the assumptions of NRC Regulatory Guides 1.183. The analysis model describing the activity release is shown in Figure 14.3-35. This figure illustrates the basic assumptions of the analysis. A portion of the Containment Vessel activity inventory is assumed to leak into the Shield Building and form the Shield Building activity inventory. Activity leaves the Shield Building by passing through the charcoal filters. Of the activity passing through the filters, a portion is released via the shield building vent and the rest is recirculated back into the Shield Building. The Shield Building activity is also a function of the assumed annular participation fraction. This fraction is a measure of the mixing efficiency in the annulus.

When ECCS recirculation is established following the LOCA, leakage is assumed to occur from ECCS equipment outside containment. There are two pathways considered for the ECCS recirculation leakage. One is the leakage directly into the auxiliary building and the other is back-leakage into the RWST. Although recirculation is not initiated until the RWST has drained to the pre-determined setpoint level the analysis conservatively considers leakage from the start of the event. For the ECCS leakage analysis, all iodine activity released from the fuel is assumed to be in the sump solution until removed by radioactive decay or leakage from the ECCS. The iodine activity that becomes airborne after being released by the leakage is modeled as 97% elemental and 3% organic.

The analysis models leakage to the auxiliary building of 12 gallon/hr. The analysis models a conservative airborne fraction of 10% when the sump temperature is above 212°F. Once the sump solution temperature drops below 212°F, the airborne fraction is reduced to 1%. The reduction in airborne fraction is conservatively delayed until 3 hours from the start of the event.

RHR back-leakage to the RWST is assumed at a rate of 3 gpm for the first 24 hours, and 1.5 gpm for the remainder of the event. It is assumed that 1% of the iodine contained in the leak flow becomes airborne. The 1% value is applied even when the sump is above 212°F since any incoming water would be cooled by the water remaining in the RWST. The RWST vents to the auxiliary building.

It is assumed that half the iodine activity that becomes airborne in the auxiliary building from the two leak sources is removed by plateout on surfaces. Releases from the auxiliary building are subject to filtration by the auxiliary building special ventilation system.

The analysis is documented in Reference 1. Dose results are listed in Table 14.3-9. Reference 1 also demonstrates that the 30 day dose to control room operators is within the limit specified in 10 CFR 50.67.

Conservatism Between Analysis and Physical Situation

Many conservative assumptions have been made in the application of the NRC Regulatory Guide 1.183 to the Kewaunee Plant. Elimination of this conservatism would be expected to reduce the calculated dose by orders of magnitude. In order to place the above analysis in perspective, major assumptions applied in the analysis which affect the calculated dose are reviewed below:

1) In accordance with NRC Regulatory Guide 1.183, a release of activity from melted fuel to the containment atmosphere was assumed. As shown in Section 14.3.2, the Safety Injection System will prevent fuel rod clad melting and will limit the zirconium-water reaction to an insignificant amount. However, as a result of the cladding temperature increase and the rapid system depressurization following the accident, cladding failures may occur in the hotter regions of the core. These failures would release only the inventory of volatile fission products in the gap between the fuel pellet and the cladding.

2) No reduction of activity has been assumed by plateout in the Shield Building.

3) Recirculation filtration has been assumed to take place with only one of two redundant systems operable. Since the combined flow capability of the two recirculating fans will be double that used in the analysis, a significant reduction of iodine and particulate activity in the Shield Building would result.

14.3.6 DELETED

14.3.7 DELETED

14.3.8 CHARCOAL FILTER IGNITION HAZARD DUE TO IODINE ABSORPTION

The radioactive iodine, which collects on the charcoal filters generates a significant amount of decay heat. A detailed analysis was made of the potential for spontaneous ignition of the charcoal during post-LOCA operation of the Shield Building Ventilation (SBV) system. To maximize the charcoal filter temperature, it was assumed that forced air-cooling is lost at the time of maximum heat load.

Using the assumptions of NRC Safety Guide 4, i.e., 50% halogen release from the fuel and 50% plateout in the Reactor Containment Vessel, the iodine released and the heat generated from that iodine are estimated to be:

Isotope	Curies (10^7)	Decay Heat (kW)
I-131	1.21	41.55
I-132	1.718	262.7
I-133	2.26	143.7
I-134	2.575	450.3
I-135	2.0	292.7

The maximum amount of heat that can be generated on the filters is limited by the rate at which the iodine leaks out of the Reactor Containment Vessel onto the filters, and by the decay of the isotopes that are collected on the filter.

In the analysis performed, the following conservative assumptions are made:

1) It is assumed that no holdup takes place in the Shield Building, i.e., all of the activity released via Containment Vessel leakage goes directly on the filters.

2) No credit is taken for plateout in the Shield Building.

3) All of the activity is assumed to collect on one train of the SBV filters with 100% efficiency.

With those assumptions, the maximum rate of heat generated on the charcoal filters was predicted to occur at one day following the accident. At this time of maximum heat load, the forced air cooling through the filter assembly is assumed to be lost. Assuming the charcoal filter is at 190°F (based on calculated post-accident shield building temperature) when forced cooling is lost, results in a charcoal filter centerline temperature of 258°F, which is significantly lower than the 626°F ignition temperature of the charcoal used. This temperature is also below the 356°F at which iodine desorption is expected to begin.

This analysis is conservative by at least an order of magnitude for the following reasons:

1) The analysis was based upon a NRC Safety Guide 4 release.

2) The maximum allowable containment vessel leak rate is assumed to occur for the first day following accident initiation. Peak containment pressure occurs only a few minutes following accident initiation and decays quickly. This would result in a leak rate much less than maximum allowable.

3) All plateout of iodine in the Shield Building was neglected.

4) All activity was assumed to collect on one of two filters.

5) All activity leakage from the Containment Vessel was assumed to collect, at 100% filter efficiency, on the filter without holdup or decay in the Shield Building.

6) No heat dissipation from the filter housing to the surrounding room environment is assumed.

A spray system is provided which is activated automatically upon occurrence of high temperature adjacent to the charcoal. The analysis has shown that actuation of this system is not expected to occur.

14.3.9 GENERATION AND DISPOSITION OF HYDROGEN

General

The design basis loss-of-coolant accident and its off-site consequences are discussed earlier in this section. It is recognized that a loss-of-coolant accident could be followed by the possible generation of hydrogen from radiolysis of water, from chemical corrosion of materials by spray liquids, and from possible metal-water reactions accompanying the accident.

The equilibrium concentrations that could theoretically result have been calculated to exceed the lower flammability limit of 4.1 volume percent hydrogen; therefore, it is necessary to provide means of limiting the accumulation of hydrogen to an acceptable lower concentration.

The simplest means of control is to purge, venting the mixture of air and hydrogen to the environment, at a rate sufficient to maintain a hydrogen concentration that is below the lower flammability limit.

The capability of venting is an essential part of any system of hydrogen control because the eventual containment cleanup must be by controlled dispersal of containment gases to the environment; hydrogen control and eventual containment purge by venting are inseparable considerations of the same loss-of-coolant accident because any primary means of control cannot be terminated until conditions permit venting to proceed at a rate sufficient to supplant it and prevent further rise in hydrogen concentration. Complete analysis must be based on a reference condition of acceptable venting at which venting can later proceed at a rate greater than that necessary to control hydrogen. This condition is conveniently defined as the occurrence of 1 MPC at the site boundary (i.e., when the summation of the fractions of the maximum permissible concentrations given in 10 CFR 20 equals unity).

Summary of Reanalysis Based on Safety Guide 7

Initial studies were based on conservative estimates of hydrogen sources provided by the reactor supplier, and they included the assumptions of Safety Guide 4. These studies indicated that 3.5 v/o concentrations would be reached in 56 days, that venting could be deferred until 86 days after the accident without the lower flammability limit being reached, and that initiation of venting through a charcoal filter at a rate sufficient to arrest hydrogen accumulation would then result in instantaneous site boundary concentrations no greater than 1 MPC. Thus, direct venting through charcoal was indicated to be sufficient means of hydrogen control.

The present reanalysis uses a Containment Vessel leak rate of 1% per day. A conservative filter efficiency of 90% for the Shield Building Ventilation System is used in the reanalysis, but doses are also given for a filter efficiency of 95%.

Also, additional considerations are now incorporated in the analysis, based on information published subsequent to initial submittal of this USAR:

1) Safety Guide 7 now prescribes even more conservative assumptions regarding hydrogen sources.

2) Test data indicate that, at least for higher initial post-accident containment temperatures, substantial amounts of hydrogen could be released from painted surfaces during the first day following the accident.

Both the time at which hydrogen control must be initiated and the doses associated with venting are extremely sensitive to the assumptions regarding hydrogen sources. The above two effects, for example, would alone advance the calculated time of occurrence of 3.5% hydrogen concentration -- from 56 days to 18 days and from 56 days to 41 days, respectively. Together they result in occurrence of 3.5% concentration on the eleventh day.

The collective assumptions prescribed in Safety Guide 4 and Safety Guide 7 are regarded as being unnecessarily conservative. Also, for conservatism it is found necessary to overestimate the potential contribution from protective coatings, which is the remaining source, because the test data are for simulated post-accident conditions of temperature much more severe than those predicted. The hydrogen generation and venting problem will consequently be far less severe than determined from these assumptions.

A method of venting is proposed which is indicated to attain reasonable off-site doses, even under these conditions of early venting requirements. In addition, means are provided to defer venting, if necessary, by compressing the containment atmosphere, thus diluting the hydrogen within the containment, and thereby reducing the potential dose from venting.

Methods of Control

Two modes of operation are being provided, any of which employed alone would provide adequate means of hydrogen control.

1) Controlled vent flow and processing of this flow with recirculation filtering by the SBVS before its release to the environment.

2) Deferment of venting, if necessary, by adding air to the containment to compress the atmosphere, and thus, dilute its hydrogen concentration.

In addition to these two methods of hydrogen control, the capability to utilize an external hydrogen recombiner, has been provided.

Venting to the Shield Building Annulus

The Shield Building Ventilation System affords the benefit of recycle through charcoal filters. When venting must first be initiated, at least one-of-two redundant trains of equipment will already be in continuous operation, maintaining vacuum and collecting and processing containment vessel leakage before its discharge to the environment. Any vent flow necessary to maintain acceptable hydrogen concentrations within the containment will be directed into the Shield Building annulus at a controlled rate, to be processed along with the containment leakage which would represent part of the required vent flow.

The effect of recycle through the filters of the Shield Building Ventilation System is to reduce the iodine effluent concentration by an additional factor that is essentially equal to the ratio of recirculation flow to discharge flow to the environment. This is the same factor that has been applied to the standby ventilation system for a boiling water reactor plant (Reference 1).

Analysis demonstrates that venting at greater than 1 MPC would be unnecessary on the basis of reasonably conservative assumptions, which include the conservative allowance for protective coatings. The need to vent at higher activity concentrations might be required only for the extremely conservative basis of Safety Guide 7. If post-accident hydrogen generation were in accordance with this most conservative estimate, the resulting doses from the processed vent flow are indicated to be a small fraction of those of the accident analysis. The time delay before initiation of venting and the conservative allowances made in the initial phases of the accident analysis for direct filtered and unfiltered release of containment leakage are not applicable during the equilibrium recycle operation which will be established before venting is necessary.

Dilution by Containment Pressure Raise

Dilution of hydrogen concentration by modest increases in containment pressure is one of several simpler methods of hydrogen control that were first proposed and investigated in 1970 for Wisconsin Public Service Corporation and other participants of the same study program.

Partial pressurizing of the containment can defer the occurrence of limiting hydrogen concentrations, and consequently defer the need to vent until appreciable decay of containment activity has occurred. The objective is to defer venting until venting at an acceptable dose level alone can arrest the further accumulation of hydrogen and permit termination of the method used to defer venting. Such deferral of venting is the identical objective of other methods, such as the use of recombiners.

Controlled venting with the accompanying replenishment of vented containment air requires that a pressure differential exist at least intermittently across the containment shell. The venting system can readily be designed for an operating differential in either direction, but design for positive internal pressure during venting provides the option of deferred venting.

Analysis indicates that a rather small increase in pressure can significantly reduce the venting doses associated with the most conservative estimates of hydrogen sources, and that the method provides a practical means of utilizing the benefits of recirculation filtration provided by the Shield Building Ventilation System.

Hydrogen Recombination

In addition to the two methods of hydrogen control described above, the capability to use an external hydrogen recombiner for recombination of hydrogen and oxygen into water vapor has been provided. Permanent piping, valving, and power source connections are provided at two separate locations for the connection and operation of an external hydrogen recombiner within the Auxiliary Building. The two locations allow recombiner placement at approximately opposite sides of Containment; in the unlikely situation that one of the two areas will be unavailable or be required for continual personnel occupancy, the remaining location will remain available for recombiner placement and operation.

Analysis of Materials of Construction and Protective Coatings

Analysis has been made of the materials of construction and the protective coatings used within the containment, particularly as they affect the potential of hydrogen generation by reaction with spray solution.

Description of Materials

The original specified coatings for structural steel items were 3 mils of Carbo-zinc 11 primer plus a 4-mil finish coat of Phenoline 305. The same protective coatings were specified for the inner surface of the steel containment vessel, plus an additional 4-mil finish coat between elevations 606 feet and 660 feet. These coatings will be maintained by application of an appropriate Service Level I protective coating or coating system, depending on the extent of repair. Appropriate Service Level I coatings for use in containment are determined by KNPP engineering specifications and procedures. Periodic inspections are performed to assess the condition of protective coatings on the vessel and structural steel.

All concrete walls, floors, ceilings, and other surfaces within the containment were originally protectively coated with sealer, surfacer, and/or finish coats of Carboline or Phenoline. Neither of these coatings included Carbo-zinc. The coatings on concrete walls, floors, ceilings, and other surfaces are also maintained by application of appropriate Service Level I protective coatings. Periodic inspections are performed to assess the condition of protective coatings on concrete surfaces.

The use of unqualified coatings in containment is minimized. Unqualified coatings were used in containment on structural and architectural steel, piping, various equipment and components, and other miscellaneous items. Analysis has shown that the current amount of unqualified paint will not affect the operability of the emergency core cooling system and the internal containment spray system following a loss of coolant accident. Components with factory coatings, which are unqualified will not be stripped of the coating and re-coated. This is in the interest of equipment reliability and nuclear safety.

Galvanized steel is used for ventilation ducts, gratings, stair treads, etc., and some aluminum is used in components and protective coatings associated with the reactor equipment and the reactor building crane. The use of these materials has been minimized in design to the extent practical.

The surface areas and amount of the materials are summarized in Table 14.3-10.

Other materials in contact with the spray solution, such as stainless steel and copper alloys, are not significant with regard to corrosive generation of hydrogen.

The effects of corrosion on component integrity are of possible concern only with regard to potential chloride stress corrosion of stainless steel by the boric acid spray solution. Sufficient caustic will be added with containment spray so that both the initial spray and the recirculated sump solution will be at pH of 7 or higher. Means are provided to monitor the chloride content of the recirculated sump water. Corrosion effects are not otherwise of concern with regard to component integrity. For example, if the zinc-bearing coating of galvanized ductwork were to be completely consumed by reaction with the spray, there would be negligible further corrosion of the exposed steel.

Zinc-Bearing Surfaces

A zinc-bearing primer is used as an undercoat on original structural steel and on the inner surface of the containment vessel.

The results of ORNL experiments indicate that substantial amounts of hydrogen can evolve from such undercoats during the initial conditions of a loss-of-coolant accident. This effect appears to be independent of the type of spray solution and of the amount and type of coating over the primer. The outer coating is reported to typically appear unaffected after exposure conditions even though measurable releases of hydrogen from the under-coatings were produced.

The most relevant experiments (Reference 2) involved exposure of vendors' test coupons to spray solution under temperature conditions intended to simulate those of a loss-of-coolant accident in a PWR: 5 minutes at 300°F, 105 minutes at 284°F, and the remainder of a day at 225°F. For a boric acid spray solution of 3000 ppm boron without additives, the most applicable paint sample (3 mils of Carbo-zinc plus a 2 mil overcoat) yielded 2.3 cc of hydrogen per cm^2 of surface or 0.075 scf/ft^2. With 0.15N NaOH added to the solution, the yield was 2.0 cc/cm^2 or 0.066 scf/ft^2. The hydrogen release from these tests was typically about 60% of that released in previous tests in which the exposure temperature was maintained constant at 266°F for 24 hours.

The test conditions for both sets of tests were much more severe than those predicted for the design basis accident, and substantially less hydrogen generation would therefore be expected. It is conservatively assumed in the analysis that the first-day contribution from the painted surface is given by half the product of the total area given in Table 14.3-10 and the release per unit area given by the higher temperature experiments intended to simulate accident conditions.

The factor of two is perhaps justified alone on the basis of the fraction of total zinc-bearing surface that will be directly exposed to the spray solution, but even greater reduction factors should result from the reduced temperatures relative to the experiment. The post-accident air and steam temperatures are predicted to be only 265°F maximum during the first 5 minutes; a decrease from 238°F initial temperature to 140°F during the next 105 minutes; and 140°F or less thereafter.

An approximate indication of the effect of temperature is given by the relation of Arrhenius for the case of constant activation energy: $R(T_1)/R(T_o) = \exp[\alpha(T_1-T_o)/T_1T_o]$. Many reactions double or triple in rate for a 10°C rise in temperature in accordance with this relation (Reference 3). Its direct application to the typical relative yields of the ORNL tests (those for the simulated accident conditions vs those for constant temperature exposure at 266°F for 24 hours) implies a doubling in rate for every 4°C rise.

To obtain indication of the reduced reaction that might be expected at the lower temperatures predicted for the design basis accident, the same relation is applied identically to the time-temperature sequence of the accident-simulation experiment and to the predicted first-day post-accident temperature curve. With reaction rates near 300°F assumed to double for temperature increments ranging from 4°C to 30°C, the resulting reduction factors in the first-day reaction are indicated to vary from 109 to 4.2, respectively, relative to the accident simulation tests.

These indications suggest that a reduction factor of 10 to 100 would be appropriate. However, in the absence of lower temperature data or direct indication of temperature sensitivity of the reaction rates, an overall reduction factor of only two is conservatively assumed.

Aluminum Surfaces

Corrosion of aluminum surfaces would be negligible with an acidic borated spray being a factor of a hundred or more less than with a basic spray solution (Reference 4).

For the case of buffered spray solution, reaction rates of aluminum are available as a function of temperature (Reference 5). Application of this information to the temperature transient predicted for the design basis accident indicates 1 mil reaction on the first day, plus a continuing rate thereafter that is less than the 200 mils/year prescribed by Safety Guide 7.

Net Effect of Spray Additive

The total hydrogen production from protective coatings (see Table 14.3-11) has been calculated with and without spray additive, and with first-day production from galvanized coating treated identically as the paint, because of the absence of relevant information for the case of no additive and because the zinc content of most of the galvanized surface is similar to that of the paint. Addition of spray additive justifies use of a lower conservative estimate for painted surfaces and neglect of the first-day contribution from the galvanized surface.

An added allowance is made for aluminum reaction with additive, based on full consumption of the 110 pounds of aluminum paint on the first day and 200 mils/year consumption of the remaining aluminum, assuming ¼-inch effective thickness and 20 scf hydrogen generation per pound consumed.

From Table 14.3-11, it can be noted that the calculated total coating contribution at 10 days would not increase with the use of additive. Also, the coating contribution is only a minor part of total continuing production; therefore, the adjustment for additive is not significant. Effectively, the extremely conservative allowance for zinc-bearing surfaces that is necessitated by the absence of lower temperature data, and the appropriate adjustments in this conservative estimate, obscure the net increase in hydrogen that should result from the directly calculable effect of spray additive on aluminum.

Analysis of Methods of Hydrogen Control

Sources and Assumptions

Studies have been based primarily on the results of conservative hydrogen generation calculations provided by the reactor supplier. The major assumptions for this "reasonably conservative case" are summarized in Table 14.3-12, and compared with those for a "most conservative case" which includes the assumptions of Safety Guide 7. Both cases described in this table include the conservative allowance for first-day reaction of the zinc-bearing surfaces described in the previous section.

The significant differences introduced by Safety Guide 7 are the increase in assumed zirconium reaction, the higher value of G, and the greater core gamma absorption in the coolant. These assumptions add a half percent more hydrogen concentration for zirconium and increase the radiolytic sources by factors of 1.67 in the core and 1.6 in the coolant. The overall effect is to alter entirely the urgency and magnitude of the hydrogen problem, as shown by comparison of the first several lines of Table 14.3-13.

The source contributions and venting requirements for the most conservative case that based on Safety Guide 7 are shown in Figure 14.3-43.

Analysis of Venting Through Shield Building Annulus

The first case analyzed is that of controlled venting through the Shield Building Ventilation System without pressurizing, and neglecting any effects of the positive pressure differential that would be used to accomplish such venting.

It is assumed that venting is initiated upon measured occurrence of 3.5 v/o concentration (as shown in Figure 14.3-43), and continued at a diminishing rate which maintains that concentration until the effects of decay and purge depletion result in instantaneous venting concentrations of 1 MPC at the site boundary. The purge and vent rate is then increased with further decay and purge depletion while maintaining 1 MPC off-site, causing containment activity and hydrogen content to decrease monotonically until purging is complete.

Calculations with regard to venting are based on the initial activity inventories described in Appendix D, Table D.1-1 (consistent with Safety Guide 4) and on the dispersion factors and breathing rate appropriate to the two-day to thirty-day period of the meteorological studies:

χ/Q	=	3.882E-6 sec/m³ at the site boundary
	=	4.473E-7 sec/m³ at 4800 meters
ß	=	2.32E-4 m³/sec

Venting is assumed to be through the Shield Building annulus with equilibrium recirculation flow of 4000 cfm through filters which remove 90% of the iodine and all solid fission products, and with constant discharge flow of 200 cfm. These conservatively chosen flow conditions are consistent with the equilibrium phase of the calculations of shield building activity discharge during the design basis accident.

The resulting venting doses are presented in Table 14.3-13.

The reasonably conservative case described in the Table 14.3-13 results in very low off-site venting doses because it represents a rather trivial case of venting. For consistency in comparison of the two cases, it has been assumed that venting is initiated upon occurrence of 3.5% hydrogen. However, the initial activity levels from venting at this time would be about 7 MPC at the site boundary and, by simply deferring venting and allowing the concentration to rise further (to 4.07%), venting could later be initiated at a rate sufficient to arrest the concentration at this higher value without exceeding 1 MPC at the site boundary.

Greater venting doses are indicated in Table 14.3-13 for the most conservative case, but these represent a minor portion of the leakage doses associated with the maximum design accident. They might be regarded as an added penalty from venting, except that occurrence of the postulated accident leakage would have deferred and greatly reduced the consequence of venting (or would have obviated the need to vent, for example, in the case of normal initial leak rate of 1% per day).

Thus, the need to vent at off-site concentrations greater than tolerance in order to maintain safe hydrogen concentrations can be predicted only on the basis of the most conservative assumptions -- those of Safety Guide 7.

For purposes of evaluation of the venting doses, it may be noted that the recycle advantage factor which is incorporated in all the thyroid doses in Table 14.3-13, and which affects the occurrence of 1 MPC vent capability, reduces effectively to nP/L_2 = .90 x 4000 cfm/200 cfm = 18, where P is the recycle flow, L_2 is the discharge flow, and n is the removal efficiency of the charcoal. This advantage factor is independent of the partition factor or effective volume fraction assumed for mixing in the annulus. The reduction factor would instead be 45 at the expected conditions of 5000-cfm recirculation and 100-cfm discharge, and the iodine doses would then be 40% of those indicated in Table 14.3-13. The thyroid doses would be similarly affected by the removal of iodine by containment spray liquid, which effect, with additive, should further reduce the indicated thyroid doses by a large factor.

Analysis of Effects of Pressure Increase

The lower limit of flammability for hydrogen in air is reported to be 4.1 volume percent at atmospheric pressure, and this limit is reported to increase slightly with pressure, rather than to decrease (Reference 6).

Thus, compression of a hydrogen-air mixture by addition of more air to a fixed volume decreases the volume fraction of hydrogen and permits more hydrogen to be accumulated for a given limiting volume fraction. The calculation assumes that once a limiting concentration C_o is reached (3.5%), as hydrogen production continues, the containment pressure P is raised by injecting air at a rate sufficient to maintain C_o, where a concentration C(t) would otherwise result if dilution by pressure increase did not occur:

$$P(t) = 14.7\frac{C(t)}{C_o}$$

Doubling of the absolute pressure, for example, would permit hydrogen to be maintained at 3.5% until 7% would otherwise have accumulated.

A second effect is that, when purging and venting are initiated, the fractional vent-flow necessary to maintain a given concentration at any time is less at pressure. The mass flow required is independent of pressure, but the flow expressed in fraction of containment volume is less by the ratio of absolute pressures, as is the fractional release rate of contained activity.

As well as deferring the need to vent, overpressure causes leakage to occur from the containment to the annulus. The assumed leakage is based on 1% per day at the design pressure of 42 psi:

$$L(t) = .01/day\sqrt{\frac{P(t)-14.7}{42}}$$

Out-leakage and vented gases are necessarily processed identically by the Shield Building Ventilation System. They are consequently equivalent from the standpoint of radiation dose, as well as with regard to their effectiveness in hydrogen control when they are replenished by purge flow. Out-leakage, thus, simply represents a portion of the venting rate that is not deferred when venting is otherwise deferred by pressure increase.

The effects of pressure increase on venting requirements and containment leakage are described by Figures 14.3-44 and 14.3-45, and the resulting doses are presented in Table 14.3-14.

The upper curves of Figure 14.3-44 are for a limited pressure rise of 6.1 psig, sufficient to maintain 3.5% hydrogen up to 30 days. Venting is then initiated at the same rate that compressed air is being added, with pressure consequently remaining constant. This constant pressure purge continues at a rate no more than necessary to maintain constant concentration, until decay and diminishing vent rate cause the concentrations at the site boundary to reduce to 1 MPC. Purging then proceeds at the rate which theoretically maintains 1 MPC and which rapidly increases as a consequence of further decay and purge depletion.

Pressure relief could be initiated between 30 and 62 days in this case, by allowing the vent rate to exceed the purging or replenishment rate necessary to maintain constant hydrogen, but this would unnecessarily increase the venting dose. Similarly, beyond 62 days, pressure relief cannot fully proceed by unreplenished vent relief at the vent rate permitted by off-site concentration limits. Continued existence of hydrogen sources requires instead that some part of the permissible vent flow be replenished as a purging flow to prevent the limiting concentration being exceeded during this period of final cleanup. The solid-line blowdown curves in Figure 14.3-44 are the lines of earliest pressure relief, which maintain both the limiting hydrogen concentration and the limiting off-site radiation concentration during a controlled blowdown plus purge. The time for complete cleanup is independent of pressure or purging considerations; it is determined only by the venting rate and venting depletion of the contained activity, which effect is also idealized in the figures for the case of earliest cleanup.

The resulting doses are shown in Table 14.3-14 and compared with the previous case of no deferment of venting. Deferment to 30 days is seen to gain a reduction of 10 in venting dose, but the containment leakage dose resulting from the overpressure reduces this factor to 5 or 6.

The out-leakage rates are shown in Figure 14.3-45 and compared with the vent rates. The dose from leakage is additive only up to the time when venting is initiated because the purge rates are the total replenished out-flow required to maintain concentration, either by leakage or venting.

Out-leakage should be less than that estimated on the basis of the design leakage, and it will be assured to be less by surveillance testing of the containment. The effects of out leakage are conservatively considered in the calculations but the compensating depletion, which it would cause is neglected because the actual leakage that would occur is uncertain.

The effect of pressure on purge rate may be noted from Figure 14.3-45, where the top solid curve is the fractional purge rate necessary to maintain concentration C_o at atmospheric pressure:

$$L(t)=\frac{Q(t)N}{C_o}$$

with Q(t) the uncompressed volumetric production rate of hydrogen at time t in the containment air volume V that is used to define concentration.

The dashed line describes the initial purge rate at time t that supplants a pressure increase that was maintaining C_o:

$$L(t)=\frac{14.7/P(t)}{C_o}\times Q(t)N=\frac{Q(t)N}{C(t)}$$

where C(t) is the concentration that would occur at time t without dilution by pressure increase.

The solid curve for constant pressure purge initiated at thirty day is:

$$L(t)=\frac{Q(t)N}{C(30)}=\frac{14.7}{P(30)}\times\frac{Q(t)N}{C_o}$$

Thus, required purge rate is reduced with pressure increase directly in the ratio of absolute pressures. Both solid curves in the figure correspond to the same mass flow of air and hydrogen, but they differ with regard to fractional release rate of the contained volume, and hence, with regard to fractional release rate of the contained activity.

The lower curves in Figure 14.3-44 correspond to sustained pressure rise until a 1 MPC venting capability is attained, sufficient to control hydrogen. As shown in Table 14.3-14, the dose is all from out-leakage and little is gained relative to initiating venting at thirty days. However, the containment is tested initially at 46 psig, and thus, its design does not preclude continuing the pressure rise as far as required.

An additional case has been considered wherein recycle credit for iodine is neglected, as would be the case if all leakage and vent flow were somehow to be discharged directly to a 90% filter without recirculation.

The diagonal line in Figure 14.3-45 corresponding to a 1 MPC vent of leak rate would effectively be displaced downward in this case by loss of the recirculation advantage factor of 18.1 for iodine, and the 1 MPC venting intersection would be accordingly deferred in time. It can also be seen from the figure that further pressure rise would cause the containment leakage to exceed the vent rate necessary to control concentration, the intersection occurring at 67 days and 12.8 psi for a 1% leak rate. At this point leakage alone would control hydrogen, the air supply could be reduced to match the leakage and prevent further rise.

A 1 MPC venting capability is deferred in this case to about 90 days, and the total leakage dose is 18.1 rad to the thyroid and 0.107 R whole body radiation. The doses would be less and the maximum useful pressure would be slightly greater for lower containment leak rates.

Thus, even without the recycle advantage at the Shield Building Ventilation System, acceptable doses are indicated for the method of pressure control alone.

Provisions for Mixing, Sampling and Venting of Containment Gases

The provisions for mixing, sampling and venting of containment gases are shown schematically in Figure 14.3-46.

Mixing

Two containment dome vent fans are provided to circulate and mix gases within the containment during the period following the postulated loss-of-coolant accident when combustible gases could conceivably accumulate. Each fan draws a combined 8000-cfm through two-of-four inlet ducts located in the dome area of the Containment Vessel. The discharge from each fan is conveyed downward through separate ductwork and returned to the containment volume near the operating floor.

These systems are completely independent and redundant to each other, and they satisfy the requirements of Engineered Safety Features. The fans are started manually from the control room, and surveillance testing of the capability of these systems to start and operate as intended is performed during refueling outages.

Venting to Shield Building Annulus

Two vent valve systems are provided to accomplish venting of pressurized containment gases to the Shield Building annulus. The two systems will each vent containment gases from the ductwork associated with one of the containment dome vent fans and transfer them by means of positive pressure differential through separate penetrations of the Containment Vessel. Each penetration has remote-manually operated isolation valves that are normally closed and that can be separately opened to permit venting, or sampling, through either penetration. Each penetration exhausts gases through a remotely controlled throttle valve directly into the Shield Building annulus where they will be processed and discharged by the Shield Building Ventilation System.

Each penetration has a remote-indicating flow meter located in the annulus and upstream of the throttle valve so as to indicate fractional containment volume vent rate, independent of containment pressure.

The vent relief systems are located entirely within the annulus to preclude concern for leakage. The systems and their power supplies meet the requirements for engineered safeguards.

The vent system flow requirements are:

- that each throttle valve can accomplish the maximum vent rate necessary to control hydrogen (~25 scfm) at a nominal driving pressure (~2 psi, or greater if necessary); and

- that the combined capacity should not present a significant limitation with regard to time requirements for the completion of containment purge when transfer is eventually made to direct filtered discharge.

Analysis indicates that, for a design leak rate of 1% per day, the resulting out-leakage limits the useful pressure increase to about 13 psi, and this, or an even lower pressure, can be set as an operational limit. The vent system was tested in conjunction with the Shield Building Ventilation System to establish acceptable limits, and limits were set by operating procedures and, if necessary, can be set by means of fixed orifices downstream of the throttle valves.

Compression and replenishment of containment gases is through either of two penetrations that span the annulus to admit fresh air through the instrument air system. These penetrations will each initially be equipped with normally closed, remote manually opened isolation valves, throttle valve, and connections for use of oil-free portable air compressors. Design supply will be 100 scfm for each penetration at the maximum anticipated pressure.

Initial tests of the vent systems included startup, calibration of flow vs control position at varying pressure following integrated leak rate tests, and establishment of limits with regard to the Shield Building Ventilation System.

Provisions for Sampling

Monitoring of the containment hydrogen concentration is accomplished by two Comsip Model K-111 hydrogen analyzers. As stated in reference 7, the analyzers fulfill the requirements of Item II.F.1.6 of NUREG-0737. The hydrogen monitors have indication in the control room and a range of 0% to 10% by volume under positive or negative containment pressure. The monitors are normally kept in standby mode, but indication is available on demand. The system is operated from its remote control panel located outside the high radiation sampling room. A hydrogen sample is drawn from the post-LOCA hydrogen control system sample ports in containment. These ports are located near the discharge of the containment dome fans, which permits rapid detection of hydrogen escaping from the reactor. The fans draw suction from the upper areas of containment, which prevents the formation of a stratified atmosphere. The fans are powered from safeguard buses and are designed to operate in a post-LOCA environment (see NRC SER in reference 7).

14.3.10 STEAM GENERATOR TUBE PLUGGING

Steam generator tube plugs may be periodically installed to remove tubes from service based on reported degradation. When installed, the plugs become the primary pressure boundary for the subject tube. Plugs are installed at both ends of the tube, effectively isolating tube wall defects (corrosion, etc.). Tube plugging levels up to and including 10% of the total tubes have been analyzed.

A number of plug types and designs have been qualified for use in the KNPP steam generators. Plug types include expanded mechanical plugs, rolled plugs and welded plugs. Plug integrity is ensured by the qualification of the design and installation process through laboratory testing and observed field performance. Analytical verification of plug integrity used design and operating transient parameters selected to bound those loads imposed during normal and postulated accident conditions. Fatigue and stress analysis of steam generator tube plugs were performed in accordance with the ASME B&PV Code Section III.

14.3.11 STEAM GENERATOR TUBE SLEEVING

DELETED

14.3.12 STEAM GENERATOR TUBE FATIGUE ANALYSIS

The NRC issued Bulletin 88-02 which required several actions to be implemented in order to minimize the potential for a steam generator tube rupture event caused by a rapidly propagating fatigue crack. This Bulletin is not applicable to the KNPP replacement steam generators due to material/manufacturing process upgrades.

14.3.13 VOLTAGE BASED REPAIR CRITERIA FOR STEAM GENERATOR TUBES

DELETED

14.3.14 F* AND ELEVATED F* ALTERNATE REPAIR CRITERIA FOR STEAM GENERATOR TUBES

DELETED

14.3.15 STEAM GENERATOR TUBE REMOVAL

Portions of steam generator tubes may be removed periodically for laboratory analysis to determine degradation morphology, extent, and cause. Upon removal, the affected tube portions remaining inside the steam generator are plugged on both ends to maintain the integrity of the pressure boundary. Analyses have been performed which justify tube plugging up to a level of 10% of the total tubes in the generator. The plugs installed to restore pressure boundary integrity are qualified to the requirements of ASME B&PV Code Section III.

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3. DELETED

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REFERENCES - SECTION 14.3.10 - DELETED

REFERENCES - SECTION 14.3.11 - DELETED

REFERENCES - SECTION 14.3.12 - DELETED

REFERENCES - SECTION 14.3.13 - DELETED

REFERENCES - SECTION 14.3.14 - DELETED

REFERENCES - SECTION 14.3.15 - DELETED

TABLE 5.2-1
ALLOWABLE STRESS CRITERIA - REACTOR CONTAINMENT VESSEL
Material: ASTM A516 Grade-70

Loading Conditions	Limits of Stress Intensity[1]
1. Dead loads plus operating loads plus DBA loads plus Operational Basis Earthquake Loads (0.06 g)	ASME Boiler and Pressure Vessel Code, Section III, Figure N-414
2. Dead loads plus operating loads plus DBA loads plus Design Basis Earthquake Loads (0.12 g)	Safe shutdown of plant can be achieved (a) $P_m \leq 1.16\ S_m$ (b) $P_L + P_b \leq 1.16\ (1.5\ S_m)$ (c) $P_L + P_b + Q \leq 3.0\ S_m$
3. Dead loads plus operating loads plus pipe rupture forces (faulted condition) plus (operational basis or Design Basis Earthquake Loads)	See Section 5.2-1, Pipe Reaction and Rupture Forces, and Tables B.7-2 and B.7-5, Appendix B

P_m - Primary general membrane stress intensity
P_L - Primary local membrane stress intensity
P_b - Primary bending stress intensity
Q - Secondary stress intensity
S_m - Allowable stress intensity value

[1] Refer to Table N-414, ASME Boiler and Pressure Vessel Code, Section III

TABLE 8.2-2
MAJOR SAFEGUARD BATTERY LOADS FOLLOWING LOSS OF OFF-SITE POWER
(All Loads in Amperes)

Load Description	**Continuous Load**		**Transient 1 Min. Load Following Loss of all AC**
BATTERY A Dist. Cab BRA-102	**At-Power**	**Hot Shutdown Following Loss of all AC**	
Dist. Cab. BRB-102 Alt. Feed	0	0	0
Dist. Cab. BRA-104			
Inverter BRA-111 Instrument Bus I	0	53	45
Inverter BRA-112 Instrument Bus IV	0	35	29
Balance of BRA-104	16	15	250
Total BRA-104	16	103	324
TOTAL BATTERY A	16	104	324
Battery Rating		Max @ 8 hr. 163 A	Max @ 1 min. 1234 A
Load Description	**Continuous Load**		**Transient 1 Min. Load Following Loss of all AC**
BATTERY B Dist. Cab BRB-102	**At-Power**	**Hot Shutdown Following Loss of all AC**	
Dist. Cab. BRA-102 Alt. Feed	0	0	0
Dist. Cab. BRB-104			
Inverter BRB-111 Instrument Bus II	0	49	41
Inverter BRB-112 Instrument Bus III	0	44	37
Balance of BRB-104	10	12	126
Total BRB-104	10	105	204
TOTAL BATTERY B	10	106	204
Battery Rating		Max @ 8 hr. 163 A	Max @ 1 min. 1234 A

SYSTEM INTERCONNECTION

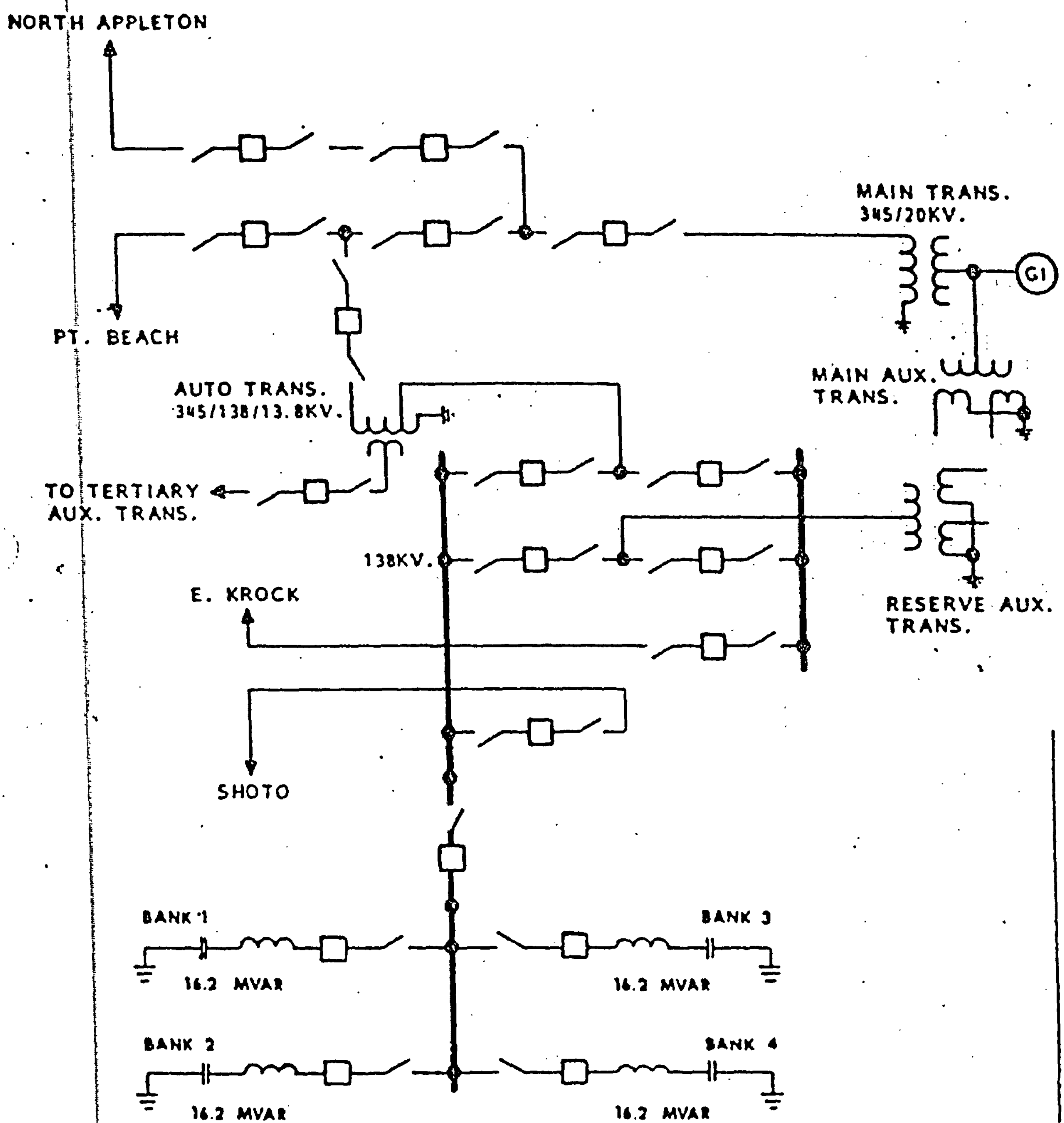
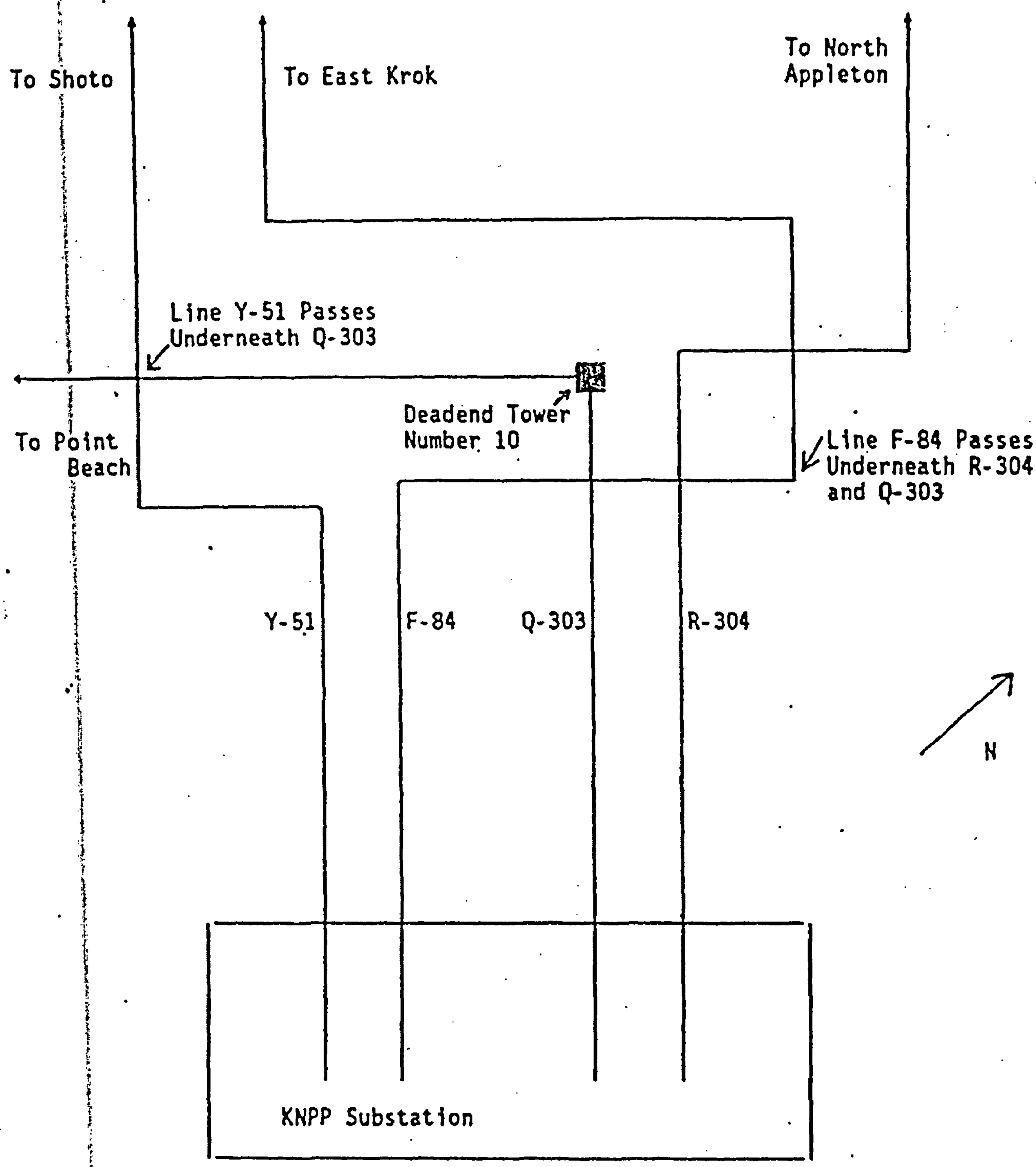


FIGURE 8 2-1

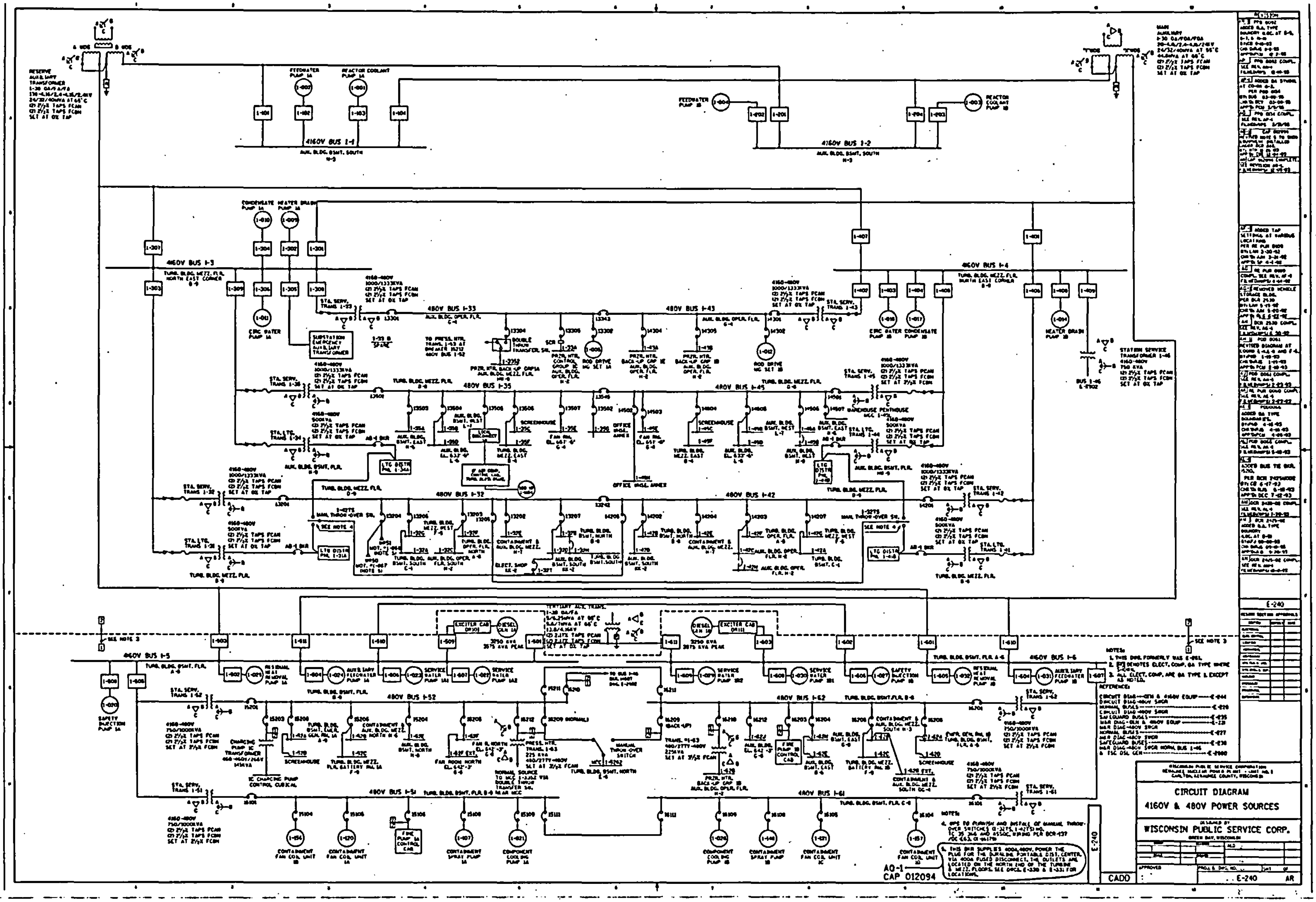
FIG8.2-1A

Offsite Power to KNPP Substation



5

Figure 8.2-1A


4160V BUS 1-1
4160V BUS 1-2
4160V BUS 1-3
4160V BUS 1-4
4160V BUS 1-5
4160V BUS 1-6
480V BUS 1-33
480V BUS 1-43
480V BUS 1-35
480V BUS 1-45
480V BUS 1-32
480V BUS 1-42
480V BUS 1-52
480V BUS 1-62
480V BUS 1-51
480V BUS 1-61
AO-1
CAP 012094
E-240
CADD
CIRCUIT DIAGRAM
4160V & 480V POWER SOURCES
WISCONSIN PUBLIC SERVICE CORP.
E-240

EPA 400-R-92-001
October 1991

MANUAL OF PROTECTIVE ACTION GUIDES

AND

PROTECTIVE ACTIONS

FOR NUCLEAR INCIDENTS

Office of Radiation Programs
United States Environmental Protection Agency
Washington, DC 20460

Revised 1991

FOREWORD

Public officials are charged with the responsibility to protect the health of the public during hazardous incidents. The purpose of this manual is to assist these officials in establishing emergency response plans and in making decisions during a nuclear incident. It provides radiological protection guidance that may be used for responding to any type of nuclear incident or radiological emergency, except nuclear war.

Under regulations governing radiological emergency planning and preparedness issued by the Federal Emergency Management Agency (47 FR 10758, March 11, 1982), the Environmental Protection Agency's responsibilities include, among others, (1) establishing Protective Action Guides (PAGs), (2) preparing guidance on implementing PAGs, including recommendations on protective actions, (3) developing and promulgating guidance to State and local governments on the preparation of emergency response plans, and (4) developing, implementing, and presenting training programs for State and local officials on PAGs and protective actions, radiation dose assessment, and decision making. This document is intended to respond to the first two responsibilities.

The manual begins with a general discussion of Protective Action Guides (PAGs) and their use in planning for protective actions to safeguard public health. It then presents PAGs for specific exposure pathways and associated time periods. These PAGs apply to all types of nuclear incidents. This is followed by guidance for the implementation of PAGs. Finally, appendices provide definitions, background information on health risks, and other information supporting the choice of the numerical values of the PAGs.

PAGs for protection from an airborne plume during the early phase of an incident at a nuclear power plant were published in the 1980 edition of this manual. These have now been revised to apply to a much broader range of situations and replace the PAGs formerly published in Chapters 2 and 5. Recommendations and background information for protection from ingestion of contaminated food were published by the Food and Drug Administration in 1982. These are reprinted here as Chapter 3 and Appendix D. Recommendations for PAGs for relocation are presented in Chapters 4 and 7. Additional radiation protection guidance for recovery will be developed at a later date. We are continuing work to develop PAGs for drinking water and, in cooperation with FDA, revised PAGs for food. When experience has been gained in the application of these PAGs, they will be reexamined and refined as necessary, proposed for review, and then recommended to the President as Federal radiation protection guidance.

This manual is being re-published to consolidate existing recommendations in a single volume. As revised and additional recommendations are developed, they will be issued as revisions to this manual. These revised PAGs are appropriate for incorporation into emergency response plans when they are revised or when new plans are developed. However, it is important to recognize that regulatory requirements for emergency response are not provided by this manual; they are established by the cognizant agency (e.g., the Nuclear Regulatory Commission in the case of commercial nuclear reactors, or the Department of Energy in the case of their contractor-operated nuclear facilities).

Users of this manual are encouraged to provide comments and suggestions for improving its contents. Comments should be sent to Allan C. B. Richardson, Criteria and Standards Division (ANR-460), Office of Radiation Programs, U.S. Environmental Protection Agency, Washington, DC 20460.

Margo T. Oge

Margo T. Oge
Director, Office of
Radiation Programs

Washington, D.C.

CONTENTS

Page

TABLES

FIGURES

APPENDICES

CHAPTER 1

Overview

1.0 Introduction

Public officials, in discharging their responsibility to protect the health of the public during hazardous situations, will usually be faced with decisions that must be made in a short period of time. A number of factors influencing the choice of protective actions will exist, so that the decisions may be complex. Further, all of the information needed to make the optimum choice will usually not be immediately available. In such situations, it will therefore be helpful if the complexity of the information upon which needed decisions are based can be reduced by careful planning during the formulation of emergency response plans.

The U.S. Environmental Protection Agency has developed this manual to assist public officials in planning for emergency response to nuclear incidents. In the context of this manual, a nuclear incident is defined as an event or a series of events, either deliberate or accidental, leading to the release, or potential release, into the environment of radioactive materials in sufficient quantity to warrant consideration of protective actions. (The term "incident" includes accidents, in the context of this manual.) A radiological emergency may result from an incident at a variety of types of facilities, including, but not limited to, those that are part of the nuclear fuel cycle, defense and research facilities, and facilities that produce or use radioisotopes, or from an incident connected with the transportation or use of radioactive materials at locations not classified as "facilities". This manual provides radiological protection criteria intended for application to all nuclear incidents requiring consideration of protective actions, other than nuclear war. It is designed for the use of those in Federal, State, and local government with responsibility for emergency response planning. The manual also provides guidance for implementation of the criteria. This has been developed primarily for incidents at nuclear power facilities. Although this implementation guidance is intended to be useful for application at other facilities or uses of radioactivity, emergency response plans will require the development of additional implementation procedures when physical characteristics of the radionuclides involved are different from those considered here.

The decision to advise members of the public to take an action to protect themselves from radiation from a nuclear incident involves a complex judgment in which the risk avoided by the protective action must be weighed in the context of the risks involved in taking the action. Furthermore, the decision may have to be made under emergency conditions, with little or no detailed information available. Therefore, considerable planning is necessary to reduce to a manageable level the complexity of decisions required to effectively protect the public at the time of an incident.

An objective of emergency planning is to simplify the choice of possible responses so that judgments are required only for viable and useful alternatives when an emergency occurs. During the planning process it is possible to make some value judgments and to determine which responses are not required, which decisions can be made on the basis of prior judgments, and which judgments must be made during an actual emergency. From this exercise, it is then possible to devise operational plans which can be used to respond to the spectrum of hazardous situations which may develop.

The main contribution to the protection of the public from abnormal releases of radioactive material is provided by site selection, design, quality assurance in construction, engineered safety systems, and the competence of staff in safe operation and maintenance. These measures can reduce both the probability and the magnitude of potential consequences of an accident. Despite these measures, the occurrence of nuclear incidents cannot be excluded. Accordingly, emergency response planning to mitigate the consequences of an incident is a necessary supplementary level of protection.

During a nuclear incident, when the source of exposure of the public is not under control, the public usually can be protected only by some form of intervention which will disrupt normal living. Such intervention is termed protective action. A Protective Action Guide (PAG) is the projected dose to reference man, or other defined individual, from an unplanned release of radioactive material at which a specific protective action to reduce or avoid that dose is recommended. The objective of this manual is to provide such PAGs for the principal protective actions available to public officials during a nuclear incident, and to provide guidance for their use.

1.1 Nuclear Incident Phases and Protective Actions

It is convenient to identify three time phases which are generally accepted as being common to all nuclear incident sequences; within each, different considerations apply to most protective actions. These are termed the early, intermediate, and late phases. Although these phases cannot be represented by precise periods and may overlap, they provide a useful framework for the considerations involved in emergency response planning.

The early phase (also referred to as the emergency phase) is the period at the beginning of a nuclear incident when immediate decisions for effective use of protective actions are required and must therefore usually be based primarily on the status of the nuclear facility (or other incident site) and the prognosis for worsening conditions. When available, predictions of radiological conditions in the environment based on the condition of the source or actual environmental measurements may also be used. Protective actions based on the PAGs may be preceded by precautionary actions during this period. This phase may last from hours to days.

The intermediate phase is the period beginning after the source and releases have been brought under control and reliable environmental measurements are available for use as a basis for decisions on additional protective actions. It extends until these additional protective actions are terminated. This phase may overlap the early and late phase and may last from weeks to many months.

The late phase (also referred to as the recovery phase) is the period beginning when recovery action designed to reduce radiation levels in the environment to acceptable levels for unrestricted use are commenced, and ending when all recovery actions have been completed. This period may extend from months to years.

The protective actions available to avoid or reduce radiation dose can be categorized as a function of exposure pathway and incident phase, as shown in Table 1-1. Evacuation and sheltering (supplemented by bathing and changes of clothing), are the principal protective actions for use during the early phase to protect the public from exposure to direct radiation and inhalation from an airborne plume. It may also be appropriate to initiate protective action for the milk supply during this period, and, in cases where emergency response plans include procedures for issuing stable iodine to reduce thyroid dose (FE-85), this may be an appropriate protective action for the early phase.

Some protective actions are not addressed by assignment of a PAG. For example, the control of access to areas is a protective action whose introduction is coupled to a decision to implement one of the other early or intermediate phase protective actions and does not have a separate PAG. And, although the use of simple, ad hoc respiratory protection may be applicable for supplementary protection in some circumstances, this protective action is primarily for use by emergency workers.

There are two types of protective actions during the intermediate phase. First, relocation and decontamination are the principal protective actions for protection of the public from whole body external exposure due to deposited material and from inhalation of any resuspended radioactive particulate materials during the intermediate and late phases. It is assumed that decisions will be made during the intermediate phase concerning whether areas from which the public has been relocated will be decontaminated and reoccupied, or condemned and the occupants permanently relocated. The second major type of protective action during the intermediate phase encompasses

TABLE 1-1. EXPOSURE PATHWAYS, INCIDENT PHASES, AND PROTECTIVE ACTIONS.

POTENTIAL EXPOSURE PATHWAYS AND INCIDENT PHASES		PROTECTIVE ACTIONS
1. External radiation from facility		Sheltering Evacuation Control of access
2. External radiation from plume	*Early*	Sheltering Evacuation Control of access
3. Inhalation of activity in plume		Sheltering Administration of stable iodine Evacuation Control of access
4. Contamination of skin and clothes		Sheltering Evacuation Decontamination of persons
5. External radiation from ground deposition of activity	*Intermediate*	Evacuation Relocation Decontamination of land and property
6. Ingestion of contaminated food and water	*Late*	Food and water controls
7. Inhalation of resuspended activity		Relocation Decontamination of land and property

Note: The use of stored animal feed and uncontaminated water to limit the uptake of radionuclides by domestic animals in the food chain can be applicable in any of the phases.

restrictions on the use of contaminated food and water. This protective action, in particular, may overlap the early and late phases.

It is necessary to distinguish between evacuation and relocation with regard to incident phases. Evacuation is the urgent removal of people from an area to avoid or reduce high-level, short-term exposure, usually from the plume or deposited activity. Relocation, on the other hand, is the removal or continued exclusion of people (households) from contaminated areas to avoid chronic radiation exposure. Conditions may develop in which some groups who have been evacuated in an emergency may be allowed to return based on the relocation PAGs, while others may be converted to relocation status.

1.2 Basis for Selecting Protective Action Guides

The PAGs in this manual incorporate the concepts and guidance contained in Federal Radiation Council (FRC) Reports 5 and 7 (FR-64 and FR-65). One of these is that the decision to implement protective actions should be based on the projected dose that would be received if the protective actions were not implemented. However, since these reports were issued, considerable additional guidance has been developed on the subject of emergency response (IC-84, IA-89). EPA considered the following four principles in establishing values for the PAGs:

1. Acute effects on health (those that would be observable within a short period of time and which have a dose threshold below which such effects are not likely to occur) should be avoided.

2. The risk of delayed effects on health (primarily cancer and genetic effects for which linear nonthreshold relationships to dose are assumed) should not exceed upper bounds that are judged to be adequately protective of public health under emergency conditions, and are reasonably achievable.

3. PAGs should not be higher than justified on the basis of optimization of cost and the collective risk of effects on health. That is, any reduction of risk to public health achievable at acceptable cost should be carried out.

4. Regardless of the above principles, the risk to health from a protective action should not itself exceed the risk to health from the dose that would be avoided.

The above principles apply to the selection of any PAG. Principles 1, 3, and 4 have been proposed for use by the international community as essential bases for decisions to intervene during an incident and Principle 2 has been recognized as an appropriate additional consideration (IA-89). Appendices C and E apply these principles to the choice of PAGs for evacuation and relocation. Although in establishing the PAGs it is prudent to consider a range of source terms to assess the costs associated with their implementation, the PAGs are chosen so as to be independent of the magnitude or type of release.

1.3 Planning

The planning elements for developing radiological emergency response plans for nuclear incidents at commercial nuclear power facilities are provided in a separate document, NUREG-0654 (NR-80), which references the PAGs in this Manual as the basis for emergency response. Planning elements for other types of nuclear incidents should be developed using similar types of considerations.

Similarly, guidance for nuclear power facilities on time frames for response, the types of releases to be considered, emergency planning zones (EPZ), and the potential effectiveness of various protective actions is provided in NUREG-0396 (NR-78). The size and shape of the recommended EPZs were only partially based on consideration of the numerical values of the PAGs. A principle additional basis was that the planning zone for evacuation and sheltering should be large enough to accommodate any urban and rural areas affected and involve the various organizations needed for emergency response. This consideration is appropriate for any facility requiring an emergency response plan involving offsite areas. Experience gained through emergency response exercises is then expected to provide an adequate basis for expanding the response to an actual incident to larger areas, if needed. It is also noted that the 10-mile radius EPZ for the early phase is large enough to avoid exceeding the PAGs for the early phase at its boundary for low-consequence, nuclear reactor, core-melt accidents and to avoid early fatalities for high-consequence, nuclear reactor core-melt accidents. The 50-mile EPZ for ingestion pathways was selected to account for the proportionately higher doses via ingestion compared to inhalation and whole body external exposure pathways.

1.4 Implementation of Protective Actions

The sequence of events during the early phase includes evaluation of conditions at the location of the incident, notification of responsible authorities, prediction or evaluation of potential consequences to the general public, recommendations for action, and implementing protection of the public. In the early phase of response, the time available to implement the most effective protective actions may be limited.

Immediately upon becoming aware that an incident has occurred that may result in exposure of the population, responsible authorities should make a preliminary evaluation to determine the nature and potential magnitude of the incident. This evaluation should determine whether conditions indicate a significant possibility of a major release and, to the extent feasible, determine potential exposure pathways, populations at risk, and projected doses. The incident evaluation and recommendations should then be presented to emergency response authorities for action. In the absence of recommendations for protective actions in specific areas from the official-responsible for the source, the emergency plan should, where practicable, provide for protective action in predesignated areas.

Contrary to the usual situation during the early phase, dose projections used to support protective action decisions during the intermediate and late phases will be based on measurements of environmental radioactivity and dose models. Following relocation of the public from affected areas to protect them from exposure to deposited materials, it will also be necessary to compile radiological and cost of decontamination data to form the basis for radiation protection decisions for recovery.

The PAGs do not imply an acceptable level of risk for normal (nonemergency conditions). They also do not represent the boundary between safe and unsafe conditions, rather, they are the approximate levels at which the associated protective actions are justified. Furthermore, under emergency conditions, in addition to the protective actions specifically identified for application of PAGs, any other reasonable measures available should be taken to minimize radiation exposure of the general public and of emergency workers.

References

FE-85 Federal Emergency Management Agency. Federal Policy on Distribution of Potassium Iodide around Nuclear Power Sites for Use as a Thyroidal Blocking Agent. Federal Register, 50, 30256; July 24, 1985.

FR-64 Federal Radiation Council. Radiation Protection Guidance for Federal Agencies. Federal Register, 29, 12056-7; August 22, 1965.

FR-65 Federal Radiation Council. Radiation Protection Guidance for Federal Agencies. Federal Register, 30, 6953-5; May 22, 1965.

IA-89 International Atomic Energy Agency. Principles for Establishing Intervention Levels for the Protection of the Public in the Event of a Nuclear Accident or Radiological Emergency. Safety Series No. 72, revision 1, International Atomic Energy Agency, Vienna (1991).

IC-84 International Commission on Radiological Protection. Protection of the Public in the Event of Major Radiation Accidents: Principles for Planning, ICRP Publication 40, Pergamon Press, Oxford (1984).

NR-78 Nuclear Regulatory Commission. Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Washington (1978).

NR-80 Nuclear Regulatory Commission. Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants. U.S. Nuclear Regulatory Commission, Washington (1980).

CHAPTER 2

Protective Action Guides for the Early Phase of an Atmospheric Release

2.1 Introduction

Rapid action may be needed to protect members of the public during an incident involving a large release of radioactive materials to the atmosphere. This chapter identifies the levels of exposure to radiation at which such prompt protective action should be initiated. These are set forth as Protective Action Guides (PAGs) for the general population. Guidance for limiting exposure of workers during such an incident is also provided. This guidance applies to any type of nuclear accident or other incident (except nuclear war) that can result in exposure of the public to an airborne release of radioactive materials.

In the case of an airborne release the principal relevant protective actions are evacuation or sheltering. These may be supplemented by additional actions such as washing and changing clothing or by using stable iodine to partially block uptake of radioiodine by the thyroid.

The former Federal Radiation Council (FRC), in a series of recommendations issued in the 1960's, introduced the concept of PAGs and issued guides for avoidance of exposure due to ingestion of strontium-89, strontium-90, cesium-137, and iodine-131. Those guides were developed for the case of worldwide atmospheric fallout from weapons testing, and are appropriate for application to intake due to long term contamination from such atmospheric releases. That is, they were not developed for protective actions relevant to prompt exposure to an airborne release from a fixed facility. The guidance in this chapter thus does not supersede this previous FRC guidance, but provides new guidance for different exposure pathways and situations.

2.1.1 Applicability

These PAGs are expected to be used for planning purposes: for example, to develop radiological emergency response plans and to exercise those plans. They provide guidance for response decisions and should not be regarded as dose limits. During a real incident, because of characteristics of the incident and local conditions that cannot be anticipated, professional judgment will be required in their application. Situations could occur, for example, in which a nuclear incident happens when environmental conditions or other constraints make evacuation impracticable. In these situations, sheltering may be the protective action of choice, even at projected doses above the PAG for evacuation. Conversely, in some cases evacuation may be useful at projected doses below the PAGs. Each case will require judgments by those responsible for decisions on protective actions at the time of an incident.

The PAGs are intended for general use to protect all of the individuals in an exposed population. To avoid social and family disruption and the complexity of implementing different PAGs for different groups under emergency conditions, the PAGs should be applied equally to most members of the population. However, there are some population groups that are at markedly different levels of risk from some protective actions -- particularly evacuation. Evacuation at higher values is appropriate for a few groups for whom the risk associated with evacuation is exceptionally high (e.g., the infirm who are not readily mobile), and the PAGs provide for this.

Some incidents may occur under circumstances in which protective actions cannot be implemented prior to a release (e.g., transportation incidents). Other incidents may involve only slow, small releases over an extended period, so that the urgency is reduced and protective action may be more appropriately treated as relocation (see Chapter 4) than as evacuation. Careful judgment will be needed to decide whether or not to apply these PAGs for the early phase under such circumstances.

The PAGs do not imply an acceptable level of risk for normal (nonemergency) conditions. PAGs also do not represent the boundary between safe and unsafe conditions; rather, they are the approximate levels at which the associated protective actions are justified. Furthermore, under emergency conditions, in addition to the protective actions specifically identified, any other reasonable measures available should be taken to reduce radiation exposure of the general public and of emergency workers. These PAGs are not intended for use as criteria for the ingestion of contaminated food or water, for relocation, or for return to an area contaminated by radioactivity. Separate guidance is provided for these situations in Chapters 3 and 4.

2.1.2 Emergency Planning Zones and the PAGs

For the purpose of identifying the size of the planning area needed to establish and test radiological emergency response plans, emergency planning zones (EPZs) are typically specified around nuclear facilities. There has been some confusion among emergency planners between these EPZs and the areas potentially affected by protective actions. It is not appropriate to use the maximum distance where a PAG might be exceeded as the basis for establishing the boundary of the EPZ for a facility. For example, the choice of EPZs for commercial nuclear power facilities has been based, primarily, on consideration of the area needed to assure an adequate planning basis for local response functions and the area in which acute health effects could occur.[1] These considerations will also be appropriate for use in selecting EPZs for most other nuclear facilities. However, since it will usually not be necessary to have offsite planning if PAGs cannot be exceeded offsite, EPZs need not be established for such cases.

2.1.3 Incident Phase

The period addressed by this chapter is denoted the "early phase." This is somewhat arbitrarily defined as the period beginning at the projected (or actual) initiation of a release and extending to a few days later, when deposition of airborne materials has ceased and enough information has become available to permit reliable decisions about the need for longer term protection. During the early phase of an incident doses may accrue both from airborne and from deposited radioactive materials. Since the dose to persons who are not evacuated will continue until relocation can be implemented (if it is necessary), it is appropriate to include in the early phase the total dose that will be received prior to such relocation. For the purpose of planning, it will usually be convenient to assume that the early phase will last for four days -- that is, that the duration of the primary release is less than four days, and that exposure to deposited materials after four days can be addressed through other protective actions, such as relocation, if this is warranted. (Because of the unique characteristics of some facilities or situations, different time periods may be more appropriate for planning purposes, with corresponding modification of the dose conversion factors cited in Chapter 5.)

[1]The development of EPZs for nuclear power facilities is discussed in the 1978 NRC/EPA document "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" NUREG-0396. EPZs for these facilities have typically been chosen to have a radius of approximately 10 miles for planning evacuation and sheltering and a radius of approximately 50 miles for planning protection from ingestion of contaminated foods.

2.2 Exposure Pathways

The PAGs for members of the public specified in this chapter refer only to doses incurred during the early phase. These may include external gamma dose and beta dose to the skin from direct exposure to airborne materials and from deposited materials, and the committed dose to internal organs from inhalation of radioactive material. Exposure pathways that make only a small contribution (e.g., less than about 10 percent) to the dose incurred in the early phase need not be considered. Inhalation of resuspended particulate materials will, for example, generally fall into this category.

Individuals exposed to a plume may also be exposed to deposited material over longer periods of time via ingestion, direct external exposure, and inhalation pathways. Because it is usually not practicable, at the time of an incident, to project these long-term doses and because different protective actions may be appropriate, these doses are not included in the dose specified in the PAGs for the early phase. Such doses are addressed by the PAGs for the intermediate phase (see Chapters 3 and 4).

The first exposure pathway from an accidental airborne release of radioactive material will often be direct exposure to an overhead plume of radioactive material carried by winds. The detailed content of such a plume will depend on the source involved and conditions of the incident. For example, in the case of an incident at a nuclear power reactor, it would most commonly contain radioactive noble gases, but may also contain radioiodines and radioactive particulate materials. Many of the these materials emit gamma radiation which can expose people nearby, as the plume passes. In the case of some other types of incidents, particularly those involving releases of alpha emitting particulate materials, direct exposure to gamma radiation is not likely to be the most important pathway.

A second exposure pathway occurs when people are directly immersed in a radioactive plume, in which case radioactive material is inhaled (and the skin and clothes may also become contaminated), e.g., when particulate materials or radioiodines are present. When this occurs, internal body organs as well as the skin may be exposed. Although exposure from materials deposited on the skin and clothing could be significant, generally it will be less important than that from radioactive material taken into the body through inhalation. This is especially true if early protective actions include washing exposed skin and changing clothing. Inhaled radioactive particulate materials, depending on their solubility in body fluids, may remain in the lungs or move via the bloodstream to other organs, prior to elimination from the body. Some radionuclides, once in the bloodstream, are concentrated in a single body organ, with only small amounts going to other organs. For example, if radioiodines are inhaled, a significant fraction moves rapidly through the bloodstream to the thyroid gland.

As the passage of a radioactive plume containing particulate material and/or radioiodine progresses, some of these materials will deposit onto the ground and other surfaces and create a third exposure pathway. People present after the plume has passed will receive exposure from gamma and beta radiation emitted from these deposited materials. If large quantities of radioiodines or gamma-emitting particulate materials are contained in a release, this exposure pathway, over a long period, can be more significant than direct exposure to gamma radiation from the passing plume.

2.3 The Protective Action Guides

The PAGs for response during the early phase of an incident are summarized in Table 2-1. The PAG for evacuation (or, as an alternative in certain cases, sheltering) is expressed in terms of the projected sum of the effective dose equivalent from external radiation and the committed effective dose equivalent incurred from inhalation of radioactive materials from exposure and intake during the early phase. (Further references to dose to members of the public in this Chapter refer to this definition, unless otherwise specified.) Supplementary guides are specified in terms of committed dose equivalent to the thyroid and dose equivalent to the skin. The PAG for the administration of stable iodine is specified in terms of the committed dose equivalent to the thyroid from radioiodine. This more complete guidance updates and replaces previous values, expressed in terms of whole-body dose equivalent from external gamma exposure and thyroid dose equivalent from inhalation of radioactive iodines, that were recommended in the 1980 edition of this document.

from evacuation itself. However, EPA recognizes the uncertainties associated with quantifying risks associated with these levels of radiation exposure, as well as the variability of risks associated with evacuation under differing conditions.

Some judgment will be necessary when considering the types of protective actions to be implemented and at what levels in an emergency situation. Although the PAG is expressed as a range of 1-5 rem, it is emphasized that, under normal conditions, evacuation of members of the general population should be initiated for most incidents at a projected dose of 1 rem. (It should be recognized that doses to some individuals may exceed 1 rem, even if protective actions are initiated within this guidance.) It is also possible that conditions may exist at specific facilities which warrant consideration of values other than those recommended for general use here.[3]

2.3.1 Evacuation and Sheltering

The basis for the PAGs is given in Appendix C. In summary, this analysis indicates that evacuation of the public will usually be justified when the projected dose to an individual is one rem. This conclusion is based primarily on EPA's judgment concerning acceptable levels of risk of effects on public health from radiation exposure in an emergency situation. The analysis also shows that, at this radiation dose, the risk avoided is usually much greater than the risk

Sheltering may be preferable to evacuation as a protective action in some situations. Because of the higher risk associated with evacuation of some special groups in the population (e.g. those who are not readily mobile), sheltering may be the preferred alternative for such groups as a

[3]EPA, in accordance with its responsibilities under the regulations governing radiological emergency planning (47FR10758; March 11, 1982) and under the Federal Radiological Emergency Response Plan, will consult with Federal agencies and the States, as requested, in such cases.

Table 2-1 PAGs for the Early Phase of a Nuclear Incident

Protective Action	PAG (projected dose)	Comments
Evacuation (or sheltering[a])	1-5 rem[b]	Evacuation (or, for some situations, sheltering[a]) should normally be initiated at 1 rem. Further guidance is provided in Section 2.3.1
Administration of stable iodine	25 rem[c]	Requires approval of State medical officials.

[a]Sheltering may be the preferred protective action when it will provide protection equal to or greater than evacuation, based on consideration of factors such as source term characteristics, and temporal or other site-specific conditions (see Section 2.3.1).

[b]The sum of the effective dose equivalent resulting from exposure to external sources and the committed effective dose equivalent incurred from all significant inhalation pathways during the early phase. Committed dose equivalents to the thyroid and to the skin may be 5 and 50 times larger, respectively.

[c]Committed dose equivalent to the thyroid from radioiodine.

protective action at projected doses up to 5 rem. In addition, under unusually hazardous environmental conditions use of sheltering at projected doses up to 5 rem to the general population (and up to 10 rem to special groups) may become justified. Sheltering may also provide protection equal to or greater than evacuation due to the nature of the source term and/or in the presence of temporal or other site-specific conditions. Illustrative examples of situations or groups for which evacuation may not be appropriate at 1 rem include: a) the presence of severe weather, b) competing disasters, c) institutionalized persons who are not readily mobile, and d) local physical factors which impede evacuation. Examples of situations or groups for which evacuation at 1 rem normally would be appropriate include: a) an incident which occurs at night, b) an incident which occurs when children are in school, and c) institutionalized persons who are readily mobile. Evacuation seldom will be justified at less than 1 rem. The examples described above regarding selection of the most appropriate protective action are intended to be illustrative and not exhaustive. In general, sheltering should be preferred to evacuation whenever it provides equal or greater protection.

No specific minimum level is established for initiation of sheltering. Sheltering in place is a low-cost, low-risk protective action that can provide protection with an efficiency ranging from zero to almost 100 percent, depending on the circumstances. It can also be particularly useful to assure that a population is positioned so that, if the need arises, communication with the population can be carried out expeditiously. For the above reasons, planners and decision makers should consider implementing sheltering at projected doses below 1 rem; however, implementing protective actions for projected doses at very low levels would not be reasonable (e.g. below 0.1 rem). (This guidance should not be construed as establishing an additional lower level PAG for sheltering.) Sheltering should always be implemented in cases when evacuation is not carried out at projected doses of 1 rem or more.

Analyses for some hypothesized accidents, such as short-term releases of transuranic materials, show that sheltering in residences and other buildings can be highly effective at reducing dose, may provide adequate protection, and may be more effective than evacuation when evacuation cannot be completed before plume arrival (DO-90). However, reliance on large dose reduction factors for sheltering should be accompanied by cautious examination of possible failure mechanisms, and, except in very unusual circumstances, should never be relied upon at projected doses greater than 10 rem. Such analyses should be based on realistic or "best estimate" dose models and include unavoidable dose during evacuation. Sheltering and evacuation are discussed in more detail in Section 5.5.

2.3.2 Thyroid and Skin Protection

Since the thyroid is at disproportionately high risk for induction of nonfatal cancer and nodules, compared to other internal organs, additional guidance is provided to limit the risk of these effects (see footnote to Table 2-1). In addition, effective dose, the quantity used to express the PAG, encompasses only the risk of fatal cancer from irradiation of organs within the body, and does not include dose to skin. Guidance is also provided, therefore, to protect against the risk of skin cancer (see Table 2-1, footnote b).

The use of stable iodine to protect against uptake of inhaled radioiodine by the thyroid is recognized as an effective alternative to evacuation for situations involving radioiodine releases when evacuation cannot be implemented or exposure occurs during evacuation. Stable iodine is most effective when administered immediately prior to exposure to radioiodine. However, significant blockage of the thyroid dose can be provided by administration within one or two hours after uptake of radioiodine. If the administration of stable iodine is included in an emergency response plan, its use may be considered for exposure situations in which the committed dose equivalent to the thyroid can be 25 rem or greater (see 47 FR 28158; June 29, 1982).

Washing and changing of clothing is recommended primarily to provide protection from beta radiation from radioiodines and particulate materials deposited on the skin or clothing. Calculations indicate that dose to skin should seldom, if ever, be a controlling pathway. However, it is good radiation protection practice to recommend these actions, even for alpha-emitting radioactive materials, as soon as practical for persons significantly exposed to a contaminating plume (i.e., when the projected dose from inhalation would have justified evacuation of the public under normal conditions).

2.4 Dose Projection

The PAGs are expressed in terms of projected dose. However, in the early phase of an incident (either at a nuclear facility or other accident site), parameters other than projected dose may frequently provide a more appropriate basis for decisions to implement protective actions. When a facility is operating outside its design basis, or an incident is imminent but has not yet occurred, data adequate to directly estimate the projected dose may not be available. For such cases, provision should be made during the planning stage for decisions to be made based on specific conditions at the source of a possible release that are relatable to ranges of anticipated offsite consequences. Emergency response plans for facilities should make use of Emergency Action Levels (EALs)[4], based on in-plant conditions, to trigger notification of and recommendations to offsite officials to implement prompt evacuation or sheltering in specified areas in the absence of information on actual releases or environmental measurements. Later, when these data become available, dose projections based on measurements may be used, in addition to plant conditions, as the basis for implementing further protective actions. (Exceptions may occur at sites with large exclusion areas where some field and source data may be available in sufficient time for protective action decisions to be based on environmental measurements.) In the case of transportation accidents or other incidents that are not related to a facility, it will often not be practicable to establish EALs.

The calculation of projected doses should be based on realistic dose

[4]Emergency Action Levels related to plant conditions at commercial nuclear power plants are discussed in Appendix 1 to NUREG-0654 (NR-80).

models, to the extent practicable. Doses incurred prior to initiation of a protective action should not normally be included. Similarly, doses that might be received following the early phase should not be included for decisions on whether or not to evacuate or shelter. Such doses, which may occur from food and water, long-term radiation exposure to deposited radioactive materials, or long-term inhalation of resuspended materials, are chronic exposures for which neither emergency evacuation nor sheltering are appropriate protective actions. Separate PAGs relate the appropriate protective action decisions to those exposure pathways (Chapter 4). As noted earlier, the projection of doses in the early phase need include only those exposure pathways that contribute a significant fraction (e.g., more than about 10 percent) of the dose to an individual.

In practical applications, dose projection will usually begin at the time of the anticipated (or actual) initiation of a release. For those situations where significant dose has already occurred prior to implementing protective action, the projected dose for comparison to a PAG should not include this prior dose.

2.5 Guidance for Controlling Doses to Workers Under Emergency Conditions

The PAGs for protection of the general population and dose limits for workers performing emergency services are derived under different assumptions. PAGs consider the risks to individuals, themselves, from exposure to radiation, and the risks and costs associated with a specific protective action. On the other hand, workers may receive exposure under a variety of circumstances in order to assure protection of others and of valuable property. These exposures will be justified if the maximum risks permitted to workers are acceptably low, and the risks or costs to others that are avoided by their actions outweigh the risks to which workers are subjected.

Workers who may incur increased levels of exposure under emergency conditions may include those employed in law enforcement, fire fighting, radiation protection, civil defense, traffic control, health services, environmental monitoring, transportation services, and animal care. In addition, selected workers at institutional, utility, and industrial facilities, and at farms and other agribusiness may be required to protect others, or to protect valuable property during an emergency. The above are examples - not designations - of workers that may be exposed to radiation under emergency conditions.

Guidance on dose limits for workers performing emergency services is summarized in Table 2-2. These limits apply to doses incurred over the duration of an emergency. That is, in contrast to the PAGs, where only the future dose that can be avoided by a specific protective action is considered, all doses received during an emergency are included in the limit. Further, the dose to workers performing emergency

Table 2-2 Guidance on Dose Limits for Workers Performing Emergency Services

Dose limit[a] (rem)	Activity	Condition
5	all	
10	protecting valuable property	lower dose not practicable
25	life saving or protection of large populations	lower dose not practicable
>25	lifesaving or protection of large populations	only on a voluntary basis to persons fully aware of the risks involved (See Tables 2-3 and 2-4)

[a]Sum of external effective dose equivalent and committed effective dose equivalent to nonpregnant adults from exposure and intake during an emergency situation. Workers performing services during emergencies should limit dose to the lens of the eye to three times the listed value and doses to any other organ (including skin and body extremities) to ten times the listed value. These limits apply to all doses from an incident, except those received in unrestricted areas as members of the public during the intermediate phase of the incident (see Chapters 3 and 4).

services may be treated as a once-in-a-lifetime exposure, and not added to occupational exposure accumulated under nonemergency conditions for the purpose of ascertaining conformance to normal occupational limits, if this is necessary. However, any radiation exposure of workers that is associated with an incident, but accrued during nonemergency operations, should be limited in accordance with relevant occupational limits for normal situations. Federal Radiation Protection Guidance for occupational exposure recommends an upper bound of five rem per year for adults and one tenth this value for minors and the unborn (EP-87). We recommend use of this same value here for the case of exposures during an emergency. To assure adequate protection of minors and the unborn during emergencies, the performance of emergency services should be limited to nonpregnant adults. As in the case of normal occupational exposure, doses received under emergency conditions should also be maintained as low as reasonably achievable (e.g., use of stable iodine, where appropriate, as a prophylaxis to reduce thyroid dose from inhalation of radioiodines and use of rotation of workers).

Doses to all workers during emergencies should, to the extent practicable, be limited to 5 rem. There are some emergency situations, however, for which higher exposure limits may be justified. Justification of any such exposure must include the presence of conditions that prevent the rotation of workers or other commonly-used dose reduction methods. Except as noted below, the dose resulting from such emergency exposure should be limited to 10 rem for protecting valuable property, and to 25 rem for life saving activities and the protection of large populations. In the context of this guidance, exposure of workers that is incurred for the protection of large populations may be considered justified for situations in which the collective dose avoided by the emergency operation is significantly larger than that incurred by the workers involved.

Situations may also rarely occur in which a dose in excess of 25 rem for emergency exposure would be unavoidable in order to carry out a lifesaving operation or to avoid extensive exposure of large populations. It is not possible to prejudge the risk that one should be allowed to take to save the lives of others. However, persons undertaking any emergency operation in which the dose will exceed 25 rem to the whole body should do so only on a voluntary basis and with full awareness of the risks involved, including the numerical levels of dose at which acute effects of radiation will be incurred and numerical estimates of the risk of delayed effects.

Tables 2-3 and 2-4 provide some general information that may be useful in advising emergency workers of risks of acute and delayed health effects associated with large doses of radiation. Table 2-3 presents estimated risks of early fatalities and moderately severe prodromal (forewarning) effects that are likely to occur shortly after exposure to a wide range of whole body radiation doses. Estimated average cancer mortality risks for emergency workers corresponding to a whole-body dose equivalent of 25 rem are given in Table 2-4, as a function of age at the time of exposure. To estimate average cancer mortality for moderately higher doses the results in Table 2-4 may be increased linearly. These values were calculated using a life table analysis that assumes the period of risk continues for the duration of the worker's lifetime. Somewhat smaller risks of serious genetic effects (if gonadal tissue is exposed) and of nonfatal cancer would also be incurred. An expanded discussion of health effects from radiation dose is provided in Appendix B.

Some workers performing emergency services will have little or no health physics training, so dose minimization through use of protective equipment cannot always be assumed. However, the use of respiratory protective equipment can reduce dose from inhalation, and clothing can reduce beta dose. Stable iodine is also recommended for blocking thyroid

Table 2-3 Health Effects Associated with Whole-Body Absorbed Doses Received Within a Few Hours[a] (see Appendix B)

Whole Body Absorbed dose (rad)	Early Fatalities[b] (percent)	Whole Body Absorbed dose (rad)	Prodromal Effects[c] (percent affected)
140	5	50	2
200	~~5~~ 15 per EPA letter	100	15
300	50	150	50
400	85	200	85
460	95	250	98

[a]Risks will be lower for protracted exposure periods.

[b]Supportive medical treatment may increase the dose at which these frequencies occur by approximately 50 percent.

[c]Forewarning symptoms of more serious health effects associated with large doses of radiation.

Table 2-4 Approximate Cancer Risk to Average Individuals from 25 Rem Effective Dose Equivalent Delivered Promptly (see Appendix C)

Age at exposure (years)	Appropriate risk of premature death (deaths per 1,000 persons exposed)	Average years of life lost if premature death occurs (years)
20 to 30	9.1	24
30 to 40	7.2	19
40 to 50	5.3	15
50 to 60	3.5	11

uptake of radioiodine in personnel involved in emergency actions where atmospheric releases include radioiodine. The decision to issue stable iodine should include consideration of established State medical procedures, and planning is required to ensure its availability and proper use.

References

DO-90 U.S. Department of Energy. Effectiveness of Sheltering in Buildings and Vehicles for Plutonium, DOE/EH-0159, U.S. Department of Energy, Washington (1990).

EP-87 U.S. Environmental Protection Agency. Radiation Protection Guidance to Federal Agencies for Occupational Exposure. Federal Register, 52, 2822; January 27, 1987.

NR-80 U.S. Nuclear Regulatory Commission. Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants. NUREG-0654, U.S. Nuclear Regulatory Commission, Washington, (1980).

CHAPTER 3

Protective Action Guides for the Intermediate Phase (Food and Water)

a) Accidental Radioactive Contamination of Human Food and Animal Feeds; Recommendations for State and Local Agencies*

b) Drinking Water**

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* These recommendations were published by FDA in 1982.

**Protective action recommendations for drinking water are under development by EPA.

DEPARTMENT OF HEALTH AND HUMAN SERVICES

Food and Drug Administration

[Docket No. 76N-0050]

Accidental Radioactive Contamination of Human Food and Animal Feeds; Recommendations for State and Local Agencies

AGENCY: Food and Drug Administration.

ACTION: Notice.

SUMMARY: The Food and Drug Administration (FDA) is publishing this notice to provide to State and local agencies responsible for emergency response planning for radiological incidents recommendations for taking protective actions in the event that an incident causes the contamination of human food or animal feeds. These recommendations can be used to determine whether levels of radiation encountered in food after a radiological incident warrant protective action and to suggest appropriate actions that may be taken if action is warranted. FDA has a responsibility to issue guidance on

appropriate planning actions necessary for evaluating and preventing contamination of human food and animal feeds and on the control and use of these products should they become contaminated.

FOR FURTHER INFORMATION CONTACT:
Gail D. Schmidt, Bureau of Radiological Health (HFX-1), Food and Drug Administration, 5600 Fishers Lane, Rockville, MD 20857, 301-443-2850.

SUPPLEMENTARY INFORMATION:

Background

This guidance on accidental radioactive contamination of food from fixed nuclear facilities, transportation accidents, and fallout is part of a Federal interagency effort coordinated by the Federal Emergency Management Agency (FEMA). FEMA issued a final regulation in the Federal Register of March 11, 1982 (47 FR 10758), which reflected governmental reorganizations and reassigned agency responsibilities for radiological incident emergency response planning. A responsibility assigned to the Department of Health and Human Services (HHS) (and in turn delegated to FDA) is the responsibility to develop and specify to State and local governments protective actions and associated guidance for human food and animal feed.

In the Federal Register of December 15, 1978 (43 FR 58790), FDA published proposed recommendations for State and local agencies regarding accidental radioactive contamination of human food and animal feeds. Interested persons were given until February 13, 1979 to comment on the proposal. Twenty-one comments were received from State agencies, Federal agencies, nuclear utilities, and others. Two of the comments from environmentally concerned organizations were received after the March 28, 1979 accident at Three Mile Island, which increased public awareness of protective action guidance. Although these comments were received after the close of the comment period, they were considered by the agency in developing these final recommendations.

The Office of Radiation Programs, Environmental Protection Agency (EPA), submitted a detailed and exhaustive critique of the proposed recommendations. EPA addressed the dosimetry data, the agricultural models used in calculating the derived response levels, and the philosophical basis for establishing the numerical value of the protective action guides. FDA advises that, to be responsive to the EPA comments, FDA staff met with staff of the Office of Radiation Programs, EPA, during the development of these final recommendations. Although EPA's formal comments are responded to in this notice, EPA staff reviewed a draft of the final recommendations, and FDA has considered their additional informal comments. These contacts were considered appropriate because EPA has indicated that it intends to use the recommendations as the basis for revising its guidance to Federal agencies on protective action guides for radioactivity in food.

Protective Action Guidance

Although not raised in the comments received, FDA has reconsidered its proposal to codify these recommendations in 21 CFR Part 1090. Because these recommendations are voluntary guidance to State and local agencies (not regulations), FDA has decided not to codify the recommendations; rather, it is issuing them in this notice. Elsewhere in this issue of the Federal Register, FDA is withdrawing the December 15, 1978 proposal.

The recommendations contain basic criteria, defined as protective action guides (PAG's), for establishing the level of radioactive contamination of human food or animal feeds at which action should be taken to protect the public health and assure the safety of food. The recommendations also contain specific guidance on what emergency protective actions should be taken to prevent further contamination of food or feeds or to restrict the use of food, as well as more general guidance on the development and implementation of emergency action. The PAG's have been developed on the basis of considerations of acceptable risk to identify that level of contamination at which action is necessary to protect the public health.

In preparing these recommendations, FDA has reviewed and utilized the Federal guidance on protective actions contained in Federal Radiation Council (FRC) Reports No. 5, July 1964 (Ref. 1) and No. 7, May 1965 (Ref. 2). The Federal guidance provides that each Federal agency, by virtue of its immediate knowledge or its operating problems, would use the applicable FRC guides as a basis for developing detailed standards to meet the particular needs of the agency. FDA's recommendations incorporate the FRC concepts and the FRC guidance that protective actions, in the event of a contaminating accident, should be based on estimates of the projected radiation dose that would be received in the absence of taking protective actions. Similarly, protective actions should be implemented for a sufficient time to avoid most of the projected radiation dose. Thus, the PAG's define the numerical value of projected radiation doses for which protective actions are recommended.

FDA has reviewed the recent report of the National Academy of Sciences/National Research Council (Ref. 3) on radiation risks and biological effects data that became available after publication of the FRC guidance and has reviewed the impact of taking action in the pasture/cow/milk/person pathway in light of the current concerns in radiation protection. Based on these considerations and the comments received on the proposed recommendations, FDA has concluded that protective actions of low impact should be undertaken at projected radiation doses lower than those recommended by FRC (Refs. 1 and 2). Accordingly, FDA is recommending low-impact protective actions (termed the Preventive PAG) at projected radiation doses of 0.5 rem whole body and 1.5 rem thyroid. FDA intends that such protective actions be implemented to prevent the appearance of radioactivity in food at levels that would require its condemnation. Preventive PAG's include the transfer of dairy cows from fresh forage (pasture) to uncontaminated stored feed and the diversion of whole milk potentially contaminated with short-lived radionuclides to products with a long shelf life to allow radioactive decay of the radioactive material.

In those situations where the only protective actions that are feasible present high dietary and social costs or impacts (termed the Emergency PAG) action is recommended at projected radiation doses of 5 rem whole body and 15 rem thyroid. At the Emergency PAG level responsible officials should isolate food to prevent its introduction into commerce and determine whether condemnation or other disposition is appropriate. Action at the Emergency PAG level is most likely for the population that is near to the source of radioactive contamination and that consumes home-grown produce and milk.

The PAG's represent FDA's judgment as to that level of food contamination resulting from radiation incidents at which action should be taken to protect the public health. This is based on the agency's recognition that safety involves the degree to which risks are judged acceptable. The risk from natural disasters (approximately a one in a million annual individual risk of death and the risk from variations in natural background radiation have provided perspective in selecting the PAC values. This issue is further discussed in the responses to specific comments later in this notice, especially in paragraph 9. A more detailed treatment of the rationale, risk factors, dosimetric and agricultural models, and methods of calculation is contained in the "Background for Protective Action Recommendations: Accidental Radioactive Contamination of Food and Animal Feeds" (Ref. 22).

Organ PAG Values

Current scientific evidence, as reflected by BEIR-I (Ref. 18), UNSCEAR-1977 (Ref. 8), and BEIR-III (Ref. 3), indicates that the relative importance of risk due to specific organ exposure is quite different from the earlier assumptions. The International Commission on Radiological Protection (ICRP) clearly recognized this in its 1977 recommendations (ICRP-26 (Ref. 6)), which changed the methodology for treating external and internal radiation doses and the relative importance of specific organ doses. ICRP-26 assigned weighting factors to specific organs based on considerations of the incidence and severity (mortality) of radiation cancer induction. For the radionuclides of concern for food PAG's, ICRP-26 assigned weighting factors of 0.03 for the thyroid and 0.12 for red bone marrow. Thus, the organ doses equal in risk to 1 rem whole body radiation dose are 33 rem to the thyroid and 8 rem to Red bone marrow. (The additional ICRP-26, nonstochastic limit, however, restricts the thyroid dose to 50 rem or 10 times the whole body occupational limit of 5 rem.)

In the Federal Register of January 23, 1981 (46 FR 7836), EPA proposed to revise the Federal Radiation Protection Guidance for Occupational Exposures using the ICRP approach for internal organ radiation doses, modified to reflect specific EPA concerns. The EPA proposal has been subject to considerable controversy. Also, the National Council on Radiation Protection and Measurements (NCRP) curently is evaluating the need to revise its recommendations. FDA does not, however, expect the protection model for internal organ radiation doses to be resolved rapidly in the United States and has based the relative PAG dose assignments in these recommendations on current U.S. standards and the 1971 recommendations in NCRP-39 (Ref. 19). Thus, the red bone marrow is assigned the same PAG dose as the whole body (0.5 rem Preventive PAG), and the thyroid PAG is greater by a factor of three (1.5 rem Preventive PAG). This results in PAG assignments for the thyroid and red bone marrow that are lower by factors of 3.3 and 8, respectively, than values based on ICRP-26 (Ref. 6). FDA advises that it will make appropriate changes in recommendations for internal organ doses when a consensus in the United States emerges.

Analysis of Comments

The following is a summary of the comments received on the December 15, 1978 proposal and the agency's response to them:

1. Several comments requested clarification of the applicability and compatibility of FDA's recommendations with other Federal actions, specifically the PAG guidance of EPA (Ref. 7), the FRC Reports No. 5 (Ref. 1) and No. 7 (Ref. 2), and the Nuclear Regulatory Commission (NRC) definition of "Extraordinary Nuclear Occurrence" in 10 CFR Part 140. A comment recommended that the term, "Protective Action Guide (PAG)", not be used because that term traditionally has been associated with the FRC, and the general public would confuse FDA's recommendations with Federal guidance.

The FRC Report No. 5 specifically recommended that the term, "protective action guide," be adopted for Federal use. The report defines the term as the "projected absorbed dose to the individuals in the general population which warrants protective action following a contaminating event," a concept that is addressed by FDA's recommendations. To use the concept with a different description would, in FDA's opinion, be unnecessarily confusing to State and local agencies as well as Federal agencies.

These recommendations are being issued to fulfill the HHS responsibilities under FEMA's March 11, 1982 regulation. FDA fully considered FRC Reports No. 5 and No. 7 and the basic concepts and philosophy of the FRC guidance form the basis for these recommendations. The specific PAG values are derived response levels included in these recommendations are based on current agricultural pathway and radiation dose models and current estimates of risk. The FRC guidance provided that protective actions may be justified at lower (or higher) projected radiation doses depending on the total impact of the protective action. Thus, FDA's recommendation that protective actions be implemented at projected radiation doses lower than those recommended by FRC doses is consistent with the FRC guidance. The FRC guidance is applicable to Federal agencies in their radiation protection activities. FDA's recommendations are for use by State and local agencies in response planning and implementation of protective actions in the event of a contaminating incident. Further, FDA's recommendations would also be used by FDA in implementing its authority for food in interstate commerce under the Federal Food, Drug, and Cosmetic Act.

FDA's recommendations are being forwarded to EPA as the basis for revising Federal guidance on food accidentally contaminated by radionuclides. EPA has advised FDA that it intends to forward the FDA recommendations to the President under its authority to "advise the President with respect to radiation matters directly or indirectly affecting health, including guidance for all Federal agencies in the formulation of radiation standards * * *". (This authority was transferred to EPA in 1970 when FRC was abolished.)

The recommendations established in this document apply only to human food and animal feeds accidentally contaminated by radionuclides. They should not be applied to any other source of radiation exposure. EPA already has issued protective action guidance for the short-term accidental exposure to airborne releases of radioactive materials and intends also to forward the EPA guides to the President as Federal guidance. EPA also is considering the development of guidance for acidentally contaminated water and for long-term exposures due to contaminated land, property, and materials. Guidance for each of these exposure pathways is mutually exclusive. Different guidance for each exposure pathway is appropriate because different criteria of risk, cost, and benefit are involved. Also, each exposure pathway may involve different sets of protective or restorative actions and would relate to different periods of time when such actions would be taken.

2. Several comments expressed concern about radiation exposure from multiple radionuclides and from multiple pathways, e.g., via inhalation, ingestion, and external radiation from the cloud (plume exposure) and questioned why particular pathways or radionuclides and the does received before assessment were not addressed in the recommendations. Several comments recommended that the PAG's include specific guidance for tap water (and potable water). Other comments noted that particular biological forms of specific radionuclides (i.e., cyanocobalamin Co 60), would lead to significantly different derived response levels.

FDA advises that the PAG's and the protective action concepts of FRC apply to actions taken to avoid or prevent projected radiation dose (or future dose). Thus, by definition, the PAG's for food do not consider the radiation doses already incurred from the plume pathway or from other sources. The population potentially exposed by ingestion of contaminated food can be divided into that population near the source of contamination and a generally much larger population at distances where the doses from the cloud are not significant. The NRC regulations provide that State and local planning regarding plume exposure should extend for 10 miles and the ingestion pathway should extend for 50 miles (see 45 FR 55402; August 19, 1980). The total population exposed by ingestion, however, is a function of the animal feed and human food production of any given area and is not limited by distance from the source of contamination. Exposure from multiple pathways would not be a concern for the more distant population group. Further, individuals in this larger population would most likely receive doses smaller than that projected for continuous intake because the contaminated food present in the retail distribution system would be replaced by uncontaminated food.

FRC Report No. 5 states that, for repetitive occurrences, the total projected radiation dose and the total impact of protective actions should be considered. Similar considerations on a case-by-case basis would then appear to be appropriate in the case of multiple exposures from the plume and the ingestion pathway. Accordingly, the final recommendations are modified to note that, specifically in the case of the population near the site that consumes locally grown produce, limitations of the total dose should be considered (see paragraph (a)(2)). The agency concludes, however, that a single unified PAG covering multiple pathways, e.g., external radiation, inhalation, and ingestion is not practical because different actions and impacts are involved. Further, FDA's responsibility in radiological incident emergency response planning extends only to human food and animal feeds.

The agency's primary charge is to set recommended PAG dose commitment limits for the food pathway. Thus, deriving response levels for only the radionuclides most likely to enter the food chain and deliver the highest dose to the population permits FDA to establish recommendations that are practical for use in an emergency. In discussing with EPA the list of definitive models, FDA and EPA staffs agreed that further pathway studies would be useful. Elsewhere in this notice, FDA references models for other radionuclides, providing a resource for those requiring more details.

The chemical form of radionuclides in the environment may be important when considering the derivation of an appropriate "response level" in specific situations, but would not change the PAG's, which are in terms of projected dose commitments. Cyanocobalamin Co 60 has not been identified as a likely constituent of health importance to be released from a nuclear reactor accident and, therefore, the agency rejects the recommendation that it provide derived response levels for this radionuclide. However, after reviewing current agricultural and dose models, the agency concludes that cesium-134 would likely be released and has added it to the tables in paragraph (d) of the recommendations identifying radionuclide concentrations equivalent to the PAG response levels.

FDA rejects the comment recommending that the PAG's include guidance for water. A memorandum of understanding between EPA and FDA provides that FDA will have primary responsibility over direct and indirect additives and other substances in drinking water (see 44 FR 42775; July 20, 1979). Thus, FDA defers to EPA for developing guides specifically for drinking water.

3. Three comments requested clarification of the proposed recommendations, including the time over which the guides apply, the time of ingestion required to reach the PAG, and the time that protective actions should be implemented.

FDA advises that the recommendations are intended to provide guidance for actions to be implemented in an emergency, and the duration of protective action should not exceed 1 or 2 months. The agency believes that the actions identified in paragraphs (a) and (h) of the recommendations should be continued for a sufficient time to avoid most of the emergency radiation dose and to assure that the remaining dose is less than the Preventive PAG. This period of time can be estimated by considering the effective half-life of the radioactive material taking into account both radioactive decay and weathering. Each case must be examined separately considering the actual levels of contamination and the effective half-life of the radioactive material present. For the pasture/cow/milk pathway, the effective half-lives are 5 days for iodine-131 and 14 days for cesium or strontium. Assuming that initial contamination by these radionuclides was at the Preventive PAG level, radioactive decay and weathering would reduce the level so that protective actions could be ceased after 1 or 2 months.

The model used to compute the derived response levels specified in paragraph (d) of the recommendations assumes a continous or infinite ingestion period, i.e., intake that is limited only by radioactive decay and weathering. This is the approach recommended in estimating the projected radiation dose (in the absence of protective actions.). Further revisions have been made in the recommendations to clarify these aspects.

4. A comment stated that action should be initiated by notification received from the facility itself. Another comment noted the importance of timely announcements to the public of the necessity for protective actions.

These recommendations on protective action guides for food and feed are not intended to cover other aspects of emergency planning for radiological incidents. The general responsibilities of NRC licensees in radiation emergencies have been further defined in a rule issued by NRC (45 FR 55402; August 19, 1980). FDA recognizes, however, that notification and public announcements are vital to effective protective actions and, in paragraph (e)(5) of the recommendations, urges that State and local emergency plans should provide for such notice.

5. A comment offered clarification of proposed § 1000.400(g) regarding verification of sample measurements, while another comment suggested that Preventive PAG's should be based on projected levels and that Emergency PAG.s require verification.

The FRC concepts and philosophy, which FDA fully endorses, use estimates of projected radiation dose as the criteria for taking protective action. FDA believes that projected radiation dose estimates should be based on verified measurements of radioactivity in the food pathway. Such verification might include the analysis of replicate samples, laboratory measurements, sample analysis by other agencies, samples of various environmental media, and descriptive data of the radioactive release and has so provided in paragraph (g) of the recommendations.

6. A comment suggested that some States do not have the resources to evaluate projected radiation doses. The comment asked what regulatory agenc would have control over interstate

shipment of contaminated foods from States without sufficient resources and what would be the applicable PAG.

FEMA, as the lead agency for the Federal effort, is providing to States guidance and assistance on emergency response planning including evaluation of projected doses. Also, NRC requires nuclear power plant licensees to have the capability to assess the off-site consequences of radioactivity releases and to provide notification to State and local agencies (45 FR 55402; August 19, 1980). FDA has authority under the Federal Food, Drug, and Cosmetic Act to remove radioactively contaminated food from the channels of interstate commerce. In this circumstance, FDA would use these PAG recommendations as the basis for implementing regulatory action.

Risk Estimates

7. Many comments questioned the risk estimates on which FDA based the proposed PAG's. The comments especially suggested that risk estimates from WASH-1400 (Ref. 4) were of questionable validity. Other comments argued that the proposed recommendations used an analysis of only lethal effects; that they used an absolute risk model; and that genetic effects were not adequately considered. The risk estimates themselves were alleged to be erroneous because recent studies show that doubling doses are lower than are those suggested by WASH-1400. The tinea capitis study by Ron and Modan, which indicates an increased probability of thyroid cancer at an estimated radiation dose of 9 rem to the thyroid (Ref. 5), was cited as evidence that the PAG limits for the thyroid were too high. The comments requested further identification and support for using the critical population selected.

Most of these issues were addressed in the preamble to the FDA proposal. The final recommendations issued in this notice employ the most recent risk estimates (somatic and genetic) of the National Academy of Sciences Committee on Biological Effects of Ionizing Radiation (Ref. 3).

The thyroid PAG limits are based on the relative radiation protection guide for thyroid compared to whole body contained in NRC's current regulations (10 CFR Part 20). The derived response levels for thyroid are based on risk factors for external x-ray irradiation. Therefore, the criticism of the PAG limits for the thyroid is not applicable, no "credit" having been taken for an apparent lower radiation risk due to Iodine-131 irradiation of the thyroid gland. Further, as discussed above under "ORGAN PAG VALUES", the use of BEIR-III risk estimates or the ICRP-26 recommendations would result in an increase of the thyroid PAG relative to the whole body PAG. For these reasons, FDA believes the PAG limits for projected dose commitment to the thyroid are conservative when considered in light of current knowledge of radiation to produce equal health risks from whole body and specific organ doses.

Although it may be desirable to consider total health effects, not just lethal effects, there is a lack of data for total health effects to use in such comparisons. In the case of the variability of natural background, as an estimate of acceptable risk, consideration of lethal effects or total health effects is not involved because the comparison is the total dose over a lifetime.

Rational

8. Several comments questioned the rational FDA used in setting the specific PAG values included in the December 1978 proposal. A comment from EPA stated that the guidance levels should be justified on the grounds that it is not practical or reasonable to take protective actions at lower risk levels. Further, EPA argued that the protective action concept for emergency planning and response should incorporate the principle of keeping radiation exposures as low as reasonably achievable (ALARA). EPA noted that the principle of acceptable risk involves a perception of risk that may vary from person to person and that the implication that an acceptable genetic risk has been established should be avoided.

FDA accepts and endorses the ALARA concept, but the extent to which a concept, which is used in occupational settings, should be applied to emergency protective actions is not clear. To use the ALARA concept as the basis for specific PAG values and also require ALARA during the implementation of emergency protective actions appears to be redundant and may not be practical under emergency conditions.

FDA advises that these guides do not constitute acceptable occupational radiation dose limits nor do they constitute acceptable limits for other applications (e.g., acceptable genetic risk). The guides are not intended to be used to limit the radiation dose that people may receive but instead are to be compared to the calculated projected dose, i.e., the future dose that the people would receive if no protective action were taken in a radiation emergency. In this respect, the PAG's represent trigger levels calling for the initiation of recommended protective actions. Once the protective action is initiated, it should be executed so as to prevent as much of the calculated projected dose from being received as is reasonably achievable. This does not mean, however, that all doses above guidance levels can be prevented.

Further, the guides are not intended to prohibit taking actions at projected exposures lower than the PAG values. They have been derived for general cases and are just what their name implies, guides. As provided in FRC Reports No. 5 and No. 7 and as discussed in paragraph 1 of this notice, in the absence of significant constraints, responsible authority may find it appropriate to implement low-impact protective actions at projected radiation doses less than those specified in the guides. Similarly, high impact actions may be justified at higher projected doses. These judgments must be made according to the facts of each situation. Paragraphs (a) (2) and (3) have been added to the final recommendations to incorporate this concept.

9. Several comments questioned the adequacy of the level of risk judged acceptable in deriving the proposed PAG values. A comment stated that the estimated one in a million annual individual risk of death from natural disasters is extremely conservative. EPA suggested that comparative risk is appropriate for perspective but not for establishing the limits. EPA further suggested that the population-weighted average of the variability in natural background dose or the variation in dose due to the natural radioactivity in food should be the basis for judging acceptable risk.

FDA concludes that the differences between EPA's suggested approach and that employed by FDA largely involve the semantics of the rationale descriptions. As discussed in the preamble to the proposal, FDA believes that safety (or a safe level of risk) needs to be defined as the degree to which the risks are judged acceptable, because it is not possible to achieve zero risk from human endeavors. Further, ICRP (Ref. 6) recommends that, for a given application involving radiation, the net benefit to society should be positive, considering the total costs and impacts and the total benefit (this is termed "justification"). FDA believes that, to establish a PAG, the primary concern is to provide adequate protection (or safe level of risk) for members of the public. To decide on safety or levels of acceptable risk to the public from a contaminating event, FDA introduced the estimates of acceptable risk from natural disasters and background radiation. These values provided background or perspective for FDA's judgment that the proposed PAG's represent that level of food or feed radiation contamination at which protective actions should be taken to protect the public health: judgment which, consistent with FRC Report No. 5, also involves consideration of the impacts of the action and the possibility of future events. The recommendations are based on the assumption that the occurrences of environmental contamination requiring protective actions in a particular area is an unlikely event, that most individuals will never be so exposed, and that any individual is not likely to be exposed to projected doses at the PAG level more than once in his or her lifetime.

FDA continues to believe that the average risks from natural disasters and variation of background radiation provide appropriate bases for judging the acceptability of risk represented by the Preventive PAG. These recommendations incorporate the philosophy that action should be taken at the Preventive PAG level of contamination to avoid a potential public health problem. Should this action not be wholly successful, the Emergency PAG provides guidance for taking action where contaminated food is encountered. FDA expects that action at the Emergency PAG level of contamination would most likely involve food produced for consumption by the population near the source of contamination. As discussed in paragraph 2, this is also the population which might receive radiation doses from multiple pathways. Thus, the Emergency PAG might be considered to be an upper bound for limiting the total radiation dose to individuals. FDA emphasizes, however, that the Emergency PAG is not a boundary between safe levels and hazardous or injury levels of radiation. Individuals may receive an occupational dose of 5 rem each year over their working lifetime with the expectation of minimal increased risks to the individual. Persons in high elevation areas such as Colorado receive about 0.04 rem per year (or 2.8 rem in a lifetime) above the average background radiation dose for the United States population as a whole. The Emergency PAG is also consistent with the upper range of PAG's proposed by EPA for the cloud (plume) pathway (Ref. 7).

FDA agrees that a population-weighted variable is as applicable to the evaluation of comparative risks as is a geographic variable. Arguments can be made for using either variable. Because persons rather than geographic areas are the important parameter in the evaluation of risk associated with these guides, FDA has used population-weighting in estimating the variability of the annual external dose from natural radiation. A recent EPA study (Ref. 20) indicates that the average population dose from external background radiation dose is 53 millirem (mrem) per year, and the variability in lifetime dose taken as two standard deviations is about 2,000 mrem. The proposal, which indicated that the variation in external background was about 600 mrem, utilized a geographic weighting of State averages.

Radioactivity in food contributes about 20 mrem per year to average population doses and about 17 mrem per year of this dose results from potassium-40 (Ref. 8). Measurements of potassium-40 (and stable potassium) indicate that variability (two standard deviations) of the potassium-40 dose is about 28 percent or a lifetime dose of 350 mrem. It should be noted that body levels of potassium are regulated by metabolic processes and not dietary selection or residence. The variation of the internal dose is about one-fifth of the variation from external background radiation. FDA has retained the proposed preventive PAG of 500 mrem whole body even though the newer data indicate a greater variation in external background radiation.

FDA did not consider perceived risks in deriving the proposed PAG values because perceived risk presents numerous problems in its appropriateness and application. If the factor of perception is added to the equation, scientific analysis is impossible.

10. Two comments questioned the assumptions that the Emergency PAG might apply to 15 million people and that the Preventive PAG might apply to the entire United States. One comment noted that 15 million persons are more than that population currently within 25 miles of any United States reactor sites: thus, using this figure results in guides more restrictive than necessary. The other comment noted that, by reducing the population involved, and unacceptably high value could result.

The ratio of total United States population to the maximum number of people in the vicinity of an operating reactor could be erroneously interpreted so that progressively smaller populations would be subject to progressively larger individual risks. This is not the intent of the recommendations. Hence, the risk from natural disasters, the variation in the population-weighted natural background radiation dose to the total population, and the variation in dose due to ingestion of food, have been used to provide the basis for the Preventive PAG. The basis for the Emergency PAG involves considerations of (1) The ratio between average and maximum individual radiation doses (taken as 1 to 10), (2) the cost of low and high impact protective actions, (3) the relative risks from natural disasters, (4) health impact, (5) the upper range of the PAG's proposed by EPA (5 rem projected radiation dose to the whole body and 25 rem projected dose to the thyroid), and (6) radiation doses from multiple pathways.

11. A comment, citing experience with other contaminants, suggested that further consideration should be given to the problem of marketability of foods containing low levels of radioactivity.

Marketability is not a concern for PAG development. However, the publication of the PAG's should enhance marketability of foods because it will enhance public confidence in food safety. Also, FEMA has been specifically directed to undertake a public information program related to radiation emergencies to allay public fears and perceptions.

12. A comment noted the difficulty in assessing the impacts of and the benefits to be gained from protective actions. Another comment suggested that there were lower impact actions which could be implemented to keep food off the market until radiation levels in the food approach normal background.

The recommendation that planning officials consider the impacts of protective actions in implementing action does not imply that a mathematical analysis is required. Rather, FDA intends that the local situation, resources, and impacts that are important in assuring effective protective actions be considered in selecting any actions to be implemented. As discussed in paragraph 8, if the local constraints permit a low impact action, this can be appropriate at lower projected doses. Because it is not possible in general guidance to consider fully all local constraints, the PAG's represent FDA's judgment as to when protective actions are appropriate.

Agricultural and Dose Models

13. Several comments noted errors either in approach or calculations regarding the proposed agricultural and dose models, while others specifically noted that there are newer and better

models for use in computation of the derived response levels.

FDA appreciates the careful review and the suggestions as to better data and models. The references suggested, as well as other current reports, have been carefully reviewed and appropriate ones are being used as the basis for computation of the derived response levels for the final PAG's. The specific models and data being used are as follows:

Agricultural Model—UCRL-51939, 1977 (Ref. 9).

Intake per unit deposition—Table B-1, UCRL-51939 (Ref. 9).

Peak milk activity—Equation 8, UCRL-51939 (Ref. 9).

Area grazed by cow—45 square meters/day, UCRL-51939 (Ref. 9).

Initial retention on forage—0.5 fraction, UCRL-51939 (Ref. 9).

Forage yield—0.25 kilogram/square meter (dry weight), UCRL-51939 (Ref. 9).

Milk consumption—0.7 liter/day infant, ICRP-23, 1974 (Ref. 10);—0.55 liter/day adult, USDA, 1965 (Ref. 11).

Dose conversion factors (rem per microcurie ingested).

	Infant	Adult	
Iodine-131	16	1.5	Wellman and Anger, 1971 (Ref. 12).
Cesium-134	0.119	0.086	Adult—ORNL/NUREG/TM-190, 1978 (Ref. 13). Infant—Extrapolated from adult based on relative body weight 70 kilograms (kg) and 7.7 kg and effective retention, 102 days and 19.5 days, adult and infant respectively.
Cesium-137	0.071	0.061	NCRP No. 52, 1977 (Ref. 14).
Strontium-89	0.194	0.012	Adult, ICRP-30, 1978 (Ref. 15).
Strontium-90	2.40	0.70	Infant, Papworth and Vennart, 1973 (Ref. 16).

The use of the newer agricultural model (Ref. 9) has resulted in a 20 percent increase in the iodine-131 derived response levels identified in paragraph (d)(1) and (d)(2) of the recommendations. Generally, similar magnitude changes are reflected in the derived response levels for the other radionuclides. Newer data on iodine-131 dose conversion factors (Ref. 17) would have further increased the derived response levels for that radionuclide by about 40 percent, but these data have not been used pending their acceptance by United States recommending authorities. In addition, the proposal contained a systematic error in that the pasture derived response levels were stated to be based on fresh weight but were in fact based on dry weight. Fresh weight values (¼ of dry weight values) are identified in the final recommendations and are listed under "Forage Concentration".

Other Comments

14. A comment addressed the definition of the critical or sensitive population for the tables in proposed § 1090.400(d) and observed that there is a greater risk per rem to the younger age groups than to adults. Another comment requested further explanation of the relative ability to protect children and adults.

FDA agrees that, ideally, the critical segment of the population should be defined in terms of the greatest risk per unit intake. However, this would introduce greater complexity into the recommendations than is justified, because the risk estimates are uncertain. The final recommendations provide derived response levels for infants at the Preventive PAG and infants and adults for the Emergency PAG.

FDA has reexamined the available data and concludes that taking action at the Preventive PAG (based on the infant as the critical or sensitive population) will also provide protection of the fetus from the mother's ingestion of milk. The definition of newborn infant in the tables in paragraph (d) of the PAG's has been revised to reflect this conclusion.

15. EPA commented that its regulations governing drinking water (40 CFR Subchapter D) permit blending of water to meet maximum contaminant levels. EPA suggested that FDA's short-term recommendations should be compatible with the long-term EPA regulations.

As stated in paragraphs 1 and 2 of this notice, FDA's recommendations apply to human food and animal feed, whereas EPA is responsible for providing guidance on contaminated water. Also, as discussed in paragraph 3 of the proposal, there is a long-standing FDA policy that blending of food is unlawful under the Federal Food, Drug, and Cosmetic Act. Further, these guides are intended for protective actions under emergency situations and are not for continuous exposure applications. For these reasons, FDA concludes that the differences between its recommendations and EPA's regulations are appropriate.

16. Two comments were received on the adequacy or availability of resources for sampling and analysis of State, local, and Federal agencies and the adequacy of guidance on sampling procedures.

These recommendations are not designed to provide a compendium of sampling techniques, methods, or resources. The Department of Energy through its Interagency Radiological Assistance Plan (IRAP) coordinates the provision of Federal assistance and an Offsite Instrumentation Task Force of the Federal Radiological Preparedness Coordinating Committee administered by FEMA is developing specific guidance on instrumentation and methods for sampling food (Ref. 21).

Cost Analysis

17. Several comments argued that FDA's cost/benefit analysis used to establish the PAG levels was inadequate. Comments stated that it is not appropriate to assign a unique fixed dollar value to the adverse health effects associated with one person-rem of dose.

FDA advises that its cost/benefit analysis was not conducted to establish the PAG levels. FDA considers such use inappropriate in part because of the inability to assess definitively the total societal impacts (positive and negative) of such actions. Rather, the cost/benefit analysis was used to determine whether protective actions at the recommended PAG's would provide a net societal benefit. To make such an assessment, it is necessary to place a dollar value on a person-rem of dose.

18. Several comments also questioned the appropriateness of the assumption in the cost/benefit analysis of 23 days of protective action, the need to address radionuclides other than iodine-131, and the need to consider the impact of other protective actions.

The cost assessments have been extensively revised to consider all the radionuclides for which derived response levels are provided in the recommendations and to incorporate updated cost data and risk estimates (Ref. 22). The cost/benefit analysis is limited to the condemnation of milk and the use of stored feed because accident analyses indicate that the milk pathway is the most likely to require protective action. Further, these two actions are the most likely protective actions that will be implemented.

FDA approached the cost/benefit analysis by calculating the concentration of radioactivity in milk at which the cost of taking action equals the risk avoided by the action taken on a daily milk intake basis. The assessment was done on a population basis and considered only the direct costs of the protective actions. The analysis indicates that, for restricting feed to stored feed, the cost-equals-benefit concentrations are about one-fiftieth to one-eightieth of the Preventive PAG level (derived peak milk concentration) for iodine-131, cesium-134, and cesium-137 and about one-third of the level for strontium-89 and strontium-90. For condemnation of milk, based on value at the farm, the cost-equals-benefit concentrations are similar fractions of the Emergency PAG levels (derived peak milk concentration). If condemnation of milk is based on retail market value, the cost-equals-benefit concentrations are greater by a factor of two. Thus, it appears that protective actions at the Preventive or Emergency PAG levels will yield a net societal benefit. However, in the case of strontium-89 and strontium-90, protective action will yield a benefit only for concentrations greater than about one-third the derived peak values. In the case of iodine-131, cesium-134, and cesium-137, protective actions could be continued to avoid 95 percent of the projected radiation dose for initial peak concentrations at the PAG level.

References

The following information has been placed on display in the Dockets Management Branch (HFA-305), Food and Drug Administration, Rm. 4-62, 5600 Fishers Lane, Rockville, MD 20857, and may be seen between 9 a.m. and 4 p.m., Monday through Friday.

1. Federal Radiation Council. Memorandum for the President. "Radiation Protection Guidance for Federal Agencies." Federal Register, August 22, 1964 (29 FR 12056), and Report No. 5 (July 1964).

2. Federal Radiation Council. Memorandum for the President. "Radiation Protection Guidance for Federal Agencies." Federal Register," May 22, 1965 (30 FR 6953), and Report No. 7 (May 1965).

3. National Academy of Sciences/National Research Council. "The Effects on Population of Exposure to Low Levels of Ionizing Radiation." Report of the Advisory Committee on Biological Effects of Ionizing Radiation (BEIR-III) (1980).

4. United States Nuclear Regulatory Commission. Reactor Safety Study. WASH-1400. Appendix VI (October 1975).

5. Ron, E. and B. Modan. "Benign and Malignant Thyroid Neoplasms After Childhood Irradiation for Tinea Capitis." *Journal of the National Cancer Institute*, Vol. 65, No. 1 (July 1980).

6. International Commission on Radiological Protection (ICRP). Recommendations of the International Commission on Radiological Protection. ICRP Publication 26, Annals of the ICRP, Pergamon Press (1977).

7. Environmental Protection Agency. "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents." EPA 520/1-75-001, revised June 1980.

8. United Nations Scientific Committee on the Effects of Atomic Radiation. 1977 Report. United Nations, New York (1977).

9. Ng, Y. C., C. S. Colsher, D. J. Quinn, and S. E. Thompson. "Transfer Coefficients for the Prediction of the Dose to Man Via the Forage-Cow-Milk Pathway from Radionuclides Released to the Biosphere." UCRL-51939, Lawrence Livermore Laboratory (July 15, 1977)

10. International Commission on Radiological Protection. Report of a Task Group of Committee 2 on Reference Man. Publication 23, p. 360 Pergamon Press. Oxford (1974).

11. U.S. Department of Agriculture. "Household Food Consumption Survey 1965-1966."

12. Wellman, H. N. and R. T. Anger. "Radioiodine Dosimetry and the Use of Radioiodines Other Than ^{131}I in Thyroid Diagnosis." *Seminars in Nuclear Medicine*, 3:356 (1971).

13. Killough, G. G., D. E. Dunning, S. R. Bernard, and J. C. Pleasant. "Estimates of Internal Dose Equivalent to 22 Target Organs for Radionuclides Occurring in Routine Releases from Nuclear Fuel-Cycle Facilities, Vol. 1." ORNL/NUREG/TM-190. Oak Ridge National Laboratory (June 1978).

14. National Council on Radiation Protection and Measurements. "Cesium-137 From the Environment to Man: Metabolism and Dose." NCRP Report No. 52, Washington (January 15, 1977).

15. International Commission on Radiological Protection. Limits for Intakes of Radionuclides by Workers. ICRP Publication 30, Part 1, Annals of the ICRP, Pergamon Press (1979).

16. Papworth, D. G., and J. Vennart. "Retention of ^{90}Sr in Human Bone at Different Ages and Resulting Radiation Doses." *Physics in Medicine and Biology*, 18:169-186 (1973).

17. Kereiakes, J. G., P. A. Feller, F. A. Ascoli, S. R. Thomas, M. J. Gelfand, and E. L. Saenger. "Pediatric Radiopharmaceutical Dosimetry" *in* "Radiopharmaceutical Dosimetry Symposium," April 26-29, 1976. HEW Publication (FDA) 76-8044 (June 1976).

18. National Academy of Sciences/National Research Council. "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation." Report of the Advisory Committee on Biological Effects of Ionizing Radiation (BEIR-I) (1972).

19. National Council on Radiation Protection and Measurements (NCRP). "Basic Radiation Protection Criteria." NCRP Report No. 39, Washington (1971).

20. Bogen, K. T., and A. S. Goldin. "Population Exposure to External Natural Radiation Background in the United States." ORP/SEPD-80-12. Environmental Protection Agency, Washington, DC (April 1981).

21. Federal Interagency Task Force on Offsite Emergency Instrumentation for Nuclear Accidents. "Guidance on Offsite Emergency Radiation Measurement Systems: Phase 2, Monitoring and Measurement of Radionuclides to Determine Dose Commitment in the Milk Pathway," developed by Exxon Nuclear Idaho Co. Inc., Idaho Falls, ID. Draft, July 1981 (to be published by FEMA).

22. Shleien, B., G. D. Schmidt, and R. P. Chiacchierini. "Background for Protective Action Recommendations: Accidental Radioactive Contamination of Food and Animal Feeds." September 1981. Department of Health and Human Services, Food and Drug Administration, Bureau of Radiological Health, Rockville, MD.

Pertinent background data and information on the recommendations are on file in the Dockets Management Branch, and copies are available from that office (address above).

Based upon review of the comments received on the proposal of December 15, 1978 (43 FR 58790), and FDA's further consideration of the need to provide guidance to State and local agencies for use in emergency response planning in the event that an incident results in the radioactive contamination of human food or animal feed, the agency offers the following recommendations regarding protective action planning for human food and animal feeds:

Accidental Radioactive Contamination of Human Food and Animal Feeds; Recommendations for State and Local Agencies

(a) *Applicability.* (1) These recommendations are for use by appropriate State or local agencies in response planning and the conduct of radiation protection activities involving the production, processing, distribution, and use of human food and animal feeds in the event of an incident resulting in the lease of radioactivity to the environment. The Food and Drug Administration (FDA) recommends that this guidance be used on a case-by-case basis to determine the need for taking appropriate protective action in the event of a diversity of contaminating events, such as nuclear facility accidents, transportation accidents, and fallout from nuclear devices.

(2) Protective actions are appropriate when the health benefits associated with the reduction in exposure to be achieved are sufficient to offset the undesirable features of the protective actions. The Protective Action Guides (PAG's) in paragraph (c) of these recommendations represent FDA's judgment as to the level of food contamination resulting from radiation incidents at which protective action should be taken to protect the public health. Further, as provided by Federal guidance issued by the Federal Radiation Council, if, in a particular situation, and effective action with low total impact is available, initiation of such action at a projected dose lower than the PAG may be justifiable. If only very high-impact action would be effective, initiation of such action at a projected dose higher than the PAG may be justifiable. (See 29 FR 12056; August 22, 1964.) A basic assumption in the development of protective action guidance is that a condition requiring protective action is unusual and should not be expected to occur frequently.

Circumstances that involve repetitive occurrence, a substantial probability of recurrence within a period of 1 or 2 years, or exposure from multiple sources (such as airborne cloud and food pathway) would require special consideration. In such a case, the total projected dose from the several events and the total impact of the protective actions that might be taken to avoid the future dose from one or more of these events may need to be considered. In any event, the numerical values selected for the PAG's are not intended to authorize deliberate releases expected to result in absorbed doses of these magnitudes.

(3) A protective action is an action or measure taken to avoid most of the radiation dose that would occur from future ingestion of foods contaminated with radioactive materials. These recommendations are intended for implementation within hours or days from the time an emergency is recognized. The action recommended to be taken should be continued for a sufficient time to avoid most of the projected dose. Evaluation of when to cease a protective action should be made on a case-by-case basis considering the specific incident and the food supply contaminated. In the case of the pasture/cow/milk/person pathway, for which derived "response levels" are provided in paragraph (d) of these recommendations, it is expected that actions would not need to extend beyond 1 or 2 months due to the reduction of forage concentrations by weathering (14-day half-life assumed). In the case of fresh produce directly contaminated by deposition from the cloud, actions would be necessary at the time of harvest. This guidance is not intended to apply to the problems of long-term food pathway contamination where adequate time after the incident is available to evaluate the public health consequences of food contamination using current recommendations and the guidance in Federal Radiation Council (FRC) Report No. 5, July 1964 and Report No. 7, May 1965.

(b) *Definitions.* (1) "Dose" is a general term denoting the quantity of radiation or energy absorbed. For special purposes it must be appropriately qualified. In these recommendations it refers specifically to the term "dose equivalent."

(2) "Dose commitment" means the radiation dose equivalent received by an exposed individual to the organ cited over a lifetime from a single event.

(3) "Dose equivalent" is a quantity that expresses all radiation on a common scale for calculating the effective absorbed dose. It is defined as the product of the absorbed dose in rads and certain modifying factors. The unit of dose equivalent is the rem.

(4) "Projected dose commitment" means the dose commitment that would be received in the future by individuals in the population group from the contaminating event if no protective action were taken.

(5) "Protective action" means an action taken to avoid most of the exposure to radiation that would occur from future ingestion of foods contaminated with radioactive materials.

(6) "Protective action guide (PAG)" means the projected dose commitment values to individuals in the general population that warrant protective action following a release of radioactive material. Protective action would be warranted if the expected individual dose reduction is not offset by negative social, economic, or health effects. The PAG does not include the dose that has unavoidably occurred before the assessment.

(7) "Preventive PAG" is the projected dose commitment value at which responsible officials should take protective actions having minimal ipact to prevent or reduce the radioactive contamination of human food or animal feeds.

(8) "Emergency PAG" is the projected dose commitment value at which responsible officials should isolate food containing radioactivity to prevent its introduction into commerce and at which the responsible officials should determine whether condemnation or another disposition is appropriate. At the Emergency PAG, higher impact actions are justified because of the projected health hazards.

(9) "Rad" means the unit of absorbed dose equal to 0.01 Joule per kilogram in any medium.

(10) "Rem" is a special unit of dose equivalent. The dose equivalent in rems is numerically equal to the absorbed dose in rads multiplied by the quality factor, the distribution factor, and any other necessary modifying factors.

(11) "Response level" means the activity of a specific radionuclide (i) initially deposited on pasture; or (ii) per unit weight or volume of food or animal feed; or (iii) in the total dietary intake which corresponds to a particular PAG.

(c) *Protective action guides (PAG's).* To permit flexibility of action for the reduction of radiation exposure to the public via the food pathway due to the occurrence of a contaminating event, the following Preventive and Emergency PAG's for an exposed individual in the population are adopted:

(1) *Preventive PAG* which is (i) 1.5 rem projected dose commitment to the thyroid, or (ii) 0.5 rem projected dose commitment to the whole body, bone marrow, or any other organ.

(2) *Emergency PAG* which is (i) 15 rem projected dose commitment to the thyroid, or (ii) 5 rem projected dose commitment to the whole body, bone marrow, or any other organ.

(d) *Response levels equivalent to PAG.* Although the basic PAG recommendations are given in terms of projected dose equivalent, it is often more convenient to utilize specific radionuclide concentrations upon which to initiate protective action. Derived response levels equivalent to the PAG's for radionuclides of interest are:

(1) *Response level for Preventive PAG.* Infant [1] as critical segment of population.

[1] Newborn infant includes fetus (pregnant women) as critical segment of population for iodine-131. For other radionuclides, "infant" refers to child less than 1 year of age.

Response levels for preventive PAG	^{131}I [2]	^{134}Cs [4]	^{137}Cs [4]	^{90}Sr	^{89}Sr
Initial Activity Area Deposition (microcuries/square meter)	0.13	2	3	0.5	8
Forage Concentration [3] (microcuries/kilogram)	0.05	0.8	1.3	0.18	3
Peak Milk Activity (microcuries/liter)	0.015	0.15	0.24	0.008	0.14
Total intake (microcuries)	0.09	4	7	0.2	2.8

[2] From fallout, iodine-131 is the only radioiodine of significance with respect to milk contamination beyond the first day. In case of a reactor accident, the cumulative intake of iodine-133 via milk is about 2 percent of iodine-131 assuming equivalent deposition.

[3] Fresh weight.

[4] Intake of cesium via the meat/person pathway for adults may exceed that of the milk pathway; therefore, such levels in milk should cause surveillance and protective actions for meat as appropriate. If both cesium-134 and cesium-137 are equally present as might be expected for reactor accidents, the response levels should be reduced by a factor of two.

(2) *Response level for Emergency PAG.* The response levels equivalent to the Emergency PAG are presented for both infants and adults to permit use of either level and thus assure a flexible approach to taking action in cases where exposure of the most critical portion of the population (infants and pregnant women) can be prevented:

Response levels for emergency PAG	131I[3]		134Cs[5]		137Cs[5]		90Sr		89Sr	
	Infant[1]	Adult	Infant[2]	Adult	Infant[2]	Adult	Infant[2]	Adult	Infant[2]	Adult
Initial Activity Area Deposition (microcuries/square meter)	1.3	18	20	40	30	50	5	20	80	1600
Forage Concentration [4] (microcuries/kilogram)	0.5	7	8	17	13	19	1.8	8	30	700
Peak Milk Activity (microcuries/liter)	0.15	2	1.5	3	2.4	4	0.08	0.4	1.4	30
Total intake (microcuries)	0.9	10	40	70	70	80	2	7	26	400

[1] Newborn infant includes fetus (pregnant women) as critical segment of population for iodine-131.

[2] "Infant" refers to child less than 1 year of age.

[3] From fallout, iodine-131 is the only radioiodine of significance with respect to milk contamination beyond the first day. In case of a reactor accident the cumulative intake of iodine-133 via milk is about 2 percent of iodine-131 assuming equivalent deposition.

[4] Fresh weight.

[5] Intake of cesium via the meat/person pathway for adults may exceed that of the milk pathway, therefore, such levels in milk should cause surveillance and protective actions for meat as appropriate. If both cesium-134 and cesium-137 are equally present, as might be expected for reactor accidents, the response levels should be reduced by a factor of 2.

(e) *Implementation.* When using the PAG's and associated response levels for response planning or protective actions, the following conditions should be followed:

(1) *Specific food items.* To obtain the response level (microcurie/kilogram) equivalent to the PAG for other specific foods, it is necessary to weigh the contribution of the individual food to the total dietary intake; thus,

$$\text{Response Level} = \frac{\text{Total Intake (microcuries)}}{\text{Consumption (kilograms)}}$$

Where: Total intake (microcuries) for the appropriate PAG and radionuclide is given in paragraph (d) of these recommendations
and
Consumption is the product of the average daily consumption specified in paragraph (e)(1)(i) of these recommendations and the days of intake of the contaminated food as specified in paragraph (e)(1)(ii) of these recommendations.

(i) The daily consumption of specific foods in kilograms per day for the general population is given in the following table:

Food	Average consumption for the general population (kilogram/day)
Milk, cream, cheese, ice cream [1]	.570
Fats, oils	.055
Flour, cereal	091
Bakery products	.150
Meat	.220
Poultry	055
Fish and shellfish	.023
Eggs	.055
Sugar, syrups, honey, molasses, etc.	.073
Potatoes, sweet potatoes	105
Vegetables, fresh (excluding potatoes)	.145
Vegetables, canned, frozen, dried	.077
Vegetables, juice (single strength)	008
Fruit, fresh	165
Fruit, canned, frozen, dried	036
Fruit, juice (single strength)	045
Other beverages (soft drinks, coffee, alcoholic)	180
Soup and gravies (mostly condensed)	.036
Nuts and peanut butter	.009
Total	2 099

[1] Expressed as calcium equivalent; that is, the quantity of whole fluid milk to which dairy products are equivalent in calcium content.

(ii) Assessment of the effective days of intake should consider the specific food, the population involved, the food distribution system, and the radionuclide. Whether the food is distributed to the retail market or produced for home use will significantly affect the intake in most instances. Thus, while assessment of intake should be on a case-by-case basis, some general comments may be useful in specific circumstances.

(*a*) For short half-life radionuclides, radioactive decay will limit the ingestion of radioactive materials and the effective "days of intake". The effective "days of intake" in this case is 1.44 times the radiological half-life. For iodine-131 (half-life—8.05 days), the effective "days of intake" is, thus, 11 days.

(*b*) Where the food product is being harvested on a daily basis, it may be reasonable to assume reduction of contamination due to weathering. As an initial assessment, it may be appropriate to assume a 14-day weathering half-life (used for forage in pasture/cow/milk pathway) pending further evaluation. In this case, the effective "days of intake" is 20 days. A combination of radioactive decay and weathering would result in an effective half-life for iodine-131 of 5 days and reduce the "days of intake" to 7 days.

(*c*) In the case of a food which is sold in the retail market, the effective "days of intake" would probably be limited by the quantity purchased at a given time. For most food, especially fresh produce, this would probably be about a 1 week supply. In some cases, however, larger quantities would be purchased for home canning or freezing. For most foods and members of the public, an effective "days of intake" 30 days is probably conservative.

(iii) For population groups having significantly different dietary intakes, an appropriate adjustment of dietary factors should be made.

(2) *Radionuclide mixtures.* If a mixture of radionuclides is present, the sum of all the ratios of the concentration of each specific radionulide to its specific response level equivalent to the PAG should be less than one.

(3) *Other radionuclides.* The response level for the Preventive and Emergency PAG for other radionuclides should be calculated from dose commitment factors available in the literature (Killough, G. G., et al., ORNL/NUREG/TM-190 (1978) (adult only), and U.S. Nuclear Regulatory Commission Reg. Guide 1.109 (1977)).

(4) *Other critical organs.* Dose commitment factors in U.S. Nuclear Regulatory Commission Reg. Guide 1.109 (1977) refer to bone rather than bone marrow dose commitments. For the purpose of these recommendations, dose commitment to the bone marrow is considered to be 0.3 of the bone dose commitment. This is based on the ratio of dose rate per unit activity in the bone marrow to dose rate per unit activity in a small tissue-filled cavity in bone and assumes that strontium-90 is distributed only in the mineral bone (Spiers, F. W., et al. *in* "Biomedical Implications of Radiostrontium Exposure," AEC Symposium 25 (1972)). The ratio for strontium-89 is the same because the mean particle energies are similar (0.56 MeV (megaelectronvolts)). Situations could arise in which an organ other than those discussed in this paragraph coul

be considered to be the organ receiving the highest dose per unit intake. In the case of exposure via the food chain, depending on the radionuclide under consideration, the gastrointestinal tract could be the primary organ exposed. The references cited in paragraph (e)(3) of these recommendations contain dose commitment factors for the following organs: bone, kidneys, liver, ovaries, spleen, whole body, and gastrointestinal tract.

(5) Prompt notification of State and local agencies regarding the occurrence of an incident having potential public health consequences is of significant value in the implementation of effective protective actions. Such notification is particularly important for protective actions to prevent exposures from the airborne cloud but is also of value for food pathway contamination. Accordingly, this protective action guidance should be incorporated in State/local emergency plans which provide for coordination with nuclear facility operators including prompt notification of accidents and technical communication regarding public health consequences and protective action.

(f) *Sampling parameter.* Generally, sites for sample collection should be the retail market, the processing plant, and the farm. Sample collection at the milk processing plant may be more effcient in determining the extent of the food pathway contamination. The geographic area where protective actions are implemented should be based on considerations of the wind direction and atmospheric transport, measurements by airborne and ground survey teams of the radioactive cloud and surface deposition, and measurements in the food pathway.

(g) *Recommended methods of analysis.* Techniques for measurement of radionuclide concentrations should have detection limits equal to or less than the response levels equivalent to specific PAG. Some useful methods of radionuclide analysis can be found in:

(1) *Laboratory Methods*—"HASL Procedure Manual," edited by John H. Harley, HASL 300 ERDA, Health and Safety Laboratory, New York, NY, 1973; "Rapid Methods for Estimating Fission Product Concentrations in Milk," U.S. Department of Health, Education, and Welfare, Public Health Service Publication No. 999-R-2, May 1963; "Evaluation of Ion Exchange Cartridges for Field Sampling of Iodine-131 in Milk," Johnson, R. H. and T. C. Reavy, *Nature*, 208, (5012): 750-752, November 20, 1965; and

(2) *Field Methods*—Kearny, C. H., ORNL 4900, November 1973; Distenfeld, C. and J. Klemish, Brookhaven National Laboratory, NUREG/CR-0315, December 1978; and International Atomic Energy Agency, "Environmental Monitoring in Emergency Situations," 1966. Analysis need not be limited to these methodologies but should provide comparable results. Action should not be taken without verification of the analysis. Such verification might include the analysis of duplicate samples, laboratory measurements, sample analysis by other agencies, sample analysis of various environmental media, and descriptive data on radioactive release.

(h) *Protective actions.* Actions are appropriate when the health benefit associated with the reduction in dose that can be achieved is considered to offset the undesirable health, economic, and social factors. It is the intent of these recommendations that, not only the protective actions cited for the Emergency PAG be initiated when the equivalent response levels are reached, but also that actions appropriate at the Preventive PAG be considered. This has the effect of reducing the period of time required during which the protective action with the greater economic and social impact needs to be taken. FDA recommends that once one or more protective actions are initiated, the action or actions continue for a sufficient time to avoid most of the projected dose. There is a longstanding FDA policy that the purposeful blending of adulterated food with unadulterated food is a violation of the Federal Food, Drug, and Cosmetic Act. The following protective actions should be considered for implementation when the projected dose equals or exceeds the appropriate PAG:

(1) *Preventive PAG.* (i) For pasture: (*a*) Removal of lactating dairy cows from contaminated pasturage and substitution of uncontaminated stored feed.

(*b*) Substitute source of uncontaminated water.

(ii) For milk: (*a*) Withholding of contaminated milk from the market to allow radioactive decay of short-lived radionuclides. This may be achieved by storage of frozen fresh milk, frozen concentrated milk, or frozen concentrated milk products.

(*b*) Storage for prolonged times at reduced temperatures also is feasible provided ultrahigh temperature pasteurization techniques are employed for processing (Finley, R. D., H. B. Warren, and R. E. Hargrove, "Storage Stability of Commercial Milk," *Journal of Milk and Food Technology*, 31(12):382-387, December 1968).

(*c*) Diversion of fluid milk for production of dry whole milk, nonfat dry milk, butter, cheese, or evaporated milk.

(iii) For fruits and vegetables: (*a*) Washing, brushing, scrubbing, or peeling to remove surface contamination.

(*b*) Preservation by canning, freezing, and dehydration or storage to permit radioactive decay of short-lived radionuclides.

(iv) For grains: (*a*) Milling and (*b*) polishing.

(v) For other food products, processing to remove surface contamination.

(vi) For meat and meat products, intake of cesium-134 and cesium-137 by an adult via the meat pathway may exceed that of the milk pathway; therefore, levels of cesium in milk approaching the "response level" should cause surveillance and protective actions for meat as appropriate.

(vii) For animal feeds other than pasture, action should be on a case-by-case basis taking into consideration the relationship between the radionuclide concentration in the animal feed and the concentration of the radionuclide in human food. For hay and silage fed to lactating cows, the concentration should not exceed that equivalent to the recommendations for pasture.

(2) *Emergency PAG.* Responsible officials should isolate food containing radioactivity to prevent its introduction into commerce and determine whether condemnation or another disposition is appropriate. Before taking this action, the following factors should be considered:

(i) The availability of other possible protective actions discussed in paragraph (h)(1) of these recommendations.

(ii) Relative proportion of the total diet by weight represented by the item in question.

(iii) The importance of the particular food in nutrition and the availability of uncontaminated food or substitutes having the same nutritional properties.

(iv) The relative contribution of other foods and other radionuclides to the total projected dose.

(v) The time and effort required to effect corrective action.

This notice is issued under the Public Health Service Act (secs. 301, 310, 311, 58 Stat. 691-693 as amended, 88 Stat. 371 (42 U.S.C. 241, 242o, 243)) and under authority delegated to the Commissioner of Food and Drugs (21 CFR 5.10).

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Arthur Hull Hayes, Jr.,

Commissioner of Food and Drugs.

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CHAPTER 4

Protective Action Guides for the Intermediate Phase (Deposited Radioactive Materials)

4.1 Introduction

Following a nuclear incident it may be necessary to temporarily relocate the public from areas where extensive deposition of radioactive materials has occurred until decontamination has taken place. This chapter identifies the levels of radiation exposure which indicate when relocation from contaminated property is warranted.

The period addressed by this chapter is denoted the "intermediate phase." This is arbitrarily defined as the period beginning after the source and releases have been brought under control and environmental measurements are available for use as a basis for decisions on protective actions and extending until these protective actions are terminated. This phase may overlap the early and late phases and may last from weeks to many months. For the purpose of dose projection, it is assumed to last for one year. Prior to this period protective actions will have been taken based upon the PAGs for the early phase. It is assumed that decisions will be made during the intermediate phase concerning whether particular areas or properties from which persons have been relocated will be decontaminated and reoccupied, or condemned and the occupants permanently relocated. These actions will be carried out during the late or "recovery" phase.

Although these Protective Action Guides (PAGs) were developed based on expected releases of radioactive materials characteristic of reactor incidents, they may be applied to any type of incident that can result in long-term exposure of the public to deposited radioactivity.

PAGs are expressed in terms of the projected doses above which specified protective actions are warranted. In the case of deposited radioactivity, the major relevant protective action is relocation. Persons not relocated (i.e., those in less contaminated areas) may reduce their dose through the application of simple decontamination techniques and by spending more time than usual in low exposure rate areas (e.g., indoors).

The PAGs should be considered mandatory only for use in planning, e.g., in developing radiological emergency response plans. During an incident, because of unanticipated local conditions and constraints, professional judgment by responsible officials will be required in their application. Situations can be envisaged, where contamination from a nuclear incident occurs at a site or time in which relocation of the public, based on the recommended PAGs, would be impracticable. Conversely, under some conditions, relocation may be quite practicable at projected doses below the PAGs. These situations require judgments by those responsible for protective action decisions at the time of the incident. A discussion of the implementation of these PAGs is provided in Chapter 7.

The PAGs for relocation specified in this chapter refer only to estimates of doses due to exposure during the first year after the incident. Exposure pathways include external exposure to radiation from deposited radioactivity and inhalation of resuspended radioactive materials. Protective Action Guides for ingestion exposure pathways, which also apply during the intermediate phase, are discussed separately in Chapter 3.

Individuals who live in areas contaminated by long-lived radionuclides may be exposed to radiation from these materials, at a decreasing rate, over the entire time that they live in the area. This would be the case for those who are not relocated as well as for persons who return following relocation. Because it is usually not practicable, at the time of a decision to relocate, to calculate the doses that might be incurred from exposure beyond one year, and because different protective actions may be appropriate over such longer periods of time, these doses are not included in the dose specified in the PAGs for relocation.

4.1.1 Exposure Pathways

The principal pathways for exposure of the public occupying locations contaminated by deposited radioactivity are expected to be exposure of the whole body to external gamma radiation from deposited radioactive materials (groundshine) and internal exposure from the inhalation of resuspended materials. For reactor incidents, external gamma radiation is expected to be the dominant source.

Almost invariably relocation decisions will be based on doses from the above pathways. (However, in rare cases where food or drinking water is contaminated to levels above the PAG for ingestion, and its withdrawal from use will create a risk from starvation greater than that from the radiation dose, the dose from ingestion should be added to the dose from the above pathways.) PAGs related specifically to the withdrawal of contaminated food and water from use are discussed in Chapter 3.

Other potentially significant exposure pathways include exposure to beta radiation from surface contamination and direct ingestion of contaminated soil. These pathways are not expected to be controlling for reactor incidents (AR-89).

4.1.2 The Population Affected

The PAGs for relocation are intended for use in establishing the boundary of a restricted zone within an area that has been subjected to deposition of radioactive materials. During their development, consideration was given to the higher risk of effects on health to children and fetuses from radiation dose and the higher risk to some other population groups from relocation. To avoid the complexity of implementing separate PAGs for individual members of the population, the relocation PAG is established at a level that will provide adequate protection for the general population.

Persons residing in contaminated areas outside the restricted zone will be at some risk from radiation dose. Therefore, guidance on the reduction of dose during the first year to residents outside this zone is also provided. Due to the high cost of relocation, it is more practical to reduce dose in this population group by the early application of simple, low-impact, protective actions other than by relocation.

4.2 The Protective Action Guides for Deposited Radioactivity

PAGs for protection from deposited radioactivity during the intermediate phase are summarized in Table 4-1. The basis for these values is presented in detail in Appendix E. In summary, relocation is warranted when the projected sum of the dose equivalent from external gamma radiation and the committed effective dose equivalent from inhalation of resuspended radionuclides exceeds 2 rem in the first year. Relocation to avoid exposure of the skin to beta radiation is warranted at 50 times the numerical value of the relocation PAG for effective dose equivalent.

Persons who are not relocated, i.e., those in areas that receive relatively small amounts of deposited radioactive material, should reduce their exposure by the application of other measures. Possible dose reduction techniques range from the simple processes of scrubbing and/or flushing surfaces, soaking or plowing of soil, removal and disposal of small spots of soil found to be highly contaminated (e.g., from settlement of water), and spending more time than usual in lower exposure rate areas (e.g., indoors), to the difficult and time-consuming processes of removal, disposal, and replacement of contaminated surfaces. It is anticipated that simple processes will be most appropriate for early application. Many can be carried out by residents themselves with support from response officials for assessment of the levels of contamination, guidance on appropriate actions, and disposal of contaminated materials. Due to the relatively low cost and risk associated with these protective actions, they may be justified as ALARA measures at low dose levels. It is, however, recommended that response officials concentrate their initial efforts in areas where the projected dose from the first year of exposure exceeds 0.5 rem. In addition, first priority should be given to cleanup of residences of pregnant women who may exceed this criterion.

Table 4-1 Protective Action Guides for Exposure to Deposited Radioactivity During the Intermediate Phase of a Nuclear Incident

Protective Action	PAG (projected dose)[a]	Comments
Relocate the general population.[b]	≥2 rem	Beta dose to skin may be up to 50 times higher
Apply simple dose reduction techniques.[c]	<2 rem	These protective actions should be taken to reduce doses to as low as practicable levels.

[a]The projected sum of effective dose equivalent from external gamma radiation and committed effective dose equivalent from inhalation of resuspended materials, from exposure or intake during the first year. Projected dose refers to the dose that would be received in the absence of shielding from structures or the application of dose reduction techniques. These PAGs may not provide adequate protection from some long-lived radionuclides (see Section 4.2.1).

[b]Persons previously evacuated from areas outside the relocation zone defined by this PAG may return to occupy their residences. Cases involving relocation of persons at high risk from such action (e.g., patients under intensive care) should be evaluated individually.

[c]Simple dose reduction techniques include scrubbing and/or flushing hard surfaces, soaking or plowing soil, minor removal of soil from spots where radioactive materials have concentrated, and spending more time than usual indoors or in other low exposure rate areas.

4.2.1 Longer Term Objectives of the Protective Action Guides

It is an objective of these PAGs to assure that 1) doses in any single year after the first will not exceed 0.5 rem, and 2) the cumulative dose over 50 years (including the first and second years) will not exceed 5 rem. For source terms from reactor incidents, the above PAG of 2 rem projected dose in the first year is expected to meet both of those objectives through radioactive decay, weathering, and normal part time occupancy in structures. Decontamination of areas outside the restricted area may be required during the first year to meet these objectives for releases consisting primarily of long-lived radionuclides. For situations where it is impractical to meet these objectives though decontamination, consideration should be given to relocation at a lower projected first year dose than that specified by the relocation PAG.

After the population has been protected in accordance with the PAGs for relocation, return for occupancy of previously restricted areas should be governed on the basis of Recovery Criteria as presented in Chapter 8.

Projected dose considers exposure rate reduction from radioactive decay and, generally, weathering. When one also considers the anticipated effects of shielding from partial occupancy in homes and other structures, persons who are not relocated should receive a dose substantially less than the projected dose. For commonly assumed reactor source terms, we estimate that 2 rem projected dose in the first year will be reduced to about 1.2 rem by this factor. The application of simple decontamination techniques shortly after the incident can be assumed to provide a further 30 percent or more reduction, so that the maximum first year dose to persons who are not relocated is expected to be less than one rem. Taking account of decay rates assumed to be associated with releases from nuclear power plant incidents (SN-82) and shielding from partial occupancy and weathering, a projected dose of 2 rem in the first year is likely to amount to an actual dose of 0.5 rem or less in the second year and 5 rem or less in 50 years. The application of simple dose reduction techniques would reduce these doses further. Results of calculations supporting these projections are summarized in Table E-6 of Appendix E.

4.2.2 Applying the Protective Action Guides for Relocation

Establishing the boundary of a restricted zone may result in three different types of actions:

1. Persons who, based on the PAGs for the early phase of a nuclear incident (Chapter 2), have already been evacuated from an area which is now designated as a restricted zone must be converted to relocation status.

2. Persons not previously evacuated who reside inside the restricted zone should relocate.

3. Persons who normally reside outside the restricted zone, but were previously evacuated, may return. A gradual return is recommended, as discussed in Chapter 7.

Small adjustments to the boundary of the restricted zone from that given by the PAG may be justified on the basis of difficulty or ease of implementation. For example, the use of a convenient natural boundary could be cause for adjustment of the restricted zone. However, such decisions should be supported by demonstration that exposure rates to persons not relocated can be promptly reduced by methods other than relocation to meet the PAG, as well as the longer term dose objectives addressed in Section 4.2.1.

Reactor incidents involving releases of major portions of the core inventory under adverse atmospheric conditions can be postulated for which large areas would have to be restricted under these PAGs. As the affected land area increases, they will become more difficult and costly to implement, especially in densely populated areas. For situations where implementation becomes impracticable or impossible (e.g., a large city), informed judgment must be exercised to assure priority of protection for individuals in areas having the highest exposure rates. In such situations, the first priority for any area should be to reduce dose to pregnant women.

4.3 Exposure Limits for Persons Reentering the Restricted Zone

Individuals who are permitted to reenter a restricted zone to work, or for other justified reasons, will require protection from radiation. Such individuals should enter the restricted zone under controlled conditions in accordance with dose limitations and other procedures for control of occupationally-exposed workers (EP-87). Ongoing doses received by these individuals from living in a contaminated area outside the restricted zone need not be included as part of this dose limitation applicable to workers. In addition, dose received previously from the plume and associated groundshine, during the early phase of the nuclear incident, need not be considered.

References

AR-89 Aaberg, Rosanne, Evaluation of Skin and Ingestion Exposure Pathways. EPA 520/1-89-016. U.S. Environmental Protection Agency, Washington, (1989).

EP-87 U.S. Environmental Protection Agency. Radiation Protection Guidance to Federal Agencies for Occupational Exposure. Federal Register, 52, 2822; January 27, 1987.

SN-82 Sandia National Laboratory. Technical Guidance for Siting Criteria Development. NUREG/CR-2239. U.S. Nuclear Regulatory Commission, Washington, (1982).

CHAPTER 5

Implementing the Protective Action Guides for the Early Phase

5.1 Introduction

This chapter provides general guidance for implementing the Protective Action Guides (PAGs) set forth in Chapter 2. In particular, the objective is to provide guidance for estimating projected doses from exposure to an airborne plume of radioactive material, and for choosing and implementing protective actions.

Following an incident which has the potential for an atmospheric release of radioactive material, the responsible State and/or local authorities will need to decide whether offsite protective actions are needed and, if so, where and when they should be implemented. These decisions will be based primarily on (a) the potential for releases, (b) projected doses as a function of time at various locations in the environment, and (c) dose savings and risks associated with various protective actions.

Due to the wide variety of nuclear facilities, incidents, and releases that could occur, it is not practical to provide specific implementing guidance for all situations. Examples of the types of sources leading to airborne releases that this guidance may be applied to are nuclear power reactors, uranium fuel cycle facilities, nuclear weapons facilities, radiopharmaceutical manufacturers and users, space vehicle launch and reentry, and research reactors. For many specific applications, however, it will be appropriate to develop and use implementing procedures that are designed for use on a case-by-case basis.

Dose conversion factors (DCF) and derived response levels (DRL) are provided for radionuclides that are most likely to be important in an incident involving an airborne release of radioactive materials. DCFs and DRLs for radionuclides not listed may be developed from the sources referenced in the tables. The values provided here are the best currently available. However, as new information is developed these values may change. This chapter will be revised from time to time to reflect such changes.

5.2 Initial Response and Sequence of Subsequent Actions

In the case of an atmospheric release, the protective actions which may be required are those which protect the population from inhalation of radioactive materials in the plume, from exposure to gamma radiation

4. Estimation of offsite consequences (e.g., calculation of the plume centerline dose rates and projected doses at various distances downwind from the release point).

5. Implementation of protective actions in additional areas if needed.

6. Decisions to terminate existing protective actions should include, as a minimum, consideration of the status of the plant and the PAGs for relocation (Chapter 4). (Withdrawal of protective actions from areas where they have already been implemented is usually not advisable during the early phase because of the potential for changing conditions and confusion.)

For other types of incidents the sequence of actions may vary in details, depending on the specific emergency response plan, but in general the sequence and general reporting requirements will be the same.

5.2.1 Notification

The nuclear facility operator or other designated individual should provide the first notification to State and/or local authorities that a nuclear incident has occurred. In the case of an incident with the potential for offsite consequences, notification of State and local response organizations by a facility operator should include recommendations, based on plant conditions, for early evacuation and/or sheltering in predesignated areas. Early estimates of the various components of projected doses to the population at the site boundary, as well as at more distant locations, along with estimated time frames, should be made as soon as the relevant source or release data become available. Emergency response planners should make arrangements with the facility operator to assure that this information will be made available on a timely basis and that dose projections will be provided in units that can be directly compared to the PAGs. Planners should note that the toxic chemical hazard is greater than the radiation hazard for some nuclear incidents, e.g. a uranium hexafluoride release.

For some incidents, such as re-entry of satellites or an incident in a foreign country, notification is most likely to occur through the responsible Federal agency, most commonly the Environmental Protection Agency or the National Aeronautics and Space Administration. In such cases projections of dose and recommendations to State and local officials for protective actions will be made at the Federal level, under the Federal Radiological Emergency Response Plan (FE-85).

5.2.2 Immediate Protective Action

Guidance for developing emergency response plans for implementation of immediate protective actions for incidents at commercial nuclear power plants is contained in NUREG-0654 (NR-80). Planning elements for measurements are unlikely to be available to project doses accurately. Doses must be projected using initial environmental measurements or estimates of the source term, and using atmospheric transport previously observed under similar meteorological conditions. These projections are needed to determine whether protective actions should be implemented in additional areas during the early phase.

Source term measurements, or exposure rates or concentrations measured in the plume at a few selected locations, may be used to estimate the extent of the exposed area in a variety of ways, depending on the types of data and computation methods available. The most accurate method of projecting doses is through the use of an atmospheric diffusion and transport model that has been verified for use at the site in question. A variety of computer software can be used to estimate exposures in real time, or to extrapolate a series of previously-prepared isopleths for unit releases under various meteorological conditions. The latter can be adjusted for the estimated source magnitude or environmental measurements at a few locations during the incident. If the model projections have some semblance of consistency with environmental measurements, extrapolation to other distances and areas can be made with greater confidence. If projections using a sophisticated site-specific model are not available, a simple, but crude, method is to measure the plume centerline exposure rate[2] at ground level (approximately one meter height) at a known distance downwind of the release point and then to calculate exposure rates at other downwind locations by assuming that the plume centerline exposure rate is a known function of the distance from the release point.

The following relationship can be used for this calculation:

$$D_2 = D_1 (R_1/R_2)^y ,$$

where D_1 and D_2 are measurements of exposure rates at the centerline of the plume at distances R_1 and R_2, respectively, and y is a constant that depends on atmospheric stability. For stability classes A and B, y = 2; for stability classes C and D, y = 1.5; and for stability classes E and F, y = 1. Classes A and B (unstable) occur with light winds and strong sunlight, and classes E and F (stable) with light winds at night. Classes C and D generally occur with winds stronger than about 10 mph. This method of extrapolation is risky because the measurements available at the reference distance may be unrepresentative, especially if the plume is aloft and has a looping

[2]The centerline exposure rate can be determined by traversing the plume at a point sufficiently far downwind that it has stabilized (usually more than one mile from the release point) while taking continuous exposure rate measurements.

behavior. In the case of an elevated plume, the ground level concentration increases with distance from the source, and then decreases, whereas any high energy gamma radiation from the overhead cloud continuously decreases with distance. For these reasons, this method of extrapolation will perform best for surface releases or if the point of measurement for an elevated release is sufficiently distant from the point of release for the plume to have expanded to ground level (usually more than one mile). The accuracy of this method will be improved by the use of measurements from many locations averaged over time.

5.4 Dose Projection

The PAGs set forth in Chapter 2 are specified in terms of the effective dose equivalent. This dose includes that due to external gamma exposure of the whole body, as well as the committed effective dose equivalent from inhaled radionuclides. Guidance is also provided on protective action levels for the thyroid and skin, in terms of the committed dose equivalent to these organs. Further references to effective or organ dose equivalent refer to these two quantities, respectively. Methods for estimating projected doses for each of these forms of exposure are discussed below. These require knowledge of, or assumptions for, the intensity and duration of exposure and make use of standard assumptions on the relation, for each radioisotope, between exposure and dose. Exposure and dose projections should be based on the best estimates available. The methods and models used here may be modified as necessary for specific sites to achieve improved accuracy.

5.4.1 Duration of Exposure

The projected dose for comparison to the early phase PAGs is normally calculated for exposure during the first four days following the projected (or actual) start of a release. The objective is to encompass the entire period of exposure to the plume and to deposited material prior to implementation of any further, longer-term protective action, such as relocation. Four days is chosen here as the duration of exposure to deposited materials during the early phase because, for planning purposes; it is a reasonable estimate of the time needed to make measurements, reach decisions, and prepare to implement relocation. However, officials at the site at the time of the emergency may decide that a different time is more appropriate. Corresponding changes to the dose conversion factors found in tables in Section 5.4.2 will be needed if another exposure period is selected.

Protective actions are taken to avoid or reduce projected doses. Doses incurred before the start of the protective action being considered should not normally be included in evaluating the need for protective action. Likewise, doses that may be incurred at later times than those affected by the specific protective action should not be included. For example, doses which may be incurred through ingestion pathways or long-term exposure to deposited radioactive materials take place over a different, longer time period. Protective actions for such exposures should be based on guidance addressed in other chapters.

The projected dose from each radionuclide in a plume is proportional to the time-integrated concentration of the radionuclide in the plume at each location. This concentration will depend on the rate and the duration of the release and meteorological conditions. Release rates will vary with time, and this time-dependence cannot usually be predicted accurately. In the absence of more specific information, the release rate may be assumed to be constant.

Another factor affecting the estimation of projected dose is the duration of the plume at a particular location. For purposes of calculating projected dose from most pathways, exposure will start at a particular location when the plume arrives and end when the plume is no longer present, due either to an end to the release, or a change in wind direction. Exposure from one pathway (whole body exposure to deposited materials) will continue for an extended period. Other factors such as the aerodynamic diameter and solubility of particles, shape of the plume, and terrain may also affect estimated dose, and may be considered on a site- and/or source-specific basis.

Prediction of time frames for releases is difficult because of the wide range associated with the spectrum of potential incidents. Therefore, planners should consider the possible time periods between an initiating event and arrival of a plume, and the duration of releases in relation to the time needed to implement competing protective actions (i.e., evacuation and sheltering). Analyses of nuclear power reactors (NR-75) have shown that some incidents may take several days to develop to the point of a release, while others may begin as early as one-half hour after an initiating event. Furthermore, the duration of a release may range from less than one hour to several days, with the major portion of the release usually occurring within the first day.

Radiological exposure rates are quite sensitive to the wind speed. The air concentration is inversely related to the wind speed at the point of release. Concentrations are also affected by the turbulence of the air, which tends to increase with wind speed and sunlight, and by meandering of the plume, which is greater at the lower wind speeds. This results in higher concentrations generally being associated with low winds near the source, and with moderate winds at larger distances. Higher windspeed also shortens the travel time. Planning information on time frames for releases from nuclear power facilities may be found in Reference NR-78. Time frames for releases from other facilities will depend on the characteristics of the facility.

Since a change in wind direction will also affect the duration of exposure, it is very important that arrangements be made for a public, private, or military professional weather service to provide information on current meteorological and wind conditions and predicted wind direction persistence during an incident, in addition to information received from the facility operator.

5.4.2 Dose Conversion Factors

This section provides dose conversion factors (DCFs) and derived response levels (DRLs) for those radionuclides important for responding to most types of incidents. These are supplemented by an example to demonstrate their application. The DCFs are useful where multiple radionuclides are involved, because the total dose from a single exposure pathway will be the sum of the doses calculated for each radionuclide. The DRLs are surrogates for the PAG and are directly usable for releases consisting primarily of a single nuclide, in which case the DRL can be compared directly to the measured or calculated concentration. (DRLs also can be used for multiple radionuclides by summing the ratios of the environmental concentration of each nuclide to its respective DRL. To meet the PAG, this sum must be equal to or less than unity.)

DCFs and DRLs for each of the three major exposure pathways for the early phase (external exposure to plume, plume inhalation, and external exposure from deposited materials) are provided separately in Section 5.6. They are all expressed in terms of the time-integrated air concentration at the receptor so they can be conveniently summed over the three exposure pathways to obtain composite DRLs and DCFs for each radionuclide. These composite values are tabulated in Table 5-1 for effective dose and in Table 5-2 for thyroid dose from inhalation of radioiodines.

The tabulated DCFs and DRLs include assumptions on particle size, deposition velocity, the presence of short-lived daughters, and exposure duration as noted. The existence of more accurate data for individual radionuclides may justify modification of the DCFs and DRLs. The procedures described in Section 5.6 for developing the DCFs and DRLs for individual exposure pathways may be referred to, to assist such modifications.

To apply Tables 5-1 and 5-2 to decisions on implementing PAGs, one may use either the DCFs or DRLs. DCFs are used to calculate the projected composite dose for each radionuclide; these doses are then summed and compared to the PAG. The DRLs may be used by summing the ratios of the concentration of each radionuclide to its corresponding DRL. If the sum of the ratios exceeds unity, the corresponding protective action should be initiated.

Table 5-1 Dose Conversion Factors (DCF) and Derived Response Levels (DRL) for Combined[a] Exposure Pathways During the Early Phase of a Nuclear Incident[b]

Radionuclide	DCF rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[c] $\mu Ci \cdot cm^{-3} \cdot h$
H-3	7.7E+01	1.3E-02
C-14	2.5E+03	4.0E-04
Na-22	1.9E+04	5.3E-05
Na-24	7.3E+03	1.4E-04
P-32	1.9E+04	5.4E-05
P-33	2.8E+03	3.6E-04
S-35	3.0E+03	3.4E-04
Cl-36	2.6E+04	3.8E-05
K-40	1.6E+04	6.5E-05
K-42	2.0E+03	5.1E-04
Ca-45	8.0E+03	1.3E-04
Sc-46	4.4E+04	2.3E-05
Ti-44	1.2E+06	8.2E-07
V-48	2.4E+04	4.2E-05
Cr-51	5.5E+02	1.8E-03
Mn-54	1.2E+04	8.5E-05
Mn-56	1.8E+03	5.7E-04
Fe-55	3.2E+03	3.1E-04
Fe-59	2.3E+04	4.4E-05
Co-58	1.7E+04	5.7E-05
Co-60	2.7E+05	3.7E-06
Ni-63	7.6E+03	1.3E-04
Cu-64	5.9E+02	1.7E-03
Zn-65	2.7E+04	3.7E-05
Ge-68	6.2E+04	1.6E-05
Se-75	1.2E+04	8.3E-05
Kr-85	1.3E+00	7.8E-01
Kr-85m	9.3E+01	1.1E-02
Kr-87	5.1E+02	2.0E-03
Kr-88	1.3E+03	7.8E-04

Table 5-1, Continued

Radionuclide	DCF rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[c] $\mu Ci \cdot cm^{-3} \cdot h$
Kr-89	1.2E+03	8.6E-04
Rb-86	8.3E+03	1.2E-04
Rb-88	5.2E+02	1.9E-03
Rb-89	1.4E+03	7.3E-04
Sr-89	5.0E+04	2.0E-05
Sr-90	1.6E+06	6.4E-07
Sr-91	2.4E+03	4.2E-04
Y-90	1.0E+04	9.9E-05
Y-91	5.9E+04	1.7E-05
Zr-93	3.9E+05	2.6E-06
Zr-95	3.2E+04	3.2E-05
Zr-97	5.5E+03	1.8E-04
Nb-94	5.0E+05	2.0E-06
Nb-95	1.0E+04	9.7E-05
Mo-99	5.2E+03	1.9E-04
Tc-99	1.0E+04	1.0E-04
Tc-99m	1.7E+02	6.0E-03
Ru-103	1.3E+04	7.7E-05
Ru-105	1.2E+03	8.2E-04
Ru/Rh-106[d]	5.7E+05	1.7E-06
Pd-109	1.3E+03	7.6E-04
Ag-110m	9.8E+04	1.0E-05
Cd-109	1.4E+05	7.3E-06
Cd-113m	1.8E+06	5.5E-07
In-114m	1.1E+05	9.4E-06
Sn-113	1.3E+04	7.8E-05
Sn-123	3.9E+04	2.6E-05
Sn-125	2.0E+04	5.1E-05
Sn-126	1.2E+05	8.4E-06
Sb-124	3.8E+04	2.6E-05

Table 5-1, Continued

Radionuclide	DCF rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[c] $\mu Ci \cdot cm^{-3} \cdot h$
Sb-126	2.6E+04	3.9E-05
Sb-127	9.5E+03	1.1E-04
Sb-129	2.0E+03	5.0E-04
Te-127m	2.6E+04	3.9E-05
Te-129	1.4E+02	7.0E-03
Te-129m	2.9E+04	3.5E-05
Te-131m	8.6E+03	1.2E-04
Te-132	1.2E+04	8.5E-05
Te/I-132[d]	2.0E+04	5.0E-05
Te-134	7.0E+02	1.4E-03
I-125	3.0E+04	3.3E-05
I-129	2.1E+05	4.8E-06
I-131	5.3E+04	1.9E-05
I-132[e]	4.9E+03	2.0E-04
I-133	1.5E+04	6.8E-05
I-134	3.1E+03	3.3E-04
I-135	8.1E+03	1.2E-04
Xe-131m	4.9E+00	2.0E-01
Xe-133	2.0E+01	5.0E-02
Xe-133m	1.7E+01	5.9E-02
Xe-135	1.4E+02	7.0E-03
Xe-135m	2.5E+02	4.1E-03
Xe-137	1.1E+02	9.3E-03
Xe-138	7.2E+02	1.4E-03
Cs-134	6.3E+04	1.6E-05
Cs-136	1.8E+04	5.6E-05
Cs/Ba-137[d]	4.1E+04	2.4E-05
Cs-138	1.6E+03	6.1E-04
Ba-133	1.1E+04	8.9E-05
Ba-139	2.3E+02	4.4E-03

Table 5-1, Continued

Radionuclide	DCF rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[c] $\mu Ci \cdot cm^{-3} \cdot h$
Ba-140	5.3E+03	1.9E-04
La-140	1.1E+04	8.8E-05
La-141	7.3E+02	1.4E-03
La-142	2.3E+03	4.3E-04
Ce-141	1.1E+04	9.0E-05
Ce-143	4.7E+03	2.1E-04
Ce-144	4.5E+05	2.2E-06
Ce/Pr-144[d]	4.5E+05	2.2E+06
Nd-147	8.8E+03	1.1E-04
Pm-145	3.7E+04	2.7E-05
Pm-147	4.7E+04	2.1E-05
Pm-149	3.6E+03	2.8E-04
Pm-151	2.8E+03	3.5E-04
Sm-151	3.6E+04	2.8E-05
Eu-152	2.7E+05	3.8E-06
Eu-154	3.5E+05	2.9E-06
Eu-155	5.0E+04	2.0E-05
Gd-153	2.9E+04	3.4E-05
Tb-160	3.5E+04	2.9E-05
Ho-166m	9.4E+05	1.1E-06
Tm-170	3.2E+04	3.2E-05
Yb-169	1.1E+04	8.9E-05
Hf-181	2.1E+04	4.8E-05
Ta-182	6.0E+04	1.7E-05
W-187	1.7E+03	6.0E-04
Ir-192	3.8E+04	2.7E-05
Au-198	5.2E+03	1.9E-04
Hg-203	9.9E+03	1.0E-04
Tl-204	2.9E+03	3.5E-04
Pb-210	1.6E+07	6.1E-08

Table 5-1, Continued

Radionuclide	DCF rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[c] $\mu Ci \cdot cm^{-3} \cdot h$
Bi-207	3.1E+04	3.2E-05
Bi-210	1.9E+04	5.3E-05
Po-210	1.1E+07	8.9E-08
Ra-226	1.0E+07	9.7E-08
Ac-227	8.0E+09	1.2E-10
Ac-228	3.7E+05	2.7E-06
Th-227	1.9E+07	5.2E-08
Th-228	4.1E+08	2.4E-09
Th-230	3.9E+08	2.6E-09
Th-232	2.0E+09	5.1E-10
Pa-231	1.5E+09	6.5E-10
U-232	7.9E+08	1.3E-09
U-233	1.6E+08	6.2E-09
U-234	1.6E+08	6.3E-09
U-235	1.5E+08	6.8E-09
U-236	1.5E+08	6.6E-09
U-238	1.4E+08	7.0E-09
U-240	2.7E+03	3.7E-04
Np-237	6.5E+08	1.5E-09
Np-239	3.6E+03	2.8E-04
Pu-236	1.7E+08	5.8E-09
Pu-238	4.7E+08	2.1E-09
Pu-239	5.2E+08	1.9E-09
Pu-240	5.2E+08	1.9E-09
Pu-241	9.9E+06	1.0E-07
Pu-242	4.9E+08	2.0E-09
Am-241	5.3E+08	1.9E-09
Am-242m	5.1E+08	2.0E-09
Am-243	5.3E+08	1.9E-09
Cm-242	2.1E+07	4.8E-08

Table 5-1, Continued

Radionuclide	DCF rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[c] $\mu Ci \cdot cm^{-3} \cdot h$
Cm-243	3.7E+08	2.7E-09
Cm-244	3.0E+08	3.4E-09
Cm-245	5.5E+08	1.8E-09
Cm-246	5.4E+08	1.9E-09
Cf-252	1.9E+08	5.3E-09

[a]Sum of doses from external exposure and inhalation from the plume, and external exposure from deposition. "Dose" means the sum of effective dose equivalent from external radiation and committed effective dose equivalent from intake.

[b]See footnote a to Table 5-4 for assumptions on inhalation and footnote b to Table 5-5 for assumptions on deposition velocity. The quantity $\mu Ci \cdot cm^{-3} \cdot h$ refers to the time-integrated air concentration at one meter height.

[c]For 1 rem committed effective dose equivalent.

[d]The contribution from the short-lived daughter is included in the factors for the parent radionuclide.

[e]These factors should only be used in situations where I-132 appears without the parent radionuclide.

Persons exposed to an airborne particulate plume will receive dose to skin from beta emitters in the plume as well as from those deposited on skin and clothing. Although it is possible to detect beta radiation, it is not practical, for purposes of decisions on evacuation and sheltering, to determine dose to skin by field measurement of the beta dose equivalent rate near the skin surface. Such doses are determined more practically through calculations based on time-integrated air concentration, an assumed deposition velocity, and an assumed time period between deposition and skin decontamination. For the purpose of evaluating the relative importance of skin dose compared to the dose from external gamma exposure and inhalation, dose conversion factors were evaluated using a deposition velocity of 1 cm/sec and an exposure time before decontamination of 12 hours. Using these conservative assumptions, it was determined that skin beta dose should seldom, if ever, be a controlling pathway during the early phase. Therefore, no DCFs or DRLs are listed for skin beta dose.

Table 5-2 Dose Conversion Factors (DCF) and Derived Response Levels (DRL) Corresponding to a 5 rem Dose Equivalent to the Thyroid from Inhalation of Radioiodine

Radionuclide	DCF rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[a] $\mu Ci \cdot cm^{-3} \cdot h$
Te/I-132[b]	2.9E+05	1.8E-05
I-125	9.6E+05	5.2E-06
I-129	6.9E+06	7.2E-07
I-131	1.3E+06	3.9E-06
I-132	7.7E+03	6.5E-04
I-133	2.2E+05	2.3E-05
I-134	1.3E+03	3.9E-03
I-135	3.8E+04	1.3E-04

[a]For a 5 rem committed dose equivalent to the thyroid.

[b]*The contribution from the short-lived daughter is included in the factors for the parent radionuclide.*

Because of large uncertainties in the assumptions for deposition, air concentrations are an inadequate basis for decisions on the need to decontaminate individuals. Field measurements should be used for this (See Chapter 7, Section 7.6.3.). It should be noted that, even in situations where the skin beta dose might exceed 50 rem, evacuation would not usually be the appropriate protective action, because skin decontamination and clothing changes are easily available and effective. However, evacuation would usually already be justified in these situations due to dose from inhalation during plume passage.

The following example demonstrates the use of the data in Tables 5-1 and 5-2 for a simple analysis involving three radionuclides.

Based on source term and meteorological considerations, it is assumed that the worst probable nuclear incident at an industrial facility is a fire that could disperse radioactive material into the atmosphere, yielding a time-integrated concentration of radionuclides at a nearby populated area, as follows:

Radionuclide	$\mu Ci \cdot cm^{-3} \cdot h$
Zr-95	2E-6
Cs-134	4E-8
I-131	1.2E-5

We examine whether evacuation is warranted at these levels, based on PAGs of 1 rem for effective dose and 5 rem for dose to the thyroid. We use the DCFs in Table 5-1 for effective dose and Table 5-2 for thyroid dose from inhalation of radioiodines to calculate the relevant doses, H, as follows:

$$H = \sum_{i}^{n} DCF_i \times C_i$$

where DCF_i = dose conversion factor for radionuclide i,

C_i = time-integrated concentration of radionuclide i,

and n = the number of radionuclides present.

For the committed effective dose equivalent (see Table 5-1):

(2 E-6 x 3.2E+4)+(4E-8 x 6.3 E+4) +(1.2E-5 x 5.3E+4) = 0.71 rem.

For the committed dose equivalent to the thyroid (see Table 5-2):

1.2E-5 x 1.3E+6 = 16 rem.

The results of these calculations show that, at the location for which these time-integrated concentrations are specified, the committed dose equivalent to the thyroid from inhalation would be over three times the PAG for dose to thyroid, thus justifying evacuation. Using meteorological dilution factors, one could calculate the additional distance to which evacuation would be justified to avoid exceeding the PAG for thyroid dose.

To use the DRLs from Table 5-1 and 5-2, find the sum,

$$\sum_{i}^{n} \frac{C_i}{DRL_i}$$

for both effective dose and thyroid dose, where DRL_i is the derived response level for radionuclide i, and C_i is defined above. If the sum in either case is equal to or greater than unity, evacuation of the general population is warranted.

For effective dose (see Table 5-1):

$$\frac{2E{-}6}{3.2E{-}5} + \frac{4E{-}8}{1.6E{-}5} + \frac{1.2E{-}5}{1.9E{-}5} = 0.7$$

For dose to the thyroid (see Table 5-2):

$$\frac{1.2E{-}5}{3.9E{-}6} = 3$$

It is apparent that these calculations yield the same conclusions as those using the DCFs.

5.4.3 Comparison with Previously-Recommended PAGs

Many emergency response plans have already been developed using previously-recommended PAGs that apply to the dose equivalent to the whole body from direct (gamma) radiation from the plume and to the thyroid from inhalation of radioiodines. For nuclear power plant incidents, the former PAG for whole body exposure provides public health protection comparable to that provided by the new PAG expressed in terms of effective dose equivalent. This is demonstrated in Table C-9 (Appendix C), which shows comparative doses for nuclear power plant fuel-melt accident sequences having a wide range of magnitudes. The PAG for the thyroid is unchanged. On the other hand, application of these PAGs to alpha emitting radionuclides leads to quite different derived response levels from those based on earlier health physics considerations, because of new dose conversion factors and the weighting factors assigned to the exposed organs (EP-88).

5.5 Protective Actions

This section provides guidance for implementing the principal protective actions (evacuation and sheltering) for protection against the various exposure pathways resulting from an airborne plume. Sheltering means the use of the closest available structure which will provide protection from exposure to an airborne plume, and evacuation means the movement of individuals away from the path of the plume.

Evacuation and sheltering provide different levels of dose reduction for the principal exposure pathways (inhalation of radioactive material, and direct gamma exposure from the plume or from material deposited on surfaces). The effectiveness of evacuation will depend on many factors, such as how rapidly it can be implemented and the nature of the accident. For accidents where the principal source of dose is inhalation, evacuation could increase exposure if it is implemented during the passage of a short-term plume, since moving vehicles provide little protection against exposure (DO-90). However, studies (NR-89a) continue to show that, for virtually all severe reactor accident scenarios, evacuation during plume passage does not increase the risk of acute health effects above the risk while sheltering. Sheltering, which in most cases can be almost immediately implemented, varies in usefulness depending upon the type of release, the shelter available, the duration of the plume passage, and climatic conditions.

Studies have been conducted to evaluate shelter (EP-78a) and evacuation (HA-75) as protective actions for incidents at nuclear power facilities. Reference EP-78b suggests one method for evaluating and comparing the benefits of these two actions. This requires collecting planning information before and data following an incident, and using calculations and graphical means to evaluate whether evacuation, sheltering, or a combination of sheltering followed by evacuation should be recommended at different locations. Because of the many interacting variables, the user is forced to choose between making decisions during the planning phase, based on assumed data that may be grossly inaccurate, or using a time-consuming more comprehensive process after the incident when data may be available. In the former situation, the decision may not have a sound basis, whereas in the latter, the decision may come too late to be useful.

The recommended approach is to use planning information for making early decisions. The planned response should then be modified following the incident only if timely detailed information is available to support such modifications.

The planner should first compile the necessary information about the emergency planning zone (EPZ) around the facility. For the case of power reactors, some of this information is described in NUREG-0654 (NR-80). It should include identifying the population distribution, the sheltering effectiveness of residences and other structures, institutions containing population groups that require special consideration, evacuation routes, logical boundaries for evacuation zones, transportation systems, communications systems, and special problem areas. In addition, the planner should identify the information that may be available following an incident, such as environmental monitoring data, meteorological conditions, and plant conditions. The planner should identify key data or information that would justify specific protective actions. The evaluation and planning should also include the selection of institutions where persons should be provided with stable iodine for thyroid protection in situations where radioiodine inhalation is projected.

The following sections discuss key factors which affect the choice between evacuation and sheltering.

5.5.1 Evacuation

The primary objective of evacuation is to avoid exposure to airborne or deposited radioactive material by moving individuals away from the path of the plume. Evacuation, if completed before plume arrival, can be 100 percent effective in avoiding future exposure. Even if evacuation coincides with or follows plume passage, a large reduction of exposure may be possible. In any case, the maximum dose avoided by evacuation will be the dose not avoidable by sheltering.

Some general conclusions regarding evacuation (HA-75) which may be useful for planning purposes are summarized below:

1. Advanced planning is essential to identify potential problems that may occur in an evacuation.

2. Most evacuees use their own personal transportation.

3. Most evacuees assume the responsibility of acquiring food and shelter for themselves.

4. Evacuation costs are highly location-dependent and usually will not

be a deterrent to carrying out an evacuation.

5, Neither panic nor hysteria has been observed when evacuation of large areas is managed by public officials.

6. Large or small population groups can be evacuated effectively with minimal risk of injury or death.

7. The risk of injury or death to individual evacuees from transportation does not change as a function of the number of persons evacuated, and can be conservatively estimated using National Highway Safety Council statistics for motor vehicle accidents (subjective information suggests that the risks will be lower).

Evacuation of the elderly, the handicapped, and inhabitants of medical and other institutions may present special problems. When sheltering can provide adequate protection, this will often be the protective action of choice. However, if the general public is evacuated and those in institutions are sheltered, there is a risk that attendants at these institutions may leave and make later evacuation of institutionalized persons difficult because of a lack of attendants. Conversely, if evacuation of institutions is attempted during evacuation of the public, traffic conditions may cause unacceptable delays. If evacuation of institutions is attempted before evacuating the public, increased risk to the public from a delayed evacuation could occur, unless the incident is very slow in developing to the point of an atmospheric release. Because of the above difficulties, medical and other institutions located within the EPZ should be evaluated to determine whether there are any logical categories of persons that should be evacuated after the public (or, when time permits, before).

5.5.2 Sheltering

Sheltering refers here to the use of readily available nearby structures for protection against exposure to an airborne plume.

Sheltering may be an appropriate protective action because:

1. It positions the public to receive additional instructions when the possibility of high enough doses to justify evacuation exists, but is small.

2. It may provide protection equal to or greater than evacuation.

3. It is less expensive and disruptive than evacuation.

4. Since it may be implemented rapidly, sheltering may be the protective action of choice if rapid evacuation is impeded by, a) severe environmental conditions--e.g. severe weather or floods; b) health constraints--e.g. patients and workers in hospitals and nursing homes; or c) long mobilization times--certain industrial and farm workers, or prisoners and guards; d) physical

constraints to evacuation--e.g. inadequate roads.

5. Sheltering may be more effective against inhalation of radioactive particulates than against external gamma exposure, especially for short-term plumes.

The use of large structures, such as shopping centers, schools, churches, and commercial buildings, as collection points during evacuation mobilization will generally provide greater protection against gamma radiation than use of small structures.

As with evacuation, delay in taking shelter during plume passage will reduce the protection from exposure to radiation. The degree of protection provided by structures is governed by attenuation of gamma radiation by structural components (the mass of walls, ceilings, etc.) and by outside/inside air-exchange rates.

If external dose from the plume or from deposited materials is the controlling criterion, shelter construction and shelter size are the most important considerations; ventilation control and filtering are less important. Although sheltering will reduce the gamma exposure rate from deposited materials, it is not a suitable protective action for this pathway for long duration exposure. The main factors which reduce whole body exposure are:

1. Wall materials and thickness and size of structure,

2. Number of stories overhead, and

3. Use of a central location within the structure.

If a major release of radioiodine or respirable particulate materials occurs, inhalation dose will be the controlling pathway. For releases consisting primarily of noble gases, external gamma exposure will be most important. However, when inhalation is the primary exposure pathway, consideration should be given to the following:

1. Ventilation control is essential for effective sheltering.

2. Dose reduction factors for sheltering can be improved in several ways for the inhalation pathway, including reducing air exchange rates by sealing cracks and openings with cloth or weather stripping, tape, etc. Although the risk to health from the action could be a constraint (particularly for infants and the infirm), using wet towels or handkerchiefs as a mask to filter the inhaled air will reduce dose from inhalation.

3. Following plume passage, people should open shelters to reduce airborne activity trapped inside, and they should leave high exposure areas as soon as possible after cloud passage to avoid exposure to deposited radioactive material.

4. Consideration should be given to the prophylactic administration of potassium iodide (KI) as a

thyroid-blocking agent to workers performing emergency services and other groups in accordance with the PAGs in Table 2-1 and the provisions in reference FD-82.[3]

5.5.3 General Guidance for Evacuation and Sheltering

The process of evaluating, recommending, and implementing evacuation or shelter for the public is far from an exact science, particularly in view of time constraints that prevent thorough analysis at the time of an incident. Their effectiveness, however, can be improved considerably by planning and testing. Early decisions should be based on information collected from the emergency planning zone during the planning phase and on information regarding conditions at the nuclear facility at the time of the incident. Best estimates of dose projections should be used for decisions between evacuation and sheltering.

The following is a summary of planning guidance for evacuation and sheltering, based on the information in Sections 5.5.1 and 5.5.2.

1. For severe incidents, where PAGs may be significantly exceeded, evacuation may be the only effective protective action close to the facility.

2. Evacuation will provide total protection from any airborne release if it is completed before arrival of the plume.

3. Evacuation may increase exposure if carried out during the plume passage, for accidents involving inhalation dose as a major contributor.

4. Evacuation is also appropriate for protection from groundshine in areas with high exposure rates from deposited materials.

5. Sheltering may be appropriate (when available) for areas not designated for immediate evacuation because:

a. It positions the public to receive additional instructions; and

b. It may provide protection equal to or greater than evacuation.

6. Sheltering is usually not appropriate where high doses are projected or for exposure lasting longer than two complete air exchanges of the shelter.

7. Because sheltering may be implemented in less time than evacuation, it may be the temporary protective action of choice if rapid evacuation is impeded by a) certain environmental conditions--e.g. severe weather or floods; b) health constraints--e.g. patients and workers

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[3]Each State has the responsibility for formulating guidance to define when (and if) the public should be given potassium iodide. Planning for its use is discussed in "Potassium Iodide as a Thyroid-blocking Agent in a Radiation Emergency: Final Recommendations on Use" (FD-82).

in hospitals and nursing homes; or c) long mobilization times--e.g. certain industrial and farm workers, or prisoners and guards; d) physical constraints to evacuation--e.g. inadequate roads.

8. If a major release of radioiodine or particulate materials occurs, inhalation dose may be the controlling criterion for protective actions. In this case:

a. Breathing air filtered through common household items (e.g., folded wet handkerchiefs or towels) may be of significant help, if appropriate precautions are taken to avoid possible suffocation.

b. After confirmation that the plume has passed, shelters should be opened to avoid airborne activity trapped inside, and persons should leave high exposure areas as soon as possible after cloud passage to avoid exposure to deposited radioactive material.

c. Consideration should be given to the prophylactic administration of potassium iodide (KI) as a thyroid-blocking agent to emergency workers, workers in critical industries, or others in accordance with the PAGs in Table 2-1 and reference FD-82.

9. If dose from external gamma radiation is the controlling criterion, shelter construction and size are the most important considerations; ventilation control and filtering are less important. The main factors which reduce whole body external dose are; a) wall thickness and size of structure, b) number of stories overhead, c) central location within the structure, and d) the height of the cloud with respect to the building.

5.6 Procedures for Calculating Dose Conversion Factors

This section provides information used in the development of the DCFs in Tables 5-1 and 5-2. Three exposure pathways are included: whole body exposure to gamma radiation from the plume, inhalation from the plume, and whole body exposure to gamma radiation from deposited materials. Although exposure of the skin from beta radiation could be significant, evaluations show that other exposure pathways will be controlling for evacuation and sheltering decisions. Therefore, DCFs for skin are not provided. Individual DCFs for the three exposure pathways are provided in the following sections. They are each expressed in terms of the time-integrated air concentration so that they may be combined to yield a composite DCF for each radionuclide that reflects all three pathways. These data may be used to facilitate revising the DCFs in Tables 5-1 and 5-2 when more specific or technically improved assumptions are available, as well as to evaluate the relative importance of the individual pathways for specific radionuclide mixes.

5.6.1 External Exposure to Gamma Radiation from the Plume

Table 5-3 provides DCFs and DRLs for external exposure to gamma radiation due to immersion in contaminated air. The values for gamma radiation will provide conservative estimates for exposure to an overhead plume. They are derived under the assumption that the plume is correctly approximated by a semi-infinite source.

The DCFs given in Table 5-3 are used to calculate the effective dose equivalent from external exposure to gamma radiation from the plume. They are based on dose-rate conversion factors for effective dose in Table A.1 of reference DO-88. The units given in Table A.1 are converted to those in Table 5-3 as follows:

$$\frac{mrem \cdot y^{-1}}{\mu Ci \cdot m^{-3}} \times 0.1142 = \frac{rem}{\mu Ci \cdot cm^{-3} \cdot h}$$

Only the short-lived daughters of Ru-106 and Cs-137 emit gamma radiation and, therefore, the DCFs from Table A.1 for these entries are attributable to their daughters. The DCF for Ce-144 is combined with that for its short-lived daughter; it is assumed they are in equilibrium. Since the DRLs apply to a PAG of 1 rem, they are simply the reciprocals of the DCFs.

5.6.2 Inhalation from the Plume

Table 5-4 provides DCFs and DRLs for committed effective dose equivalent due to inhalation of an airborne plume of radioactive particulate materials and for committed dose equivalent to the thyroid due to inhalation of radioiodines. It is assumed that the radionuclides are in the chemical and physical form that yields the highest dose, and that the particle size is one micrometer mean aerodynamic diameter. For other chemical and physical forms of practical interest the doses may differ, but in general only by a small factor. If the chemical and/or physical form (e.g. solubility class or particle size) is known or can be predicted, the DCFs for inhalation should be adjusted as appropriate.

The dose factors and breathing rate used to develop the DCFs in Table 5-4 are those given in Table 2.1 of Federal Guidance Report No.11 and were derived for "standard man" (EP-88). Although the DCFs for some radionuclides would be slightly higher for children, the conservatism in the PAGs and procedures for their application provide an adequate margin for safety. The advantage of using a single source of current data for the development and timely revision of DCFs for these and any other relevant radionuclides is also a consideration in the selection of this data base for use in emergency response applications.

The units given in Table 2-1 of EP-88 are converted to the units in Table 5-4, using a breathing rate of $1.2E+6\ cm^3 \cdot h^{-1}$, by the factor

$$Sv \cdot Bq^{-1} \cdot 4.4E+12 = \text{rem per } \mu Ci \cdot cm^{-3} \cdot h.$$

The DRLs are simply the reciprocal of the DCF.

5.6.3 External Dose from Deposited Materials

Table 5-5 provides DCFs and DRLs for 4-day exposure to gamma radiation from selected radionuclides following deposition of particulate materials on the ground from a plume. The deposition velocity (assumed to be 1 cm/s for iodines and 0.1 cm/s for other particulate materials) could vary widely depending on the physical and chemical characteristics of the deposited material and the surface, and meteorological conditions. In the case of precipitation, the amount of deposition (and thus the dose conversion factors for this exposure pathway) will be much higher. To account for the ingrowth of short-lived daughters in deposited materials after measurements are made, the tabulated values include their contribution to dose over the assumed 4-day period of exposure. Because the deposition velocity can be much lower or higher than assumed in developing the dose conversion factors for deposited materials, decision makers are cautioned to pay particular attention to actual measurements of gamma exposure from deposited materials for evacuation decisions after plume passage.

The objective is to calculate DCFs for single radionuclides in terms of effective dose equivalent from 4 days exposure to gamma radiation from deposited radioactive materials. In order to be able to sum the dose conversion factors with those for other exposure pathways, the DCF is expressed in terms of dose per unit time-integrated air concentration, where the deposition from the plume is assumed to occur at approximately the beginning of the incident. The following equation was used to generate Table 5-5:

$$DCF = V_g \cdot DCRF \cdot 1.14E\text{-}3\left[\frac{1-e^{-\lambda t}}{\lambda}\right]$$

Where:

DCF = the dose per unit air concentration ($\mu Ci \cdot cm^{-3} \cdot h$)

V_g = the deposition velocity, assumed to be 3600 $cm \cdot h^{-1}$ for iodines and 360 $cm \cdot h^{-1}$ for other particulate materials

$DRCF$ = the dose rate conversion factor ($mrem \cdot y^{-1}$ per $\mu Ci \cdot m^{-2}$) (DO-88)

1.14E-3 = a factor converting $mrem \cdot y^{-1}$ per m^2 to $rem \cdot h^{-1}$ per cm^2

λ = the decay constant for the radionuclide (h^{-1})

t = duration of exposure (hours),assumed to be 96 hours (4 days)

Table 5-3 Dose Conversion Factors (DCF) and Derived Response Levels (DRL) for External Exposure Due to Immersion in Contaminated Air

Radionuclide	DCF[a] rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[b] $\mu Ci \cdot cm^{-3} \cdot h$
H-3	0.0E+00	0.0E+00
C-14	0.0E+00	0.0E+00
Na-22	1.3E+03	7.8E-04
Na-24	2.7E+03	3.7E-04
P-32	0.0E+00	0.0E+00
P-33	0.0E+00	0.0E+00
S-35	0.0E+00	0.0E+00
Cl-36	4.8E-06	2.1E+05
K-40	9.2E+01	1.1E-02
K-42	1.7E+02	6.0E-03
Ca-45	9.3E-09	1.1E+08
Sc-46	1.2E+03	8.4E-04
Ti-44	7.7E+01	1.3E-02
V-48	1.7E+03	5.8E-04
Cr-51	1.8E+01	5.6E-02
Mn-54	5.0E+02	2.0E-03
Mn-56	1.1E+03	9.4E-04
Fe-55	1.3E-02	7.6E+01
Fe-59	7.0E+02	1.4E-03
Co-58	5.8E+02	1.7E-03
Co-60	1.5E+03	6.7E-04
Ni-63	0.0E+00	0.0E+00
Cu-64	1.1E+02	9.2E-03
Zn-65	3.4E+02	2.9E-03
Ge-68	5.2E-02	1.9E+01
Se-75	2.3E+02	4.4E-03
Kr-85	1.3E+00	7.8E-01
Kr-85m	9.3E+01	1.1E-02
Kr-87	5.1E+02	2.0E-03
Kr-88	1.3E+03	7.8E-04

Table 5-3, Continued

Radionuclide	DCF[a] rem per µCi · cm^{-3} · h	DRL[b] µCi · cm^{-3} · h
Kr-89	1.2E+03	8.6E-04
Rb-86	5.6E+01	1.8E-02
Rb-88	4.1E+02	2.5E-03
Rb-89	1.3E+03	7.7E-04
Sr-89	8.2E-02	1.2E+01
Sr-90	0.0E+00	0.0E+00
Sr-91	4.1E+02	2.4E-03
Y-90	0.0E+00	0.0E+00
Y-91	2.1E+00	4.7E-01
Zr-93	0.0E+00	0.0E+00
Zr-95	4.3E+02	2.3E-03
Zr-97	1.1E+02	9.3E-03
Nb-94	9.3E+02	1.1E-03
Nb-95	4.5E+02	2.2E-03
Mo-99	9.1E+01	1.1E-02
Tc-99	3.0E-04	3.3E+03
Tc-99m	7.6E+01	1.3E-02
Ru-103	2.8E+02	3.6E-03
Ru-105	4.6E+02	2.2E-03
Ru/Rh-106[c]	1.2E+02	8.4E-03
Pd-109	3.9E-01	2.5E+00
Ag-110m	1.6E+03	6.2E-04
Cd-109	1.3E+00	8.0E-01
Cd-113m	0.0E+00	0.0E+00
In-114m	5.2E+01	1.9E-02
Sn-113	4.8E+00	2.1E-01
Sn-123	4.1E+00	2.4E-01
Sn-125	1.8E+02	5.4E-03
Sn-126	2.8E+01	3.6E-02
Sb-124	1.1E+03	8.8E-04

Table 5-3, Continued

Radionuclide	DCF[a] rem per µCi · cm^{-3} · h	DRL[b] µCi · cm^{-3} · h
Sb-126	1.6E+03	6.2E-04
Sb-127	3.9E+02	2.6E-03
Sb-129	8.6E+02	1.2E-03
Te-127m	1.8E+00	5.6E-01
Te-129	3.1E+01	3.2E-02
Te-129m	2.0E+01	5.1E-02
Te-131m	8.5E+02	1.2E-03
Te-132	1.2E+02	8.0E-03
Te-134	5.1E+02	2.0E-03
I-125	6.3E+00	1.6E-01
I-129	4.8E+00	2.1E-01
I-131	2.2E+02	4.6E-03
I-132	1.4E+03	7.4E-04
I-133	3.5E+02	2.9E-03
I-134	1.6E+03	6.4E-04
I-135	9.5E+02	1.1E-03
Xe-131m	4.9E+00	2.0E-01
Xe-133	2.0E+01	5.0E-02
Xe-133m	1.7E+01	5.9E-02
Xe-135	1.4E+02	7.0E-03
Xe-135m	2.5E+02	4.1E-03
Xe-137	1.1E+02	9.2E-03
Xe-138	7.1E+02	1.4E-03
Cs-134	9.1E+02	1.1E-03
Cs-136	1.3E+03	7.8E-04
Cs/Ba-137[c]	3.5E+02	2.9E-03
Cs-138	1.4E+03	6.9E-04
Ba-133	2.1E+02	4.8E-03
Ba-139	2.1E+01	4.9E-02
Ba-140	1.1E+02	9.3E-03

Table 5-3, Continued

Radionuclide	DCF[a] rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[b] $\mu Ci \cdot cm^{-3} \cdot h$
La-140	1.4E+03	7.1E-04
La-141	2.5E+01	3.9E-02
La-142	1.8E+03	5.6E-04
Ce-141	4.4E+01	2.3E-02
Ce-143	1.5E+02	6.6E-03
Ce-144	1.0E+01	9.7E-02
Ce/Pr-144[c]	3.1E+01	3.2E-02
Nd-147	7.6E+01	1.3E-02
Pm-145	9.5E+00	1.0E-01
Pm-147	2.1E-03	4.8E+02
Pm-149	6.7E+00	1.5E-01
Pm-151	1.9E+02	5.2E-03
Sm-151	5.2E-04	1.9E+03
Eu-152	6.7E+02	1.5E-03
Eu-154	7.4E+02	1.3E-03
Eu-155	3.3E+01	3.1E-02
Gd-153	5.1E+01	2.0E-02
Tb-160	6.4E+02	1.6E-03
Ho-166m	9.4E+02	1.1E-03
Tm-170	2.7E+00	3.8E-01
Yb-169	1.6E+02	6.1E-03
Hf-181	3.1E+02	3.2E-03
Ta-182	7.6E+02	1.3E-03
W-187	2.7E+02	3.6E-03
Ir-192	4.7E+02	2.1E-03
Au-198	2.3E+02	4.3E-03
Hg-203	1.3E+02	7.6E-03
Tl-204	5.8E-01	1.7E+00
Pb-210	7.6E-01	1.3E+00
Bi-207	9.1E+02	1.1E-03

Table 5-3, Continued

Radionuclide	DCF[a] rem per µCi · cm^{-3} · h	DRL[b] µCi · cm^{-3} · h
Bi-210	0.0E+00	0.0E+00
Po-210	5.1E-03	2.0E+02
Ra-226	3.9E+00	2.6E-01
Ac-227	7.2E-02	1.4E+01
Ac-228	5.5E+02	1.8E-03
Th-227	6.0E+01	1.7E-02
Th-228	1.1E+00	8.9E-01
Th-230	2.2E-01	4.5E+00
Th-232	1.1E-01	9.4E+00
Pa-231	1.7E+01	5.8E-02
U-232	1.5E-01	6.6E+00
U-233	1.4E-01	7.3E+00
U-234	8.7E-02	1.1E+01
U-235	8.8E+01	1.1E-02
U-236	6.9E-02	1.4E+01
U-238	5.9E-02	1.7E+01
U-240	4.1E-01	2.4E+00
Np-237	1.3E+01	7.6E-02
Np-239	9.6E+01	1.0E-02
Pu-236	6.8E-02	1.5E+01
Pu-238	5.0E-02	2.0E+01
Pu-239	4.7E-02	2.1E+01
Pu-240	4.9E-02	2.0E+01
Pu-241	0.0E+00	0.0E+00
Pu-242	4.2E-02	2.4E+01
Am-241	1.1E+01	9.2E-02
Am-242m	2.7E-01	3.7E+00
Am-243	2.9E+01	3.4E-02
Cm-242	5.6E-02	1.8E+01
Cm-243	7.3E+01	1.4E-02

Table 5-3, Continued

Radionuclide	DCF[a] rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[b] $\mu Ci \cdot cm^{-3} \cdot h$
Cm-244	4.8E-02	2.1E+01
Cm-245	4.1E+01	2.5E-02
Cm-246	4.0E-02	2.5E+01
Cf-252	4.3E-02	2.3E+01

[a]DCFs are expressed in terms of committed effective dose equivalent and are based on data from reference (DO-88).

[b]Assumes a PAG of one rem committed effective dose equivalent.

[c]The contribution from the short-lived daughter is included in the factors for the parent radionuclide.

Table 5-4 Dose Conversion Factors (DCF) and Derived Response Levels (DRL) for Doses Due to Inhalation[a]

Radionuclide	Lung Class	DCF rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[b] $\mu Ci \cdot cm^{-3} \cdot h$
H-3	V[c]	7.7E+01	1.3E-02
C-14	L ORG C[d]	2.5E+03	4.0E-04
Na-22	D	9.2E+03	1.1E-04
Na-24	D	1.5E+03	6.9E-04
P-32	W	1.9E+04	5.4E-05
P-33	W	2.8E+03	3.6E-04
S-35	W	3.0E+03	3.4E-04
Cl-36	W	2.6E+04	3.8E-05
K-40	D	1.5E+04	6.7E-05
K-42	D	1.6E+03	6.1E-04
Ca-45	W	7.9E+03	1.3E-04
Sc-46	Y	3.6E+04	2.8E-05
Ti-44	Y	1.2E+06	8.2E-07
V-48	W	1.2E+04	8.2E-05
Cr-51	Y	4.0E+02	2.5E-03
Mn-54	W	8.0E+03	1.2E-04
Mn-56	D	4.5E+02	2.2E-03
Fe-55	D	3.2E+03	3.1E-04
Fe-59	D	1.8E+04	5.6E-05
Co-58	Y	1.3E+04	7.7E-05
Co-60	Y	2.6E+05	3.8E-06
Ni-63	Vapor	7.5E+03	1.3E-04
Cu-64	Y	3.3E+02	3.0E-03
Zn-65	Y	2.4E+04	4.1E-05
Ge-68	W	6.2E+04	1.6E-05
Se-75	W	1.0E+04	9.8E-05
Rb-86	D	7.9E+03	1.3E-04
Rb-88	D	1.0E+02	1.0E-02
Rb-89	D	5.2E+01	1.9E-02
Sr-89	Y	5.0E+04	2.0E-05

Table 5-4, Continued.

Radionuclide	Lung Class	DCF rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[b] $\mu Ci \cdot cm^{-3} \cdot h$
Sr-90	Y	1.6E+06	6.4E-07
Sr-91	Y	2.0E+03	5.0E-04
Y-90	Y	1.0E+04	9.9E-05
Y-91	Y	5.9E+04	1.7E-05
Zr-93	D	3.8E+05	2.6E-06
Zr-95	D	2.8E+04	3.5E-05
Zr-97	Y	5.2E+03	1.9E-04
Nb-94	Y	5.0E+05	2.0E-06
Nb-95	Y	7.0E+03	1.4E-04
Mo-99	Y	4.8E+03	2.1E-04
Tc-99	W	1.0E+04	1.0E-04
Tc-99m	D	3.9E+01	2.6E-02
Ru-103	Y	1.1E+04	9.3E-05
Ru-105	Y	5.5E+02	1.8E-03
Ru/Rh-106[a]	Y	5.7E+05	1.7E-06
Pd-109	Y	1.3E+03	7.6E-04
Ag-110m	Y	9.6E+04	1.0E-05
Cd-109	D	1.4E+05	7.3E-06
Cd-113m	D	1.8E+06	5.5E-07
In-114m	D	1.1E+05	9.4E-06
Sn-113	W	1.3E+04	7.8E-05
Sn-123	W	3.9E+04	2.6E-05
Sn-125	W	1.9E+04	5.4E-05
Sn-126	W	1.2E+05	8.4E-06
Sb-124	W	3.0E+04	3.3E-05
Sb-126	W	1.4E+04	7.1E-05
Sb-127	W	7.2E+03	1.4E-04
Sb-129	W	7.7E+02	1.3E-03
Te-127m	W	2.6E+04	3.9E-05
Te-129	D	1.1E+02	9.3E-03

Table 5-4, Continued.

Radionuclide	Lung Class	DCF rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[b] $\mu Ci \cdot cm^{-3} \cdot h$
Te-129m	W	2.9E+04	3.5E-05
Te-131m	W	7.7E+03	1.3E-04
Te-132	W	1.1E+04	8.8E-05
Te/I-132[e]	W	1.2E+04	8.5E-05
Te-134	D	1.5E+02	6,5E-03
I-125	D	2.9E+04	3.4E-05
I-129	D	2.1E+05	4.8E-06
I-131	D	3.9E+04	2.5E-05
I-132	D	4.6E+02	2.2E-03
I-133	D	7.0E+03	1.4E-04
I-134	D	1.6E+02	6.3E-03
I-135	D	1.5E+03	6.8E-04
Cs-134	D	5.6E+04	1.8E-05
Cs-136	D	8.8E+03	1.1E-04
Cs/Ba-137[e]	D	3.8E+04	2.6E-05
Cs-138	D	1.2E+02	8.2E-03
Ba-133	D	9.4E+03	1.1E-04
Ba-139	D	2.1E+02	4.9E-03
Ba-140	D	4.5E+03	2.2E-04
La-140	W	5.8E+03	1.7E-04
La-141	D	7.0E+02	1.4E-03
La-142	D	3.0E+02	3.3E-03
Ce-141	Y	1.1E+04	9.3E-05
Ce-143	Y	4.1E+03	2.5E-04
Ce-144	Y	4.5E+05	2.2E-06
Ce/Pr-144[e]	Y	4.5E+05	2.2E-06
Nd-147	Y	8.2E+03	1.2E-04
Pm-145	Y	3.7E+04	2.7E-05
Pm-147	Y	4.7E+04	2.1E-05
Pm-149	Y	3.5E+03	2.8E-04

Table 5-4, Continued.

Radionuclide	Lung Class	DCF rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[b] $\mu Ci \cdot cm^{-3} \cdot h$
Pm-151	Y	2.1E+03	4.8E-04
Sm-151	W	3.6E+04	2.8E-05
Eu-152	W	2.7E+05	3.8E-06
Eu-154	W	3.4E+05	2.9E-06
Eu-155	W	5.0E+04	2.0E-05
Gd-153	D	2.9E+04	3.5E-05
Tb-160	W	3.0E+04	3.3E-05
Ho-166m	W	9.3E+05	1.1E-06
Tm-170	W	3.2E+04	3.2E-05
Yb-169	Y	9.7E+03	1.0E-04
Hf-181	D	1.9E+04	5.4E-05
Ta-182	Y	5.4E+04	1.9E-05
W-187	D	7.4E+02	1.3E-03
Ir-192	Y	3.4E+04	3.0E-05
Au-198	Y	3.9E+03	2.5E-04
Hg-203	D	8.8E+03	1.1E-04
Tl-204	D	2.9E+03	3.5E-04
Pb-210	D	1.6E+07	6.1E-08
Bi-207	W	2.4E+04	4.2E-05
Bi-210	D	1.9E+04	5.4E-05
Po-210	D	1.1E+07	8.9E-08
Ra-226	W	1.0E+07	9.7E-08
Ac-227	D	8.0E+09	1.2E-10
Ac-228	D	3.7E+05	2.7E-06
Th-227	Y	1.9E+07	5.2E-08
Th-228	Y	4.1E+08	2.4E-09
Th-230	W	3.9E+08	2.6E-09
Th-232	W	2.0E+09	5.1E-10
Pa-231	W	1.5E+09	6.5E-10
U-232	Y	7.9E+08	1.3E-09

Table 5-4, Continued.

Radionuclide	Lung Class	DCF rem per µCi · cm^{-3} · h	DRL[b] µCi · cm^{-3} · h
U-233	Y	1.6E+08	6.2E-09
U-234	Y	1.6E+08	6.3E-09
U-235	Y	1.5E+08	6.8E-09
U-236	Y	1.5E+08	6.6E-09
U-238	Y	1.4E+08	7.0E-09
U-240	Y	2.7E+03	3.7E-04
Np-237	W	6.5E+08	1.5E-09
Np-239	W	3.0E+03	3.3E-04
Pu-236	W	1.7E+08	5.8E-09
Pu-238	W	4.7E+08	2.1E-09
Pu-239	W	5.2E+08	1.9E-09
Pu-240	W	5.2E+08	1.9E-09
Pu-241	W	9.9E+06	1.0E-07
Pu-242	W	4.9E+08	2.0E-09
Am-241	W	5.3E+08	1.9E-09
Am-242m	W	5.1E+08	2.0E-09
Am-243	W	5.3E+08	1.9E-09
Cm-242	W	2.1E+07	4.8E-08
Cm-243	W	3.7E+08	2.7E-09
Cm-244	W	3.0E+08	3.4E-09
Cm-245	W	5.5E+08	1.8E-09
Cm-246	W	5.4E+08	1.8E-09
Cf-252	Y	1.9E+08	5.3E-09
Thyroid Dose			
Te/I-132[e]	W/D	2.9E+05	1.8E-05
I-125	D	9.6E+05	5.2E-06
I-129	D	6.9E+06	7.2E-07
I-131	D	1.3E+06	3.9E-06

Table 5-4, Continued.

Radionuclide	Lung Class	DCF rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[b] $\mu Ci \cdot cm^{-3} \cdot h$
I-132	D	7.7E+03	6.5E-04
I-133	D	2.2E+05	2.3E-05
I-134	D	1.3E+03	3.9E-03
I-135	D	3.8E+04	1.3E-04

[a]These factors and levels apply to adults (IC-75) and are based on Federal Guidance Report No. 11 (EP-88). They are also based on the lung class that results in the most restrictive value. DCFs are expressed in terms of committed effective dose equivalent, except for those for thyroid dose, which are in terms of committed dose equivalent.

[b]DRLs are based on a dose of 1 rem committed effective dose equivalent, except those for thyroid dose radionuclides, which are based on a committed dose equivalent of 5 rem.

[c]V denotes water vapor.

[d]L ORG C denotes labelled organic compounds.

[e]Contributions from short-lived daughters are included in the factors for parent radionuclides.

Table 5-5 Dose Conversion Factors (DCF) and Derived Response Levels (DRL) for a 4-Day Exposure to Gamma Radiation from Deposited Radionuclides[a]

Radionuclide	DCF[b] rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[b,c] $\mu Ci \cdot cm^{-3} \cdot h$
H-3	0.0E+00	0.0E+00
C-14	0.0E+00	0.0E+00
Na-22	8.3E+03	1.2E-04
Na-24	3.1E+03	3.2E-04
P-32	0.0E+00	0.0E+00
P-33	0.0E+00	0.0E+00
S-35	0.0E+00	0.0E+00
Cl-36	1.8E-04	5.4E+03
K-40	5.4E+02	1.9E-03
K-42	1.8E+02	5.7E-03
Ca-45	8.4E-07	1.2E+06
Sc-46	7.5E+03	1.3E-04
Ti-44	6.7E+02	1.5E-03
V-48	1.0E+04	1.0E-04
Cr-51	1.3E+02	7.8E-03
Mn-54	3.3E+03	3.0E-04
Mn-56	2.4E+02	4.1E-03
Fe-55	8.7E-01	1.1E+00
Fe-59	4.2E+03	2.4E-04
Co-58	3.8E+03	2.6E-04
Co-60	8.9E+03	1.1E-04
Ni-63	0.0E+00	0.0E+00
Cu-64	1.5E+02	6.8E-03
Zn-65	2.1E+03	4.7E-04
Ge-68	4.5E+00	2.2E-01
Se-75	1.7E+03	5.9E-04
Rb-86	3.3E+02	3.0E-03
Rb-88	1.0E+01	9.8E-02
Rb-89	2.9E+01	3.4E-02
Sr-89	5.2E-01	1.9E+00

Table 5-5, Continued.

Radionuclide	DCF[b] rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[b,c] $\mu Ci \cdot cm^{-3} \cdot h$
Sr-90	0.0E+00	0.0E+00
Sr-91	3.8E+02	2.6E-03
Y-90	0.0E+00	0.0E+00
Y-91	1.3E+01	7.8E-02
Zr-93	0.0E+00	0.0E+00
Zr-95	2.9E+03	3.5E-04
Zr-97	1.7E+02	5.8E-03
Nb-94	6.3E+03	1.6E-04
Nb-95	2.9E+03	3.4E-04
Mo-99	4.0E+02	2.5E-03
Tc-99	2.5E-03	4.0E+02
Tc-99m	5.3E+01	1.9E-02
Ru-103	1.9E+03	5.2E-04
Ru-105	2.1E+02	4.7E-03
Ru/Rh-106[d]	8.3E+02	1.2E-03
Pd-109	5.6E-01	1.8E+00
Ag-110m	1.2E+02	8.2E-03
Cd-109	3.7E+01	2.7E-02
Cd-113m	0.0E+00	0.0E+00
In-114m	3.8E+02	2.7E-03
Sn-113	5.9E+01	1.7E-02
Sn-123	2.6E+01	3.9E-02
Sn-125	1.0E+03	1.0E-03
Sn-126	2.4E+02	4.1E-03
Sb-124	6.8E+03	1.5E-04
Sb-126	9.9E+03	1.0E-04
Sb-127	1.9E+03	5.2E-04
Sb-129	3.7E+02	2.7E-03
Te-127m	2.6E+01	3.8E-02
Te-129	3.9E+00	2.6E-01

Table 5-5, Continued.

Radionuclide	DCF[b] rem per µCi · cm^{-3} · h	DRL[b,c] µCi · cm^{-3} · h
Te-129m	1.4E+02	7.2E-03
Te-131m	3.5E+01	2.8E-02
Te-132	6.6E+02	1.5E-03
Te/I-132[d]	6.7E+03	1.5E-04
Te-134	3.8E+01	2.7E-02
I-125	9.5E+02	1.0E-03
I-129	8.7E+02	1.2E-03
I-131	1.3E+04	7.4E-05
I-132	3.1E+03	3.2E-04
I-133	7.3E+03	1.4E-04
I-134	1.3E+03	7.5E-04
I-135	5.7E+03	1.8E-04
Cs-134	6.2E+03	1.6E-04
Cs-136	7.6E+03	1.3E-04
Cs/Ba-137[d]	2.4E+03	4.1E-04
Cs-138	6.8E+01	1.5E-02
Ba-133	1.7E+03	6.1E-04
Ba-139	3.2E+00	3.1E-01
Ba-140	7.0E+02	1.4E-03
La-140	4.1E+03	2.4E-04
La-141	8.9E+00	1.1E-01
La-142	2.3E+02	4.3E-03
Ce-141	3.3E+02	3.0E-03
Ce-143	4.8E+02	2.1E-03
Ce-144	8.5E+01	1.2E-02
Ce/Pr-144[d]	2.0E+02	5.0E-03
Nd-147	5.2E+02	1.9E-03
Pm-145	1.1E+02	8.7E-03
Pm-147	1.6E-02	6.2E+01
Pm-149	2.8E+01	3.6E-02

Table 5-5, Continued.

Radionuclide	DCF[b] rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[b,c] $\mu Ci \cdot cm^{-3} \cdot h$
Pm-151	5.5E+02	1.8E-03
Sm-151	2.1E-02	4.9E+01
Eu-152	1.5E+01	6.7E-02
Eu-154	4.8E+03	2.1E-04
Eu-155	2.8E+02	3.5E-03
Gd-153	5.0E+02	2.0E-03
Tb-160	4.1E+03	2.4E-04
Ho-166m	6.5E+03	1.5E-04
Tm-170	2.4E+01	4.1E-02
Yb-169	1.3E+03	7.4E-04
Hf-181	2.2E+03	4.5E-04
Ta-182	4.8E+03	2.1E-04
W-187	6.6E+02	1.5E-03
Ir-192	3.4E+03	3.0E-04
Au-198	1.1E+03	9.5E-04
Hg-203	9.6E+02	1.0E-03
Tl-204	5.1E+00	2.0E-01
Pb-210	1.2E+01	8.5E-02
Bi-207	6.0E+03	1.7E-04
Bi-210	0.0E+00	0.0E+00
Po-210	3.4E-02	3.0E+01
Ra-226	3.0E+01	3.3E-02
Ac-227	8.4E-01	1.2E+00
Ac-228	3.3E+02	3.0E-03
Th-227	4.3E+02	2.3E-03
Th-228	1.1E+01	9.2E-02
Th-230	3.6E+00	2.8E-01
Th-232	2.6E+00	3.8E-01
Pa-231	1.4E+02	7.1E-03
U-232	4.1E+00	2.5E-01

Table 5-5, Continued.

Radionuclide	DCF[b] rem per $\mu Ci \cdot cm^{-3} \cdot h$	DRL[b,c] $\mu Ci \cdot cm^{-3} \cdot h$
U-233	2.0E+00	5.1E-01
U-234	3.2E+00	3.1E-01
U-235	6.7E+02	1.5E-03
U-236	2.9E+00	3.5E-01
U-238	2.5E+00	3.9E-01
U-240	3.3E+00	3.0E-01
Np-237	1.3E+02	7.8E-03
Np-239	4.5E+02	2.2E-03
Pu-236	3.9E+00	2.6E-01
Pu-238	3.4E+00	3.0E-01
Pu-239	1.5E+00	6.7E-01
Pu-240	3.2E+00	3.1E-01
Pu-241	0.0E+00	0.0E+00
Pu-242	2.7E+00	3.7E-01
Am-241	1.2E+02	8.5E-03
Am-242m	1.1E+01	9.2E-02
Am-243	2.6E+02	3.8E-03
Cm-242	3.7E+00	2.7E-01
Cm-243	5.8E+02	1.7E-03
Cm-244	3.3E+00	3.1E-01
Cm-245	3.4E+02	3.0E-03
Cm-246	2.9E+00	3.5E-01
Cf-252	2.5E+00	4.0E-01

[a]Entries are calculated for gamma exposure at 1 meter above the ground surface (DO-88).

[b]All radioactivity is assumed to be deposited at the beginning of the incident. Deposition velocities are taken as 1 $cm \cdot sec^{-1}$ for radioiodines and 0.1 $cm \cdot sec^{-1}$ for other radionuclides. (See p. 5-24).

[c]Assumes a PAG of 1 rem committed effective dose equivalent.

[d]Contributions from short-lived daughters are included in the factors for parent radionuclides.

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CHAPTER 6

Implementing the PAGs for the Intermediate Phase
(Food and Water)

See Chapter 3 and Appendix D for Current Implementation Recommendations for Food. Also refer to the following documents:

Federal Emergency Management Agency Guidance Memorandum IN-1, The Ingestion Exposure Pathway. February 26, 1988 Federal Emergency Management Agency. Washington, DC 20472

Guidance on Offsite Emergency Radiation Measurement Systems Phase 2, The Milk Pathway, FEMA REP-12, September 1987.

Guidance on Offsite Emergency Radiation Measurement Systems. Phase 3, Water and Non-Dairy Food Pathway, September 1989.

HHS Publication FDA 82-8196

Background for Protective Action Recommendations: Accidental Radioactive Contamination of Food and Animal Feeds

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WHO Collaborating Centers for:
- Standardization of Protection Against Nonionizing Radiations
- Training and General Tasks in Radiation Medicine
- Nuclear Medicine

August 1982

U.S. DEPARTMENT OF HEALTH AND HUMAN SERVICES
Public Health Service
Food and Drug Administration
Bureau of Radiological Health
Rockville, Maryland 20857

FOREWORD

The Bureau of Radiological Health develops and carries out a national program to control unnecessary human exposure to potentially hazardous ionizing and nonionizing radiations and to ensure the safe, efficacious use of such radiations. The Bureau publishes the results of its work in scientific journals and in its own technical reports.

These reports provide a mechanism for disseminating results of Bureau and contractor projects. They are distributed to Federal, State, and local governments; industry; hospitals; the medical profession; educators; researchers; libraries; professional and trade organizations; the press; and others. The reports are sold by the Government Printing Office and/or the National Technical Information Service.

The Bureau also makes its technical reports available to the World Health Organization. Under a memorandum of agreement between WHO and the Department of Health and Human Services, three WHO Collaborating Centers have been established within the Bureau of Radiological Health, FDA:

WHO Collaborating Center for Standardization of Protection Against Nonionizing Radiations;

WHO Collaborating Center for Training and General Tasks in Radiation Medicine; and

WHO Collaborating Center for Nuclear Medicine.

Please report errors or omissions to the Bureau. Your comments and requests for further information are also encouraged.

John C. Villforth

John C. Villforth
Director
Bureau of Radiological Health

PREFACE

By FEDERAL REGISTER action of March 11, 1982 (47 FR 10758), the Federal Emergency Management Agency (FEMA) outlined the responsibilities of several Federal agencies concerning emergency response planning guidance that the agencies should provide to State and local authorities. This updated a prior notice published in the FEDERAL REGISTER by the General Services Administration (GSA) on December 24, 1975 (40 FR 59494), on the same subject. GSA responsibility for emergency management was transferred by Executive Order to FEMA. The Department of Health and Human Services (HHS) is responsible for assisting State and local authorities in developing plans for preventing adverse effects from exposure to radiation in the event that radioactivity is released into the environment. This includes developing and specifying protective actions and associated guidance to State and local governments for human food and animal feeds.

Proposed recommendations were published in the FEDERAL REGISTER on December 15, 1978 (43 FR 58790) and a background document accompanied their publication. Twenty-one comment letters were received in response to the proposal in addition to comments from various Federal agencies. Review of these comments led to changes in the recommendations and supporting rationale, dosimetric and agricultural models, and cost/benefit analysis. These changes have been incorporated into this background document, which is intended to accompany and support FDA's final recommendations on Accidental Radioactive Contamination of Human Foods and Animal Feeds: Recommendations for State and Local Agencies. The final recommendations will appear in the FEDERAL REGISTER.

This background report discusses the rationale for the Protective Action Guides; the dosimetric and agricultural models used in their calculation; some methods of analysis for radionuclide determination; appropriate protective actions; and cost considerations.

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ABSTRACT

Shleien, B., G.D. Schmidt, and R.P. Chiacchierini. Background for Protective Action Recommendations: Accidental Radioactive Contamination of Food and Animal Feeds. HHS Publication FDA 82-8196 (August 1982) (pp. 44).

This report provides background material for the development of FDA's Protective Action Recommendations: Accidental Radioactive Contamination of Food and Animal Feeds. The rationale, dosimetric and agricultural transport models for the Protective Action Guides are presented, along with information on dietary intake. In addition, the document contains a discussion of field methods of analysis of radionuclides deposited on the ground or contained in milk and herbage. Various protective actions are described and evaluated, and a cost-effectiveness analysis for the recommendations performed.

CONTENTS

somatic risk. Mortality rather than incidence estimates are employed in the comparisons. In the case of comparisons to natural background radiation, use of mortality data or incidence estimates would yield the same numerical PAG limits, because these limits are based on a comparison between risks rather than an evaluation of absolute risk.

The radiation doses in the event of a contaminating accident will most likely result from ingestion of the fission products cesium-134 and -137; strontium-89 and -90; and iodine-131. For the purpose of this analysis it is assumed that all projected extra cancers can be attributed to internal radiation via the food pathway (i.e., the risks from ingested radioactive material is the same as that from external radiation).

The BEIR-III (3) best estimate of lifetime cancer risk (linear quadratic model) for a single exposure to low-dose, low LET radiation is from 0.77 to 2.26 x 10^{-4} deaths per person-rem, depending on whether the absolute or relative-risk projection model is used (calculated from Table 1). The equivalent risk estimate from BEIR-I (2) is 1.17 to 6.21 x 10^{-4} deaths per person-rem.

Table 1. Risk estimates for single dose

Dose response model	Deaths per million persons per 10 rads single dose whole-body BEIR-III	
	Absolute risk	Relative risk
Linear quadratic	766	2255
Linear	1671	5014
Quadratic	95	276

These risk estimates are for a single dose of 10 rem, because limitations of the scientific information do not justify estimates at lower doses according to the BEIR Committee. Because of the uncertainty of risk estimates at low doses, BEIR-III provided risk estimates based on a linear model and a pure quadratic dose response model as well as estimates based on the preferred linear quadratic model. The risk estimates for the linear model are about a factor of 2 higher and those of the quadratic model and about a factor of 8 lower than those of the linear quadratic model. It should further be noted, that BEIR-III does not recommend that their risk estimates be extrapolated to lower doses because of the inadequacies of the scientific basis. BEIR-III does recognize however that Federal agencies have a need to estimate impacts at lower doses. While BEIR-III prefers the linear-quadratic dose response model as the best estimate, regulatory agencies have continued to favor the linear model as the basis for making risk estimates. While the BEIR-III estimates will be used here to estimate the impact (health effects) at lower doses, it is fully recognized that current scientific opinion leaves alternatives as to which dose response and risk model to use.

As previously stated, for the purpose of setting PAG's, comparison of radiation risks to those from natural disasters is considered the approach of choice in this document.

1.2.2 Genetic Risk Evaluation

The model for genetic risks from radiation exposure is described in the BEIR-III report (3). In the first generation, it is estimated that 1 rem of parental exposure throughout the general population will result in an increase of 5 to 75 additional serious genetic disorders per million liveborn offspring. The precision for estimating genetic risks is less precise than those for somatic risks. Given the broad range, genetic risks are evaluated, but are not precise enough to be a basis for setting the PAG's.

1.3 ASSESSMENT OF COMMON SOCIETAL AND NATURAL BACKGROUND RADIATION RISKS

1.3.1 Common Societal Risks

As previously stated, one method of determining the acceptability of a risk is by comparing prevalent or normal risks from hazards common to society. A list of the annual risks from common societal hazards is given in Table 2. Comparision of radiation risks to commonly accepted societal risks assumes that the age dependencies are similar and that all individuals are equally exposed to the hazard. This latter assumption is, of course, not entirely valid in that persons nearer a nuclear power plant or a dam, or in an earthquake or tornado area might be expected to be at greater risk than persons living at a distance from the particular hazard.

Table 2. Annual risk of death from hazards common to society

Category	Reference	Risk of death (per person per year)
All disease	(4)	8×10^{-3}
Leukemia and all other cancer	(5)	1.5×10^{-3}
Motor vehicle accidents	(6)	3×10^{-4}
Accidental poisoning	(6)	1×10^{-5}
Air travel	(7)	9×10^{-6}
Tornadoes (Midwest)	(8)	2×10^{-5}
Earthquakes (Calif.)	(8)	2×10^{-6}
Floods (46 million at risk)	(9)	2.2×10^{-6}
Catastrophic accidents (tornadoes, floods, hurricanes, etc.)	(10)	1.2×10^{-6}
Natural disasters	(11)	9×10^{-7}
	(6)	8×1^{-7}
Tornadoes	(7)	0.4×10^{-6}
	(9)	0.6×10^{-6}
Hurricanes	(7)	0.4×10^{-6}
	(9)	0.3×10^{-6}
Floods	(8)	2×10^{-6}
	(9)	0.5×10^{-6}
Lightning	(7)	0.5×10^{-6}
Winter storms	(9)	0.4×10^{-6}
Natural disasters (sum of above)		2.1 to 3.9×10^{-6}

Table 2 indicates that the annual individual risk from natural disasters is approximately 1 to 4×10^{-6}. This risk represents a common risk level, which is generally not considered in selecting place of residence. At this level of risk, some action to prevent further loss of life could be expected by society following the occurrence of a natural disaster. It thus appears prudent to evaluate the somatic risks from radiation in relation to the risk of death from a natural disaster. For comparison purposes, a value of 1 in a million (1×10^{-6}) annual risk of death, which is often quoted as an acceptable risk, will be used as the risk of natural disasters. Actual data indicate that the risk of natural disasters may be a 2 or 3 times greater risk than this value. For a risk of death of 1×10^{-6} per year, the lifetime accepted societal risk would be about 70×10^{-6}. This is equivalent to a single radiation dose of 140 to 420 mrem, using the linear model, or 310 to 910 mrem using the BEIR-III linear quadratic model (see Table 1). The upper and lower ranges are those obtained from employment of relative and absolute risk models and the dose response extrapolations mentioned above (from calculations based on data in Table 1). Genetic effects are not considered in evaluating common societal hazards because of the difficulty in assessing deaths occurring from genetic consequences, either natural or radiation induced. If spontaneous abortions are deleted from this category, then fatal genetic effects are a small portion of the overall genetic impact on health. However, it is difficult to accurately evaluate genetic effects, and even more difficult to compare its impact to the impact of somatic effects in an effective manner.

1.3.2 Risks From Natural Radiation

Further perspective on acceptable risk can be obtained by examining the risks of natural background radiation. In risk assessments where a radiation risk is compared to that from the natural radiation background, the question is which variable associated with natural background should be used to determine "acceptable risk?" Since background radiation has always been a part of the natural environment, a plausible argument might be to assume that the risks associated with the average natural radiation dose represent an "acceptable risk."

It has also been argued that because of the ever present risk from natural radiation, a level of manmade radiation ought to be acceptable if it is "small" compared to natural background (12). It has been suggested that "small" be taken as the standard deviation of the population-weighted natural background (13). In previous evaluations that led to the FDA's proposed PAG recommendations (14) the geographic variable (two standard deviations) in the natural radiation dose was used as a point of comparison for judging acceptable radiation risk (15). This value, calculated on a State-by-State basis assuming a log-normal distribution and not weighted for population, is 8.5 mrem per year. The cumulative lifetime dose equivalent would thus be about 500 mrem, which was the basis for the proposed PAG recommendation for the whole body at the Preventive PAG level. The Environmental Protection Agency (EPA), in a further analysis of previously published data (16), has calculated the cumulative distribution of dose equivalent in the U.S. population. These data show that 95 percent of the population receives between 28 and 84 mrem/year from cosmic and terrestrial background radiation (17). The actual distribution is asymmetric and not log-normal. Thus, one-half of this 95-percent increment range, or 28 mrem/year, will be taken as the value for judging acceptable risk. Adler (13) notes that one standard deviation of the natural external and internal radiation background derived from earlier sources (18) is 20 mrem. Personal conversations with Adler revealed that this estimate is based on air exposures rather than dose equivalent (mean whole body) and involved a broad rounding off of values. At the 95-percent increment value (latest EPA data) of 28 mrem/year (19), the additional lifetime dose over 70 years is about 2000 mrem. About 6 million persons (2-1/2 percent of the population) receive lifetime doses that exceed the mean background radiation dose by this amount or more.

Another possibility, especially applicable to setting limits for internal emitters, is using the variation in internal natural radiation dose as a reference for establishing an acceptable standard for PAG's. For PAG limits for radionuclides via the ingestion pathway, doses to organs other than the lungs are most pertinent. Using this suggestion still requires a judgmental decision as to whether the variation in internal natural radiation dose is "small." A summary of internal natural radiation doses is given in Table 3. It is apparent that natural radiation doses to human tissues and organs is determined mainly by potassium-40 concentration. The average annual internal whole-body radiation dose per person from ingested natural radioactivity is 19.6 mrem, of which 17 mrem is due to potassium-40.

In potassium-40 whole-body measurements of 10,000 persons, a standard deviation of about 12 percent (95-percent confidence level of 23.52 percent) was observed (20). The study further concluded that the standard deviation is also the same for different groups of age and sex, and therefore, it may be concluded that the same biological variation exists for all the different age-sex groups. In another study based on the chemical determinations of total body potassium the average amount in a 70-kg man was estimated to be 136 g with a standard deviation of ±28 g or ±20 percent (95-percent confidence increment of ±40 percent) (21).

Table 3. Annual internal radiation dose per person for non-inhaled natural radioactivity[a]

Annual dose (mrads/year) whole-body average (unless otherwise noted)	
H-3	0.001
Be-7	0.008
C-14	1.3
Na-22	0.02
K-40	17
Rb-87	0.4
U-238-U-234 series	0.043[b]
Ra-222	0.064[b]
Po-210	0.7
Ra-226	0.031[b]
Th-230	0.04[b]
Th-232	0.04[b]
Total	19.65

[a]UNSCEAR (1977).
[b]Based on soft tissue dose (lung, testes, and ovaries)

An indirect means of determining the variability of whole-body potassium values is based on the constant ratio of mean potassium values to total body water up to age 50 (20). The 95-percent confidence increment for the variability of total body water in males, ages 16 to 90 is 16 percent, while for females it is 13 percent for ages 16 to 30 and 21 percent for ages 31 to 90 (22).

From the above data, it appears that the increment for the 95-percent confidence level for whole-body potassium, and hence potassium-40, is between ± 15 percent and ± 40 percent. Note that this variability may be due to differences in body water or body weight. Only in the case of one study (21) is it clear the total body weight is considered a constant. It is apparent that a range of values between approximately 3 to 7 mrad per year may be used to describe the variability in natural potassium-40 dose to the population on a whole-body dosimetric basis. The mid-point of this range is 5 mrad per year or a lifetime dose commitment (70 years) of 350 mrem.

Thus, the lifetime radiation dose associated with the variability in natural radiation is about 350 mrem (internal) and 2000 mrem (external).

1.4 PREVENTIVE AND EMERGENCY PAG'S

PAG's have been proposed for two levels of response:

1. Preventive PAG - applicable to situations where protective actions causing minimal impact on the food supply are appropriate. A preventive PAG establishes a level at which responsible officials should take protective action to prevent or reduce the concentration of radioactivity in food or animal feed.

2. Emergency PAG - applicable to incidents where protective actions of great impact on the food supply are justified because of the projected health hazards. An Emergency PAG establishes a level at which responsible officials should isolate food containing radioactivity to prevent its introduction into commerce, and at which the responsible officials must determine whether condemnation or another disposition is appropriate.

1.4.1 Preventive PAG

During recent years numerous reports on risks and risk/benefit assessments for the evaluation of technological insults have been published. A number of these have concluded that an annual risk of death of 1 in a million is acceptable to the public (8). The total average annual risk to the U.S. population from natural disasters appears to be about 2 or 3 times greater than the 1 in a million annual risk. Those individuals living in certain flood plains, tornado, or earthquake areas accept risks that may be greater than the average by a factor of 2 or more (See data for tornadoes and earthquakes in Table 2).

As previously mentioned, based on BEIR-III (3) upper risk estimates, a 1 in a million annual risk of death corresponds to a single radiation dose of 140 to 910 mrem.

It is our conclusion that an annual risk of 1 in a million provides a proper perspective for setting food protective actions guides (PAG's) for radiation contamination accidents of low probability. It appears that most individuals in the United States will never be exposed to such a radiation contamination accident and that any one individual is not likely to be potentially exposed more than once in his lifetime.

Based on the above considerations, the uncertainty in radiation risk estimates and the uncertainty in the average natural disaster risks, a value of 0.5 rem whole body is selected for the Preventive PAG. Thus, at projected doses of 0.5 rem from contaminated food, it is recommended that protective actions having low impacts be taken for protection of the public. The specific value of 0.5 rem represents a judgment decision rather than a specifically derived value from specific models and assumptions.

Further perspective on acceptable risks for setting the PAG's is the risks associated with natural background radiation. The discussion above indicates that lifetime dose associated with the 95-percent increment of the variability in natural radiation is about 350 mrem internal and 2000 mrem external (that is, 2-1/2 percent of the population receives doses greater than the average by this amount or more).

This Preventive PAG is applicable to whole-body radiation exposure and to major portions of the body including active marrow (ingestion of strontium) in conformity with current U.S. radiation protection practice. Coincidently, 0.5 rem is the Federal Radiation Council's (FRC) annual limit for individuals of the general population (23).

Present convention, recommended by the Federal Radiation Council (23) based on prior estimates of relative radiation risks for various organs indicates that radiation limits for the thyroid gland be set at 3 times those for the whole body. More recent scientific information indicates that the risks from organ doses relative to whole body differ from those assumed when the current U.S. regulations and FRC guidance were established. The International Commission on Radiological Protection (ICRP) in revising its recommendations on internal exposure derived weighting factors that represent the ratio of risk from irradiation to a given tissue (organ) to the total cancer risk due to uniform irradiation to the whole body. The ICRP weighting factors are 0.12 for red bone marrow and 0.03 for thyroid, indicating that the cancer risk is 8 times less for red bone marrow and 33 times less for thyroid than for whole body exposure (24). Further considerations of effects other than cancer resulted in the limitation of organ doses to 50 rems per year for occupational workers. Thus the ICRP recommendations in effect provide for or allow single organ doses that are 8 times greater for red bone marrow and 10 times greater for thyroid than for whole body. The EPA has recently proposed Federal guidance for occupational radiation protection that incorporates the basic ICRP recommendations (46 FR 7836, Jan. 23, 1980). Setting the Preventive PAG at 0.5 rem for whole body and red bone marrow and 1.5 rem for thyroid provides significantly more protection from the actual risks of organ doses than from whole-body risks. To the extent that the whole-body risk is considered acceptable, the red bone marrow and thyroid limits are conservative by factors of 8 and 3.3, respectively.

1.4.2 Emergency PAG

The philosophy of the protective action guidance of FDA is that low impact protective actions should be initiated when contamination of food exceeds the Preventive PAG. The intent is that such protective actions be implemented to prevent the appearance of radioactivity in food at levels that would require the condemnation of food. If such actions are ineffective, or high levels appear in food, then the Emergency PAG is that level at which higher impact (cost) protective actions are warranted. At the Emergency PAG radiation level, action should be taken to isolate and prevent the introduction of such food into commerce and to determine whether condemnation or other disposition is appropriate.

With regard to the numerical relationship between the Preventive PAG level and the Emergency PAG level, prior conventions may be considered. For example, the Federal Radiation Council (23) assumed that the dose to the most highly exposed individual does not vary from the average dose to the whole population by a factor greater than three; Hence, a factor of 3 was used to define the difference between maximum and average population limits. Traditionally, it has been more common to use a factor of 10 as a safety factor, such as between occupational and general public limits. A factor of 10 difference between the Emergency and Preventive PAG levels, based on these traditional radiation protection approaches has in the past been thought to introduce a sufficient level of conservatism. The proposed PAG's (14) adopted this rationale in setting the Emergency PAG's. The analyses of costs, to follow, also indicate that a factor of 10 between the Preventive PAG and Emergency PAG is appropriate. As calculated in the last chapter of this report the cost of condemnation of milk (high impact protective action) is about a factor of 10 greater than the cost of using uncontaminated stored feed (low impact protective action). Since contamination of the milk pathway is considered to be the most probable and significant food problem, this is the only pathway that is cost analyzed.

The use of a factor of 10 adopted here results in an Emergency PAG of 5 rem for the whole body which numerically is equivalent to the current occupational annual limit. This limit permitted each year over a working lifetime is associated with the expectation of minimal increased radiation risks.

1.5 EVALUATION OF PAG RISKS

The risks associated with a radiation dose equal to the PAG's can be readily calculated from the BEIR-III risk estimates in Table 1. For the Preventive PAG of 0.5 rem, the deaths per million persons exposed are one-twentieth of those given for the 10-rad single dose (or about 38 to 250 deaths for the linear quadratic and linear models respectively). On an individual basis, this is a risk of death of 0.38 to 2.50 x 10^{-4} (0.0038 to 0.025 percent) over a lifetime. BEIR-III gives the expectation of cancer deaths in the U.S. population as 167,000 per million or an individual expectation of cancer death of 16.7 percent.

As noted above, the BEIR-III estimate of serious first generation genetic disorders is 5 to 75 per million live offspring per rem of parental exposure. Thus, for a dose of 0.5 rem, the expectation is 2.5 to 38 disorders per million live offspring. BEIR-III notes the current estimate of the incidence of serious human disorders of genetic origin as roughly 10 percent of liveborn offspring.

CHAPTER 2. DOSIMETRIC MODELS, AGRICULTURAL TRANSPORT MODELS, DIETARY INTAKE, AND CALCULATIONS

2.1 DOSIMETRIC MODELS

2.1.1 Introduction

The dosimetric models and metabolic parameters for estimating the dose from internally deposited radionuclides are in a state of flux. The recent reports and current activity represent the first major revision since the adoption of ICRP Publication 2 (25) and NCRP Report 22 (26) in 1959. ICRP Publication 30 (27) superseded ICRP Publication 2 and revised the basic approach in setting limits for intake of radionuclides by workers. The ICRP recommendations are intended to avoid nonstochastic effects and to limit the occurrence of stochastic effects to an acceptable level. This approach includes the use of weighting factors to sum the risk from organ doses in setting the limits for intakes. This system contrasts with the earlier approach where limits were based on the dose to the "critical organ."

The ICRP Publication 30 approach has considerable merit, but is not yet widely accepted in the United States. Its use in calculating the derived response levels would represent a major change. Accordingly, the approach is to use the organ to whole-body dose relationship of current U.S. regulations and to select critical organ dose conversion factors that are based on current dosimetric models and metabolic parameters. Where apropriate, the recent ICRP and NCRP organ dose models will be accepted as representing current scientific opinion. It should be noted that future reports and revisions by NCRP, MIRD (Medical Internal Radiation Dose Committee of The Society of Nuclear Medicine) and other Federal agencies may necessitate a revision of the dose conversion values selected here.

The PAG's are applicable to the most critical or sensitive segment of the population. In most cases this means that the infant or child is the critical segment. In the case of the Emergency PAG, derived response levels are also presented for the adults. This permits greater flexibility in the choice of protective actions in cases where infants are not present or can be excluded from use of the specific food item being considered.

2.1.2 Iodine-131: Dose To Thyroid

Fetal uptake of iodine begins at about the 9th week of gestation and reaches a maximum in the newborn infant (28,29) before returning to adult levels. Kereiakes et al. (30) report that thyroid uptake during the first 2 weeks of life is very high and report a value of 70 percent of that administered for the newborn. The radioiodine uptake expressed per unit thyroid weight remains high for the newborn and infant and only gradually decreases throughout childhood and adolescence to adult levels. The newborn infant will be taken as the most critical segment of the population because factors concerning intake, uptake, and radiosensitivity indicate that the thyroid gland of an infant receives a higher radiation dose per unit I-131 ingested than any other age group (30). However, it is interesting to note that data indicate that only about 3 percent of infants are given whole cow's milk at 1 month of age and about 1 percent at 10 days (31). Hence, assuming that all infants are given whole milk provides a conservative estimate of infant thyroid doses.

Data on the dose to the fetal thyroid from I-131 ingested by the mother is rather limited. The study of Dyer and Brill (29) reports an increase in the fetal thyroid dose from 0.7 to 5.9 rads per µCi administered to the mother for fetal ages of 13 to 22 weeks. It thus appears that the fetal thyroid dose is less than that of the newborn infant ingesting I-131 contaminated milk.

The current literature on normal thyroid uptake in U.S. adults shows 24-hour uptake has decreased from about 30 percent reported in the 1960's to about 20 percent or less in current comparable studies. Kereiakes et al. (32) use a 20-percent uptake for all ages. This reduced uptake apparently results from changes in the U.S. diet, whereas ICRP 30 (27) has continued to use an uptake of 30 percent to reflect world averages.

The Medical Internal Radiation Dose (MIRD) Committee schema (33) has been used by Wellman and Anger (34) to calculate dose factors per µCi ingested for the newborn for the 1-, 5-, 10-, and 15-year-old child, and for the adult. These factors were then modified by Kereiakes et al. (32) to reflect a 20-percent uptake for all ages. A biological half-life of 68 days is used for all ages, which results in an effective half-life of 7.2 days. Although there is some evidence that the biological half-life for the infant is less, the radiation dose is largely controlled by the radiological half-life and use of a single value appears appropriate here. Because of some uncertainty regarding the fetal thyroid dose and lack of acceptance by national and international groups, the older data (27-percent uptake) of Wellman and Anger (34) will be used. This provides some additional conservatism in the derived response levels for I-131. The cumulative activity is 2.08 µCi-days per µCi ingested (administered).

Table 4. Summary of I-131 dose conversion factors

Age	Thyroid weight (g)	Dose rad/ µCi
Newborn	1.5	16.0
1 yr.	2.2	10.9
5 yrs.	4.7	5.1
10 yrs.	8.0	3.0
15 yrs.	11.2	2.1
Adult	16.0	1.5

For the infant and adult, the dose conversion factors to be used are 16.0 and 1.5 rad/ µCi ingested.

2.1.3 Cesium-137 and -134: Dose To Whole Body

The NCRP (35) has reviewed the behavior of Cs-137 in the environment and its metabolism and dose to man. From studies of Cs-137 in food chains, the biological half-life is found to vary from 15 ±5 days in infants to 100 ±5 days in adults. The biological half-life in pregnant women is reported to be 1/2 to 2/3 that in nonpregnant women and consequently the dose to the fetus is also reduced.

Retention of Cs-137 in the adult is stated to be well represented by a 2-exponential equation with biological half-times of 1.4 days and 135 days applicable to retention in body fluid and soft tissues, respectively. Integration of this equation yields an accumulated activity of 170 µCi-days per µCi of intake. This accumulated activity may be expressed in terms of a single retention function yielding a value of 118 days.

The dosimetry of internally deposited Cs-137 in infants and adults is treated separately for the beta particle and photon components. The difference between infants and adults is a smaller photon contribution to the infant because of the smaller body size. For a uniform concentration of 1 µCi/kg of body weight, the total beta and photon dose rate is 19 mrad/day to the infant and 25 mrad/day to the adult. Use of the above accumulated activity factor of 170 µCi-days per µCi intake yields a dose of 0.061 rad/ µCi intake for the adult. Assuming an effective retention time of 20 days for the infant, the corresponding factor for the infant is 0.071 rad/ µCi intake. The use of a smaller effective retention time (15 days as noted in NCRP (26)) or 10 days as used in NUREG-0172 (36) would reduce the infant dose conversion factor. The use of 20 days thus tends to overestimate the infant dose.

It is also important to consider the dose from Cs-134 which, depending on operating history, occurs in nuclear reactor fuel at levels equal to or greater than that of Cs-137. Unfortunately, published dosimetry data for Cs-134 for the infant are rather limited. Conversion factors for the adult that use current models and metabolic data are found in ORNL/NUREG/TM-190 (37). Johnson et al. (38) have used this same data base to compute committed effective dose equivalent conversion factors for both infants and adults. The mean absorbed dose per cumulated activity factor for the infant were modified by the ratio of adult and infant organ mass (with a further correction to photon component based on absorbed fraction) to produce the infant factors. It was stated that this procedure may underestimate the infant dose.

The approach adopted here will be to modify the adult dose conversion factor in ORNL/NUREG/TM-190 (37) based only on relative body weight and cumulated activity (effective retention half-times). The dose conversion factor for cesium-134 from ORNL/NUREG/TM-190 adult whole body is 0.068 rem/μCi ingested and the estimated factor for the infant is then 0.118 rem/μCi ingested. This value should overestimate the infant dose and is conservative.

Summary of Cs-134 and Cs-137 Dose Conversion Factors

Table 5. Summary of dose factors for Cs-134 and Cs-137

	Infant	Adult
Body Mass	7,700g	70,000g
Uptake to whole body[a]	1.0	1.0
T biological (days)	20	118
T effective Cs-134 (days)	19.5	102
Cs-137 (days)	20	118
Dose conversion (rem/μCi ingestion)		
Cs-134	0.118	0.068
Cs-137	0.071	0.061

[a]For cesium-134, ORNL/NUREG/TM-190 uses uptake of 0.95.

2.1.4 Strontium: Dose To Bone Marrow

The tissues at greatest risk in the skeleton have been identified as the active red marrow in trabecular bone and endosteal cells near bone surfaces (generally referred to as bone surface). Spiers and his coworkers (39,40) have developed methods to calculate the absorbed doses, D_m and D_s, received by red marrow and bone surfaces, respectively from beta-emitting radionuclides uniformly distributed throughout the volume of bone. They consider the dose, D_o, in a small, tissue-filled cavity in an infinite extent of mineral bone uniformly contaminated with the radionuclide and give dose factors D_s/D_o and D_m/D_o for obtaining the absorbed doses. For both Sr-89 and Sr-90, the ratio of D_s/D_m is about 1.5. Therefore, since the dose limit recommendations are 15 rem to bone and 5 rem to red marrow (occupational limits), the dose to red marrow is the limiting criterion and will be used in this report (26).

The work of Spiers and his coworkers has been used by the ICRP (27) in calculating dose commitment factors for adults and by Papworth and Vennart (41) for doses as a function of age at times of ingestion. The dose commitment values from Papworth and Vennart in red marrow per μCi of Sr-89 and Sr-90 ingested are as follows (Table 6).

Table 6. Dose commitment values for Sr-89 and Sr-90

Age at Ingestion (years)	rem per µCi ingested[a] Sr-89	Sr-90
0	0.414	4.03
0.5	0.194	2.49
1	0.130	1.83
2	0.080	1.20
3	0.060	0.91
4	0.050	0.77
5	0.044	0.70
6	0.039	0.69
7	0.035	0.71
8	0.032	0.76
9	0.029	0.83
10	0.026	0.89
11	0.023	0.94
Adults[a]	0.012	0.70

[a]Ages 0-11 from Papworth and Vennart (41), Adults from ICRP-30 Supplement to Part 1 (27).

Thus the dose conversion factors adopted in rem/µCi ingested are for Sr-89, 0.194 and 0.012, and for Sr-90, 2.49 and 0.70, for the infant and adult, respectively. As before, the 0.5-year infant is taken as the critical population. The values for the adult are those given in ICRP 30 (27) which also used the work of Spiers and coworkers.

2.1.5 Other Radionuclides

Adult - The most authoritative reference using current data and models for dose conversion factors per µCi ingested is that of the ICRP 30 (27) Part 1, 1979; Part 2, 1980; Part 3, 1981; and the Supplements, Pergamon Press (42). ICRP 30 Parts 1, 2, and 3 (and Supplements) provide data only for adults (occupational workers) for the radionuclides of 94 elements.

As a further resource, ORNL/NUREG/TM-190 is suggested (37). This document represents initial efforts by ORNL under contract to NRC to provide review and update on internal dosimetry.

Infant, Child - Unfortunately the initial efforts to update internal dosimetry have all been directed at the adult or occupational worker and comprehensive dosimetry using current data for the younger age groups do not exist. Further efforts are in process or contemplated by the U.S. Nuclear Regulatory Commission (NRC), Medical Internal Radiation Dose Committee, (MIRD), Society of Nuclear Medicine, and NCRP, and generally will involve or use the models developed by Oak Ridge National Laboratory (ORNL). Until such time that new dosimetric calculations appear for the younger age group, it is suggested that Nuclear Regulatory Commission Regulatory Guide 1.109 be used: "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Regulatory Guide 1.109, Revision 1, Oct. 1977, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

2.2 AGRICULTURE TRANSPORT MODELS

A review of the agricultural transport mechanism for radionuclides, which employs parameters appropriate for the U.S. experience, is contained in the Reactor Safety Study, WASH-1400, Appendix VI (7). An analysis specific for calculating derived response levels (concentration values) in agricultural media for emergency action that reflects the British experience is found in a report of the Medical Research Council (43).

A more recent and comprehensive assessment of the transport mechanisms for the forage-cow-milk pathway is found in UCRL-51939 and will be used here (44)

2.2.1 Transport Models

According to Ng et al., the time dependency of the concentration of a radionuclide in the milk of a cow continuously grazing pasture contaminated by a single event can be described by (44):

$$C_M(t) = I_C(0) \sum_{i=1}^{n} A_i \frac{e^{-\lambda_{MEi}t} - e^{-\lambda_p t}}{\lambda_p - \lambda_{MEi}}$$

where $C_M(t)$ = concentration of radionuclide in milk at time t (μCi/l)

$I_C(0)$ = initial rate of ingestion of radionuclide by the cow (μCi/d)

A_i = coefficient of i^{th} exponential term, which describes the secretion in milk (liter^{-1})

λ_{MEi} = effective elimination rate of the i^{th} milk component (d^{-1})

$\lambda_{MEi} = \lambda_R + \lambda_{MBi}$

λ_R = radiological decay constant (d^{-1})

λ_{MBi} = biological elimination rate of i^{th} milk component (d^{-1})

λ_p = effective rate of removal of the nuclide from pasture (d^{-1}), and

$\lambda_p = \lambda_R + \lambda_W$, where λ_W is removal rate for a stable element from pasture (d^{-1}), and

t = time of milk secretion (d).

The total activity ingested by a person who drinks this milk can now be determined by integration:

$$\int_0^\infty I\,dt = \int_0^\infty J C_M(t) dt = \int_0^\infty J I_C(0)$$

where I = rate of ingestion of the radionuclide by a person (μCi/d) and

J = rate of consumption of milk (liter/d).

The solution for the total activity ingested by man is:

$$\int_0^\infty I\,dt = \frac{J I_C(0) f_M}{\lambda_p}$$

where f_M = transfer coefficient; i.e., the fraction of daily intake by cow that is secreted per liter of milk at equilibrium (d/liter)

$$f_M = \sum_{i=1}^{n} \frac{A_i}{\lambda_{MBi}}.$$

Ng et al. have conducted a comprehensive review of the literature relevant to the determination of transfer coefficients for both stable elements and radionuclides (44). These data are summarized in UCRL-51939, which also include values of the normalized coefficients, A_i, the biological half-life T_{MBi} (related to λ_{MBi}) for selected elements and values of f_M for all stable elements. This information is then used with the radioactive half-life to calculate values of f_M for specific radionuclides of interest (Table B-1 of Ng et al. (44)).

An examination of the logistics of the forage-cow-milk-man pathway shows that there is generally a delay time between the production of milk by the cow and its consumption by the general public. Therefore, it is appropriate to introduce a factor, S, to account for radioactive decay between production and consumption, where

$S = e^{-\lambda t}$ where λ = the decay constant for a given radionuclide (d^{-1})
and t = the delay between production and consumption (d).

Since the delay time for fresh whole milk is assumed to be 3 days only I-131 of the radionuclides of interest here has a sufficiently short half-life to result in a value of S significantly smaller than one. Thus, for I-131:

$S(I\text{-}131) = e^{-0.086 \times 3}$
$S(I\text{-}131) = 0.772$

Therefore, the total activity ingested as calculated by the above formula ($\int I dt$) must be multiplied by 0.772 in the case of iodine-131.

2.2.2 Total Intake

The calculated values of integrated activity ingested per $\mu Ci/m^2$ deposition from Appendix B of UCRL-51939 will be accepted as the basis for deriving the response levels equivalent to the PAG (44). The values in Appendix B are based on these values of parameters:

(1) $I_C(0)$, initial rate of intake by cow

UAF = "utilized area factor" (93)

UAF = 45 m^2/d

Initial Retention on Forage = 0.5 fraction

Initial Deposition = 1 $\mu Ci/m^2$

thus $I_C(0) = 22.5$ $\mu Ci/d$,

(2) J = 1 liter/day consumption of milk, and

(3) Half-residence time on forage is 14 days.

The UAF of 45 m^2/d assumes a forage consumption by the cow of 11.25 kg/d dry weight or 56 kg/d wet weight based on a forage yield of 0.25 kg/m^2 (dry weight) (45).

The values of the total intake per unit deposition (µCi µCi m^2) for a 1-liter per day milk intake from Ng et al., are given in Table 7, line 1 (44).

Table 7. Derivation of response levels equivalent to 1 rem dose commitment to critical organ

	I - 131		Cs - 134		Cs - 137		Sr - 90		Sr - 89	
1. Pathway intake factory (42) (uCi/uCi/m^2 per l/d)	1.34		3.12		3.22		.636		.46	
	Infant	Adult	Infant	Adult	Infant	Adult	Infant	Adult	Infant	Adult
2. Dose conversion factor (rem/uCi ingested)	16	1.5	.118	.068	.071	.061	2.49	.70	.194	.012
3. Intake per rem ($\frac{1}{\text{line 2}}$) (uCi intake per rem)	.063	.67	8.5	14.7	14.1	16.4	.40	1.43	5.2	83
4. Specific intake factor (line 1 x 0.7 or 0.55) - (uCi per u Ci/m^2	.72[a]	.37[a]	2.18	1.72	2.25	1.77	.45	.35	.322	.253
5. Initial surface deposition ($\frac{\text{line 3}}{\text{line 4}}$) (uCi/$m^2$ per rem)	.086	1.17	3.9	8.6	6.3	9.3	.90	4.1	16	329
6. Peak concentration factor (see text) (uCi/l per u Ci/m^2)	0.116		0.078		0.078		0.0196		0.018	
7. Peak milk concentration (line 5 x line 6) (uCi/l per rem)	.0100	.136	.303	.67	.49	.72	.0177	.080	.288	5.9
8. Initial grass concentration (line 5 x 0.4) (uCi/kg per rem)	.0345	.47	1.55	3.43	2.50	3.70	.361	1.63	6.40	132

[a]Corrected for decay during distribution (3 days factor - .772)

2.3 DIETARY INTAKE

Infant less than 1 year old - For the purposes of these recommendations, the dietary intake of milk is estimated to be 0.7 liters per day for a newborn infant.

Based on the average intake up to and including 1 year of age, the daily intake of milk for an infant less than 1 year of age is 0.7 liters (46). An additional 300 g of food may also be assumed to be ingested by an infant less than 1 year of age (based on intake of 6 month-old infants - Kahn, B. (47)).

Adult - Based on U.S. Department of Agriculture Household Food Consumption Survey 1965-1966, the average consumption for the general population is given in Table 8. The dietary intake of milk is taken to be 0.55 liters per day for the adult.

In addition to water ingested in food and drink, an estimated 150 ml of tap water is also ingested each day (46) for a total daily food intake of 2.2 kg.

Table 8. Average consumption for the general population

Food	Average consumption for the general population g/day	% of total diet
Milk, cream, cheese, ice cream[a]	567.5	27.2
Fats, oils	54.5	2.6
Flour, cereal	90.8	4.3
Bakery products	149.8	7.2
Meat	217.9	10.4
Poultry	54.5	2.6
Fish and shellfish	22.7	1.1
Eggs	54.5	2.6
Sugar, syrups, honey, molasses, etc.	72.6	3.5
Potatoes, sweet potatoes	104.4	5.0
Vegetables (excluding potatoes) fresh	145.3	7.0
Vegetables canned, frozen, dried	77.2	3.7
Vegetables juice (single strength)	9.1	0.4
Fruit, fresh	163.4	7.8
Fruit canned, frozen, dried	36.3	1.7
Fruit juice (single strength)	45.4	2.2
Other beverages (soft drinks, coffee, alcoholic bvgs.)	177.1	8.5
Soup and gravies (mostly condensed)	36.3	1.7
Nuts and peanut butter	9.1	0.4
Total	2088.1	99.9

[a]Expressed as calcium equivalent; that is, the quantity of whole fluid milk to which dairy products are equivalent in calcium content.

(From the U.S. Department of Agriculture Household Food Consumption Survey, 1965-1966)

2.4 CALCULATIONS

The calculation of the derived response levels equivalent to 1 rem dose commitment to critical organ for the grass-cow-milk-man pathway is summarized in Table 7. An explanation of Table 7 and the calculations follow:

<u>Line 1</u> Pathway Intake Factor is the total intake (μCi) for a 1 liter per day milk ingestion per 1 μCi/m^2 of initial area deposition (44).

<u>Line 2</u> Dose Conversion Factor is the dose commitment in rem/μCi ingested. See section 1 for summary.

<u>Line 3</u> Intake per rem is intake in μCi to yield a 1 rem organ dose.

<u>COMPUTATION</u> - The reciprocal of line 2.

<u>Line 4</u> Specific Intake Factor is the product of the Pathway Intake Factors (line 1) and the milk ingestion rate of 0.7 l/day infant or 0.55 l/day adult. In the case of I-131, a factor of 0.772 is included to adjust for 3 day's decay between production and consumption.

<u>COMPUTATION</u> - Line 1 x (1 or .772) x (0.7 or 0.55)

<u>Line 5</u> Initial Surface Deposition is initial area deposition of a specific radionuclide in μCi/m^2 which gives a 1-rem dose commitment.

COMPUTATION - Line 3 divided by Line 4

Line 6 Peak Concentration Factor is the peak maximum concentration in milk ($\mu Ci/l$) from an initial area deposition of 1 $\mu Ci/m^2$. Summary in Section 2.2.3 per model of Ng et al. (44).

Line 7 Peak Milk Concentration is the maximum milk concentration ($\mu Ci/l$) that yields a dose commitment of 1 rem from continuous ingestion of the contaminated milk supply.

COMPUTATION - Line 5 x Line 6

Line 8 Initial Grass Concentration is the activity concentration ($\mu Ci/kg$) on grass (edible forage) that results from the Initial Surface Deposition giving a 1-rem dose commitment.

Retention fraction on forage - 0.5

Forage yield - 1.25 kg/m^2 (wet weight)

COMPUTATION - Line 5 x $\frac{0.5}{1.25\ kgm^2}$

The derived response levels that correspond to the Preventive PAG (1.5 rem thyroid: 0.5 rem whole body and bone marrow) and the Emergency PAG (15 rem thyroid; 5 rem whole body and bone marrow) are given in Tables 9 and 10.

Table 9. Derived response levels for grass-cow-milk pathway equivalent to Preventive PAG dose commitment of 1.5 rem thyroid, 0.5 whole body or red bone marrow to infant[1]

Response levels for Preventive PAG	I-131[2]	Cs-134[3]	Cs-137[3]	Sr-90	Sr-89
Initial activity area deposition ($\mu Ci/m^2$)	0.13	2	3	0.5	8
Forage concentration[4] ($\mu Ci/kg$)	0.05	0.8	1.3	0.18	3
Peak milk acitivity ($\mu Ci/l$)	0.015	0.15	0.24	0.009	0.14
Total intake (μCi)	0.09	4	7	0.2	2.6

[1]Newborn infant includes fetus (pregnant women) as critical segment of population for iodine-131. For other radionuclides, "infant" refers to child less than 1 year of age.
[2]From fallout, iodine-131 is the only radioiodine of significance with respect to milk contamination beyond the first day. In case of a reactor accident, the cumulative intake of iodine-133 via milk is about 2 percent of iodine-131, assuming equivalent deposition.
[3]Intake of cesium via the meat-man pathway for adult may exceed that of the milk pathway; therefore, such levels in milk should cause surveillance and protective actions for meat, as appropriate. If both Cs-134 and Cs-137 are equally present, as might be expected in reactor accidents, the response levels should be reduced by a factor of 2.
[4]Fresh weight.

Table 10. Derived response levels for grass-cow-milk pathway equivalent to emergency PAG dose commitment of 15 rem thyroid, 5 rem whole body or red bone marrow

Response levels for emergency PAG	I - 131[3]		Cs - 134[5]		Cs - 137[5]		Sr - 90		Sr - 89	
	Infant[1]	Adult	Infant[2]	Adult	Infant[2]	Adult	Infant[2]	Adult	Infant[2]	Adult
Initial activity area deposition ($\mu Ci/m^2$)	1.3	18	20	40	30	50	5	20	80	1600
Forage concentration ($\mu Ci/kg$)[4]	0.5	7	8	17	13	19	1.8	8	30	700
Peak milk activity ($\mu Ci/\ell$)	0.15	2	1.5	3	2.4	4	0.09	0.4	1.4	30
Total intake (μCi)	0.9	10	40	70	70	80	2	7	26	400

[1]Newborn infant includes fetus (pregnant women) as critical segment of population for iodine-131.
[2]"Infant" refers to child less than 1 year of age.
[3]From fallout, iodine-131 is the only radioiodine of significance with respect to milk contamination beyond first day. In case of a reactor accident, the cumulative intake of iodine-133 via milk is about 2 percent of iodine-131 assuming equivalent deposition.
[4]Fresh weight.
[5]Intake of cesium via the meat-man pathway for adult may exceed that of the milk pathway: therefore, such levels in milk should cause surveillance and protective actions for meat, as appropriate. If both Cs-134 and Cs-137 are equally present as might be expected for reactor accidents, the response levels should be reduced by a factor of 2.

CHAPTER 3. METHODS OF ANALYSES FOR RADIONUCLIDE DETERMINATION

3.1 INTRODUCTION

The measurement of radionuclides in food can be accomplished by either laboratory methods or field methods using portable survey instrumentation. Unfortunately, neither method of analysis was developed expressly for the purpose of implementing protective actions. In order to provide instrumentation guidance to the States, the Federal Radiological Preparedness Coordinating Committee formed a Task Force on offsite instrumentation. A draft report on instrumentation analysis methods for the milk pathway is now undergoing review by the Task Force and a second report on other food is under development. This effort is being fostered by past and current contracts of Nuclear Regulatory Commission (NRC) and Federal Emergency Management Agency (FEMA) with Brookhaven National Laboratory and Idaho National Engineering Laboratory.

The material and methods are given as interim guidance until these more definitive reports are available. It should be noted that laboratory methods of Chapter 3.2, below, were developed for environmental monitoring purposes and are more sensitive than required for protective actions implementation. And, conversely, the field methods of Chapter 3.3 are generally inadequate for the purpose of implementing action at the Preventive PAG level. Analysis methods should be able to measure radionuclide concentrations in food lower by a factor of 10 than the derived levels for Preventive PAG. Thus, it may be necessary to use a combination of laboratory and field methods in implementing and ceasing protective actions.

3.2 DETERMINATIONS OF RADIONUCLIDE CONCENTRATIONS BY SENSITIVE LABORATORY METHODS

Many compendia of methods of analysis of environmental samples are available. The EML Procedure Manual recommended is noted for its up-to-date methodologies, which continuously undergo revision and improvement (48). Analysis need not be limited to reference 48 but laboratory analysis of food should provide limits of detection as listed below, which are lower than required for protective action:

Radionuclide	Limit of detection* pCi/liter or kg
I-131	10
Cs-137	10
Sr-90	1
Sr-89	5

*Concentration detectable at the 95-percent confidence level.

A source of more rapid methods of analysis of radionuclides in milk, applicable to these recommendations is described by B. Kahn et al. (49). The methods for gamma radionuclide analysis (applicable to I-131 and Cs-137 presented in this reference are also applicable to pasture. The gamma scan determinations of I-131 and Cs-137 in milk (or water) can generally be accomplished within 2 or 3 hours. For samples measuring in the 0-100 pCi/liter range, the error at the 95-percent confidence level (2 sigma) is 5 to 10 pCi/liter. For samples measuring greater than 100 pCi/liter the error is 5 to 10 percent.

Radiostrontium procedures permit analyses of several samples simultaneously in 5 hours of laboratory bench time, plus 1-2 weeks for ingrowth of yttrium daughters. If the laboratory is set up for routine analysis of these radionuclides recovery in tracer studies is 80 ± 5 percent.

An ion exchange field method for determination of I-131 in milk, which uses gamma spectroscopy after sample collection, has also been described (50). The main advantage of this method is that it permits a large number of samples to be processed in the field or shipped and analyzed in a central laboratory.

For analysis of samples other than milk the HASL reference (48) is recommended.

3.3 DETERMINATIONS OF RADIONUCLIDE CONCENTRATIONS BY FIELD METHODS

3.3.1 Ground Contamination (Beta Radiation)

The conversion of ground survey readings to contamination levels can be accomplished by using the following equations and factors (assuming a metal tube wall thickness (steel) of 30 mg/cm^2):

1. Use a G-M survey meter calibrated to yield 3,000 counts/min per 1 mR/h of Ra γ.

2. Hold the probe not more than 5 cm above the ground with the beta shield open.

3. Assure that 100 counts/min can be detected above a normal 50 to 100 counts/min background.

4. Take readings in open terrain; that is, not in close proximity to heavy vegetation, cover, or buildings.

For determinations of ground deposition:

$D = R \times F$

where, D = ground deposition (in μCi/m^2),

R = G-M reading (in units of 10^2 counts/min) (background corrected), and

F = factor given in Table 11.

For determination of concentrations in vegetation:

$C = (D \times f)\ d$

where, C = concentration (in μCi/kg),

D = ground deposition (in μCi/m^2),

f = fraction of deposited nuclide in the vegetation, and

d = density of vegetation cover (in kg/m^2).

Generally, f ranges from 0.1 to 1 and is usually taken to be 0.25 for I-131 in the United Kingdom, and 0.5 in the United States.

Data of a similar nature may also be found in "Emergency Radiological Plans and Procedures," in the Chapter (Item 04.3.4) on "Conversion of Survey Meters to Concentration," (51).

Table 11. Ground surface contamination levels[a] of various nuclides required to yield 100 counts/min (net) on a G-M meter (open window)

Nuclide	F (μCi/m^2 per 100 counts/min)
Zr-95 + Nb-95	6
Ce-141	2
I-131, Ru-103m, mixed Ru-Rh (100-d old)[b]	1
Co-60, Sr-89, Y-90, Y-91, Cs-137, Ba-140, La-140 Ce-144 + Pr-144, Ru-105 + Rh-106, mixed radioiodines (1-h to 1-week old), mixed fission-products (100-d old)	0.3

[a]Level varies with background readings, ground roughness, and vegetation cover.
[b]Age refers to time since irradiation of the fuel from which the Fission Products were released.

3.3.2 Herbage

A field method for estimation of radionuclide contamination at the response levels equivalent to the Emergency PAG for pasture (forage) which has been suggested by International Atomic Energy Agency (52) is as follows:

1. Obtain enough vegetation to fill a 30 cm x 40 cm plastic bag approximately hal full. This is about one-third of a kilogram (Note: the vegetation cover should be obtained from at least 1 m^2 of ground. The vegetation should be cut at approximately 1 to 2 cm from the ground and should not be contaminated in the process by soil).
2. Note the area represented by this quantity of material.
3. Compress the air from the bag and seal.
4. Transfer the sample to a low background area.
5. Flatten bag and lay probe of a portable G-M survey meter on the center of the bag.
6. Wrap bag around probe and note reading (window open and background corrected).
7. Calculate the contamination from the following equation:

   $C = R/k$

   where, C = vegetation concentration (in μCi/kg),

   R = G-M reading (in units of 10^2 counts/min) (background corrected), and

   k = 10^2 counts/min per μCi/kg as given in Table 12.
8. Convert μCi/kg to μCi/m^2 on the basis of the area represented by the sample.

9. The limiting radionuclide (i.e., having the lowest recommended PAG relative to its deposition on pasture) is iodine-131. According to Table 12, this radionuclide is detected with the lowest efficiency. Thus, if the operator assumes this radionuclide to be exclusively present in the pasture the most conservative estimates relative to the Emergency PAG would be reached.

Table 12. Typical G-M survey-meter readings probe inserted in the center of a large sample of vegetation

Nuclide	k (10^4 x counts/min per µCi/kg)
Sr-89, Sr-90 + Y-90	20
Ru-106 + Rh-106	50
Ba-140 + La-140	10
I-131, Cs-137	4

3.3.3 Milk

The experimental data for field determination of radionuclides in milk are limited to determination of iodine-131, and the details are rather sketchy. What material is available is abstracted below.

1. Although no data are available for field determination of radionuclides other than iodine-131 in milk, Table 13 presents experimental information obtained in water (52,53). To the extent milk is more self-shielding than water, the following data is presented as a guide rather than a means of analysis.

Table 13. G-M survey-meter open window readings (counts/min per µCi/liter) (probe immersed in contaminated water)

Nuclide	Size of sample container: 1 liter	5 liters	>10 liters
	Counts/min per µCi/l		
Sr-89	2000	2000	2000
Sr-90 + Y-90	2000	2000	2000
Ru-106 + Rh-105	6000	8000	10000
I-131	500	800	1000
Cs-137	400	600	800
Ba-140 + La-140	1000	1500	2000

2. Method of Kearney (54):

a. Instrument: CDV-700 Model No. 6B with beta window closed at all times.

b. Geometry: See Figure 1.

c. The count rate is determined by ear. If the counts per minute recorded by ear are more than 60 to 70 cpm, then the milk should be diluted with "pure" water so as to produce a sample having a 50%-50%, 25%-75% or other concentration low enough to produce counts per minute somewhat less than 60 to 70 cpm.

d. Background: The experimental conditions duplicated "sky shine" from a trans-Pacific transfer of fallout. If the exposure around the test hole was 0.75 mR/hr a couple of feet above the hole, the sky shine increased cpm recorded by the probe shielded within the test hole by 3 cpm. Background, on the average, was measured (in "pure" water) to be 12 to 15 cpm (in an uncontaminated environment). Thus, total background counts (with sky shine) is on the average around 19 cpm.

e. From Figure 2, the net counts per minute equivalent to the response level for the Emergency PAG applicable for milk (infant as critical segment of population) is approximately 20.

3. Method of C. Distenfeld and J. Klemish (55).

   a. Instruments:

      i. CDV 700 instrument turned to 10 X scale and calibration adjustment turned to require the meter to indicate 2 mR/hr. (NB: Discrepencies were noted between data from scale and pulse counting with an oscilloscope).

      ii. Modified CDV 700 M with factory calibration. Good agreement between scale reading and electronic check.

   b. Geometry: The basic container was a 5-gallon heavy-wall polyethylene "Jerri" type measuring 12x9x10 inches. A 2-inch O.D. blind tube was installed to allow the G-M probe to sample the center of the container.

   c. Counts were taken inside and at the top (external) to the container.

   d. Background was determined in a water filled plastic container (15x25x20 cm) about 7 meters from the sample vessel.

   e. Net counts per minute equivalent to the response levels for the Emergency PAG for milk are summarized in Table 14.

4. The International Atomic Energy Agency reports on a series of experiments (53). The data are duplicated in Table 15.

5. A forthcoming report of the Federal Interagency Task Force on Offsite Emergency Instrumentation for Nuclear Incidents to be published by FEMA is "Monitoring and Measurements of Radionuclides to Determine Dose Commitment in the Milk Pathway." A subsequent report by the Task Force will address field methods and monitoring strategies for other food pathways.

6. The relative sensitivity of the various techniques is summarized in Table 16.

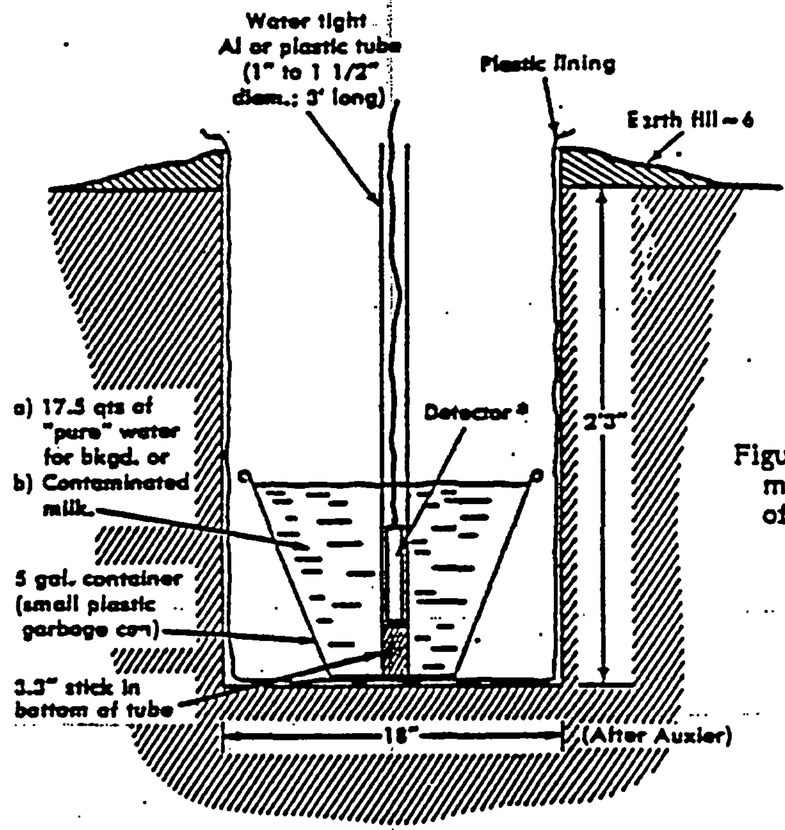
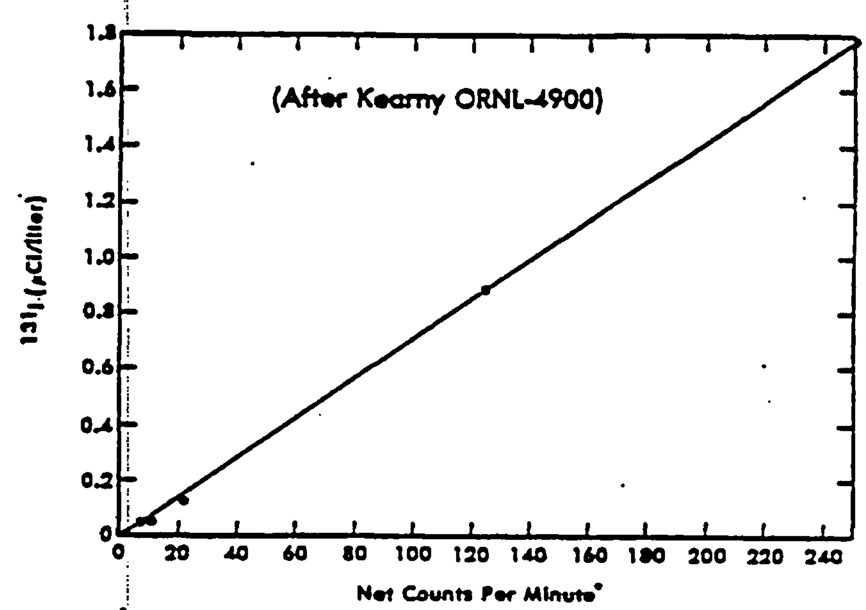


Figure 1. Geometry for making measurements within a volume of liquid.

*CDV-700, Model No. 68 covered with small plastic bag taped to cable of probe to further protect against dampness.



*Net counts per minute determined by ear. Net counts per minute = grosscounts per minute — background in pure water.

Figure 2. Net counts per minute*

Table 14. Net counts per minute equivalent to the response levels for the Emergency PAG for milk

Instrument Probe Position	Approximate net counts per minute: CDV-700 Inside	CDV-700 Outside	CDV-700M Inside	CDV-700M Outside
Response level for Emergency PAG				
(Milk-Infant)	20[a]	8[a]	220	100
(Milk-Adult)	260	110	2,900	1,300

[a]At or below background cpm - precision not adequate.

Table 15. Survey-meter readings versus concentration of I-131 in a 40-liter milk can

Meter used	μCi I-131/liter milk	Net counts per minute Inside can	Net counts per minute Outside can
Al-walled GM probe	0.9	1,500	300
	0.5	500	200
	0.1	100	50
	0.05	50	50
	Background	50	50
Mica-window GM probe	0.9	600	250
	0.5	400	100
	0.1	100	50
	0.05	50	50
	Background	50	50
α,β,γ scintillation survey-meter[a]	0.9	5,500	3,000
	0.5	3,000	1,500
	0.1	600	300
	0.05	250	150
	Background	100	100
Transportable single-channel analyzer system	0.10	1,200	-
	0.05	650	-
	0.01	140	-
	0.005	80	-
	Background	30	-

[a]Crystal is 3-mm thick disc of "Bioplastic" scintillator sprayed with 10 mg/cm^2 of ZnS. Effective area is 6.4 cm^2.

Table 16. Comparison of methodologies

	cpm per μCi liter^{-1} Inside	Outside
IAEA - 1974 (52)	1,000	-
Kearney - ORNL (54)	140	-
Distenfeld and Klemish - Brookhaven (55)		
CDV-700	134	56
CDV-700M	1,450	660
IAEA - 1966 (53)		
Al Walled GM Probe	1,100	350
Mica-Window GM Probe	700	250
α,β,γ Scintillation Survey Meter	6,000	3,300
Transportable Single Channel Analyser	12,000	-

CHAPTER 4. PROTECTIVE ACTIONS

The National Advisory Committee on Radiation (56) (NACOR) made the following recommendation that applies to action taken to reduce potential exposure following the accidental release of radioactivity:

"A countermeasure, useful to public health, must fulfill a number of requirements. First, it must be <u>effective</u>; that is, it must substantially reduce population exposures below those which would prevail if the counter measure were not used. Second, it must be <u>safe</u>; i.e., the health risks associated with its use must be considerably less than those of the contaminant at the level at which the countermeasure is applied. Third, it must be <u>practical</u>. The logistics of its application must be well worked out; its costs must be reasonable; and all legal problems associated with its use must be resolved. Next, <u>responsibility and authority</u> for its application must be well identified. There must be no indecision due to jurisdictional and misunderstandings between health and other agencies concerned with radiation control. Finally, careful attention must be given to such <u>additional considerations</u> as its impact on the public, industry, agriculture, and government."

An action, in order to be useful must be effective, safe, and practical. An action may be applied at the source in an attempt to control the release of radioactivity from the source; or, the action may be applied at the beginning of the food chain (soil, vegetation, or cattle), to the immediate vector prior to ingestion by man (milk or food), or to the population itself. For the most part these recommendations suggest protective actions to milk, human and animal foods, or soil and this chapter is limited to actions concerning these media. Further recommendations by NACOR (56) extend the discussion of protective measures to public health actions directly affecting the exposed population. For details of agriculture actions, several Department of Agriculture reports are available that deal with specific actions for crops and soil (57,58,59).

Potential actions relative to the pasture-milk-man pathway are summarized in Table 17. For this pathway, only four countermeasures are rated as effective, safe, and practical (a somewhat arbitrary scale of judgment was used). Of the four, one has distinctive disadvantages. Although removal of radionuclides from milk has been shown to be practical no facilities for doing this exist. Another, diverting fresh milk to processed milk products, freezing and/or storage, is effective only for short-lived radionuclides. Thus, changing dairy cattle to an alternate source of uncontaminated feed and condemnation of milk are the only two protective actions rated good for effectiveness, safety, and practicality.

Of course the other countermeasures should also be considered, but they appear less promising.

Actions for fruits and vegetables are presented in Table 18 (60,61,62). Note that studies in which these products were contaminated under actual conditions with fallout (Studies 2 and 3) yielded a lower reduction in the radioactivity removed during preparation than was the case in an investigation (Study 1) in which radionuclides were sprayed on the food. Depending on the food, reductions between 20 and 60 percent in strontium-90 contaminations are possible by ordinary home preparation or by food canning processes (60).

Milled grains contain only a small portion of the total radioactive contamination of the whole grain; removal of bran from wheat and polishing of rice are effective methods of reducing contaminating fallout (58). Todd indicated average concentration of Sr-90 (pCi/kg) in wheat of wheat berry (22%), wheat bran (68%), and only 4.4% in flour. In rice the corresponding values are: whole grain (4.9%), and milled rice (0.7%) (58).

Although these recommendations are intended for implementation within hours or days after an emergency, long-term actions applicable to soil are shown for information purposes only in Table 19. Alternatives to decontamination and soil management should be considered, especially if the radioactive material is widespread, because great effort is required for effective treatment of contaminated land.

The concept of Protective Action assumes that the actions implemented will continue for a sufficient period of time to avoid most of the projected dose. The concentration of radioactivity in a given food will decrease because of radioactive decay and weathering as a function of time after the incident. Thus, as discussed in Chapter 5 of this report, actions that have a positive cost-benefit ratio at the time of initial contamination or maximum concentration may not have a positive cost-benefit ratio at later times. Therefore, dependent on the particular food and food pathway, it may be appropriate to implement a series of protective actions until the concentrations in the food have essentially reached background levels.

As an example of the implementation of protective actions, consider the case where an incident contaminates the pasture-cow-milk-man pathway with a projected dose of 2-3 times the Emergency PAG due to Iodine-131. *In such a situation* these protective actions may be considered appropriate:

1. Immediately remove cows from pasture and place them on stored feed in order to prevent as much Iodine-131 as is possible from entering the milk;

2. Condemn any milk that exceeds the Emergency PAG response at the farm or milk plant receiving station;

3. Divert milk contaminated at levels below the Emergency PAG to milk products; and

4. Since the supply of stored feed may be limited and the costs of this protective action greater than diversion to milk products, the use of stored feed may be the first action to cease; this should not be done, however, until the concentration of I-131 has dropped below the Emergency PAG and preferably is approaching the Preventive PAG.

5. The diversion of fresh milk to milk products must continue until most of the projected dose has been avoided; this action might be ceased when the cost-effectiveness point is reached or the concentration of Iodine-131 approaches the background levels.

This discussion assumes that there is an adequate supply of whole milk from noncontaminated sources, that there is an available manufacturing capacity to handle the diverted milk, and that the Iodine-131 is the only radionuclide involved. In an actual situation these conditions may not be present and other factors may affect the practicality of proposed protective actions. The agency responsible for emergency action must identify and evaluate those factors that affect the practicality of protective action, and thus develop a response plan (with tentative protective action) that is responsive to local conditions and capabilities.

Table 17. Actions applicable to the pasture-milk-man pathway (compiled from references 57 and 59)

Action	Radionuclide(s) for which protective action is applicable	Effectiveness	Safety	Practicality (effort required)	
Applicable to cattle					
Provide alternate source of uncontaminated animal feed	^{131}I, ^{89}Sr, ^{90}Sr, ^{137}Cs	(+)[a]	(+)	(+)	Good
Add stable iodine to cattle ration	^{131}I	Marginal[b]	Some hazard	(+)	
Add stable calcium to cattle ration	^{89}Sr, ^{90}Sr	Marginal	Some hazard	(+)	
Add binders to cattle ration	^{137}Cs, ^{89}Sr, ^{90}Sr	Marginal	Questionable	(+)	
Substitute sources of uncontaminated water	^{137}Cs, ^{89}Sr, ^{90}Sr	(+)	(+)	(+)[c]	
Applicable to milk					
Condemnation of milk	^{131}I, ^{89}Sr, ^{90}Sr, ^{137}Cs	(+)	(+)	(+)[d]	Good
Divert fresh milk to processed milk products	^{131}I, ^{89}Sr	(+)	(+)	(+)	Good
Process fresh - store	^{89}Sr, ^{137}Sr	Marginal	Questionable	(+)	
Process fresh - store	^{131}I	(+)	(+)	(+)	Good
Remove radionuclides from milk	^{131}I, ^{89}Sr, ^{90}Sr, ^{137}Cs	(+)	(+)	(+)[e]	Good

[a] (+): 90% effective
[b] Marginal: less than 90% effective
[c] Depends on availability
[d] Somewhat dependent on volume
[e] No processing plant presently available

Table 18. Percent reduction in radioactive contamination of fruits and vegetables by processing

	Study 1 (60) Normal food preparation for freezing, canning or dehydration				Study 2 (61)	Study 3 (62)
	External ^{90}Sr	Contamination[a] ^{137}Cs	Internal ^{90}Sr	Contamination[a] ^{137}Cs	Canning ^{90}Sr	Home preparation ^{90}Sr
Spinach	92	95	64	88	22	-
Snap beans	-	-	-	-	62	-
Carrots	-	-	-	-	19	19
Tomatoes	-	-	65	-	21	28
Broccoli	94	92	72	89	-	-
Peaches	~100	~100	~100	~100	50	-
Onions	-	-	-	-	-	37
Potatoes	-	-	-	-	-	24
Cabbage	-	-	-	-	-	53
Green beans	-	-	-	-	-	36

[a]Contamination on surface is referred to as external contamination. Internal contamination is contamination of fleshy portion of product from surface deposition of radionuclide.

Table 19. Actions applicable to soil (compiled from references 57 and 59)

Action	Radionuclide(s) for which protective action is applicable	Effectiveness[a]	Safety	Practicality (effort required)[b]
Applicable to soil				
Soil management—minimum tillage:	^{90}Sr	Poor to fair	Not applicable	Good
deep plowing with root inhibition	^{90}Sr	Good to fair	"	Poor
irrigation & leaching	^{90}Sr	Poor	"	Good
liming & fertilizing	^{90}Sr	Poor to fair	"	Good
Removing contaminated surface crops	^{90}Sr	Most poor	"	Poor to fair
Removal of soil surface contamination:				
warm weather with vegetation cover	^{90}Sr	Good to fair	"	Poor
cold weather no cover	^{90}Sr	Good to poor	"	Good to poor

[a]Rating for reducing strontium-90
Good - 95% reduction
Fair - 75-95% reduction
Poor - 75% reduction

[b]Rating for effort required
Good - not significantly more than normal field practice
Fair - extra equipment or labor required
Poor - very great requirement of equipment, materials, and labor

CHAPTER 5. COST CONSIDERATIONS

5.1 COST/BENEFIT ANALYSIS

5.1.1 Introduction

The general expectation is that protective action taken in the event of a nuclear incident will result in a net societal benefit considering the cost of the action and the corresponding avoided dose. These cost assessments, including cost/benefit analysis, have not been used to set the numerical value of the PAG's but rather to evaluate the feasibility of specific protective actions.

At least two basically different approaches can be used to assess the cost/benefit ratio of protective actions for the milk pathway. One approach would be to assume a protective action scenario (maximum milk concentration and length of time of protective action) and to calculate the total cost of the action and the benefit because of the avoided dose. The ratio of the cost/benefit can then be used to scale the maximum milk concentration to that concentration that yields equal costs and benefits. The problem with this approach is that positive net benefits when milk concentration of radioactivity is high are used to offset the negative net benefits during the later times of action.

This deficiency leads to the second approach of calculating the milk concentration on a per liter basis where the cost of the protective action equals the benefit because of the dose avoided. This approach will be used here since it gives a clearer picture of the identified costs and benefits. The specific concentration at which costs equals benefits should not however be viewed as the appropriate level for taking protective action. The philosophy of protective action is to take action to avoid most of the projected doses. Further, the simple analysis considered here treats only the direct cost of protective actions and ignores the administrative costs of starting, monitoring, and ceasing action, and other related social and economic impacts.

Although the PAG recommendations provide that protective actions be taken on the basis of projected dose to the infant, cost/benefit analysis must consider the cost impact on the milk supply and the benefit on a whole population basis. Accordingly the benefit realized from avoiding the dose associated with a given level of milk contamination C (μCi/l) must be summed over the age groups having different Intakes (I) and Dose Factors (DF) and is:

$$\text{Benefit} = C\ (\mu\text{Ci/l}) \times \text{Value}\ (\$/\text{rem})\ \Sigma \text{Ii}(\text{l/d}).\text{DFi}(\text{rem}/\mu\text{Ci}).$$

The total cost of the protective actions, which must also be summed over all the age intake groups is:

$$\text{Cost} = \text{PA COST} \times \Sigma \text{Ii}$$

Costs are in 1980 - 1981 dollars. These equations can then be solved for the concentration (C) at which cost = benefit giving:

$$C\ (C{=}B) = \frac{\text{PA COST}\ \Sigma \text{Ii}}{\text{Value}\ (\$/\text{rem})\ \Sigma(\text{Ii} \cdot \text{DF}_i)}$$

5.1.2 Benefit of Avoided Dose

In situations in which there has been an uncontrolled release of radioactivity to the environment, both the health savings and cost of a protective action can be expressed in terms of dollar values. This does not exclude the probability that undesirable features will result from an action that is difficult to evaluate in economic terms.

A previous cost-benefit analysis described the radioactive concentration of iodine-131 in milk at which it would be justifiable to initiate condemnation of milk (63). Following is a summary of the monetary benefit of radiation dose avoided using the approach suggested, with changes because of increased costs over time and new data on the relative incidence of various tumors.

The International Commission on Radiological Protection, (64) has endorsed the principle of expressing the detriment from radiation in monetary terms in order to facilitate simplified analysis of costs and benefits. This permits a direct comparison between the societal advantage gained in a reduction of the radiation dose and the cost of achieving this reduction. Cost-benefit analysis is the evaluation process by which one can determine the level at which, or above which, it would be justifiable to initiate the protective action because the health savings equaled or exceeded, the economic costs of the protective action. Certain factors, such as loss of public confidence in a food supply, are not considered; nor are economic factors because of hoarding and a shortage of supply considered. A similar treatment of the problem with almost the same result has been published (65). This type of exercise is useful prior to taking an action as one, and only one, of a series of inputs into decisionmaking.

The costs, and hence health savings to society, of 1 person-rem of whole-body dose (the product of a dose of 1 rem to the whole body and 1 person) has been estimated by various authors to be between $10 and $250 (66). The Nuclear Regulatory Commission (NRC) value for a cost-benefit analysis for augmented equipment for light-water reactors to reduce population dose, sets radiation costs at $1000 per person-rem (67).

Based on medical expenses in 1970, the total future cost of the consequences of all genetic damage of 1 person-rem (whole-body) was estimated by the BEIR Committee (2) to be between $12 and $120. These costs are in good agreement with estimates made by Arthur D. Little, Inc., for the Environmental Protection Agency, which calculated that in terms of 1973 dollars, 1 person-rem of radiation yielded a tangible cost of between $5 and $181 due to excess genetic disorders. A tangible cost of between $7 and $24 per person-rem was estimated to be the result of excess cancer in the same report. Therefore, the total health cost of a person-rem from these studies is between $12 and $205 (68).

Assuming that $200 is a reasonable estimate for the overall somatic health cost to society per person-rem whole body, the proportionate cost for individual organ doses must then be derived. For the purposes of assessing health cost, it is appropriate to use the relative incidence of cancer estimated to result from organ doses vs. whole body doses. From BEIR-III (3) (Table V-14 and V-17, and using an average of the male and female incidence) the thyroid contributes 20 percent of cancer and leukemia (red bone marrow doses) 11 percent of the total cancer incidence. Hence, the monetary costs per rem of radiation dose avoided are: to thyroid $40; and to red bone marrow $22.

5.1.3 Protective Action Costs

The direct cost of protective action will be assessed for (1) cost of stored feed, (2) condemnation at the farm (farm value), and (3) condemnation at the processing plant (retail value).

1. Cost of stored feed. For the participating herds (May 1, 1978 - April 30, 1979) the Dairy Herd Improvement Letter (69) reports a consumption of 12,600 lbs. of succulent

forage and 3,000 lbs. of dry forage, with a corresponding annual milk production of 14,129 lbs. (6200 liters). (The cows also consumed 5,800 lbs. of concentrates, which are not of concern here.) Taking 3 lbs. of succulent forage (silage) as equivalent to 1 lb. of hay, the annual hay equivalent consumption is 7,200 lbs (70). Thus, 1.16 lbs. of hay equivalent is consumed per liter of milk production. The 1980 average price received by farmers for all baled hay was $67.10 per ton or $0.0335 per lb. (71). The Protective Action (PA) cost of buying baled hay to replace pasture as the sole forage source is then $0.039 per liter.

2. Milk-farm value. The average price received by farmers for fluid-eligible milk, sold to plants and dealers in 1980, was $13.71 per hundred weight (monthly range $12.70 to $14.20) (71). The lower prices are received during the pasture season of April through August. For 44 liters per hundred weight, the farmer value of milk is $0.30 per liter.

3. Milk-retail value. Since it may be necessary to take protective action at some stage in the milk processing and distribution system it is appropriate to consider the retail value of milk. If condemnation of milk is taken at the receiving station or processing plant there will be additional costs above farm value associated with disposal. It is felt that retail price should represent an appropriate value. The average city retail price of fortified fresh whole milk sold in stores, January through October 1980 was $1.037 per 1/2 gallon (72). The monthly price increased from $1.015 in January to $1.067 in October, apparently because of inflation. Based on the average price the value of $0.56 per liter will be used.

5.1.4 Population Milk Intake and Dose

Table 20 summarizes the milk intake by population age groups and gives values of the age group intake factor Ii(l/d). The total intake by a population of 1000 is 281 l/d or an average individual daily intake of 0.28 l. The intake factor (Ii) is used with the dose factor DFi listed in Table 21 to calculate the dose factor summed over the whole population weighted by age per µCi/l of milk contamination.

5.1.5 Milk Concentration For Cost = Benefit

The above results are then used to calcultate the milk concentration at which the Protective Action (PA) costs equals the benefit from the dose avoided. The results are presented in Table 22. The cost = benefit concentration for use of stored feed in place of contaminated pasture is about 0.2 to 0.3 of the Preventive PAG for strontium and 0.01 to 0.02 for iodine and cesium. For condemnation, the cost = benefit concentrations based on farm value of milk have ratios of the Emergency PAG similar to those above. The cost = benefit concentrations based on retail value of milk are about a factor of 2 greater than those based on the milk's farm value. The fact that the cost = benefit concentrations are a significant percent of the PAG for strontium results largely because the value of the person-rem dose to red bone marrow is one-ninth that of whole-body doses while the PAG's are set at equal doses consistent with current regulations. Further the controlling PAG's are for the infant, while the cost/benefit reflects population averaged benefits.

Table 20. Population milk intake (Σ Ii)

Age group	Persons per 1000 population	Milk intake[a] (l/d)	Intake (Ii) by age group (l/d)
In utero	11	.4	4.4
0 < 1	14	.775	10.9
1 - 10	146	.470	68.6
11 - 20	196	.360	70.6
>20	633	.200	126.7
		ΣIi	281.2

[a]ICRP, 1974

Table 21. Population dose factors

Age group	Sr-89 DFI rem/μCi	Sr-89 Ii x DFI $\frac{rem}{\mu Ci} \cdot \frac{1}{d}$	Sr-90 DFI rem/μCi	Sr-90 Ii x DFI $\frac{rem}{\mu Ci} \cdot \frac{1}{d}$	Reference for DFI values[a]
In utero	.414	1.82	4.03	17.7	0 yr old
0 < 1	.194	2.12	2.49	27.2	0.5 yr-old
1 - 10	.0565	3.88	.929	63.8	Average
11 - 20	.0175	1.24	.82	57.9	Av 11 yr & adult
> 20	.012	1.52	.70	88.7	Adult
	ΣIiDFI	10.58		255.3	

[a]See Chapter 2.

Age group	Cs-134 DFI rem/μCi	Cs-134 Ii x DFI $\frac{rem}{\mu Ci} \cdot \frac{1}{d}$	Cs-137 DFI rem/μCi	Cs-137 Ii x DFI $\frac{rem}{\mu Ci} \cdot \frac{1}{d}$	Reference for DFI values[a]
In utero	.068	.3	.061	.27	Adult[b]
0 < 1	.118	1.29	.071	.77	Infant
1 - 10	.093	6.39	.066	4.53	Av. infant & adult
11 - 20	.093	6.57	.066	4.66	"
> 20	.068	8.51	.061	7.73	Adult
	ΣIiDFI	23.06		17.966	

[a]See Chapter 2.
[b]No credit taken for reduced biological half-life in pregnant women.

Age group	I-131 DFI rem/μCi	I-131 Ii x DFI $\frac{rem}{\mu Ci} \cdot \frac{1}{d}$	Reference for DFI values[a]
In utero	.8	35	Max. estimate
0 < 1	16	174	Newborn
1 - 10	5.7	391	Average from smooth curve
11 - 20	2.1	148	15 yr old
> 20	1.5	190	Adult
	ΣIi DFI	938	

[a]See Chapter 2.

Table 22. Milk concentration at which cost = benefit
(Population basis - value of ΣIixDFi for 1000 persons)

		Sr-89	Sr-90	I-131	Cs-134	Cs-137
ΣIi x DFi $\frac{rem}{\mu Ci} \cdot \frac{1}{d}$		10.58	255	938	23.1	17.96
Value (\$/rem)		22	22	40	200	200
	PA cost	CONC. (Cost = Benefit) (μCi/l)				
Stored Feed	\$0.039	.047	.002	.0003	.0025	.003
Farm Milk	0.30	.36	.015	.0023	.018	.025
Retail Milk	0.56	.68	.028	.0042	.034	.044
		Peak CONC. (μCi/l)				
Preventive PAG		.14	.009	.015	.15	.24
Emergency PAG infant		1.4	.09	.15	1.5	2.4
Emergency PAG adult		30	.4	2.0	3.0	4.0

5.2 ECONOMIC IMPACT

The Emergency Planning Zone (EPZ) for the ingestion pathway has been set at 50 miles (73). The area impacted that requires protective action is the major factor influencing cost. Assessment of the economic impact will be considered for the case of contamination of the milk pathway in one 22.5° Sector out to a distance of 50 miles. Table 23 gives data on the annual sales of whole milk and total area of leading dairy States and selected States. The annual milk sales in Wisconsin of 3.52×10^5 lbs. per sq. mile exceeds that of any other State and will be used to assess the economic impact. There may, of course, be individual counties and areas surrounding nuclear power plants where milk production exceeds the Wisconsin State average. The Wisconsin average should, however, represent a maximum for most areas of the United States.

Table 23. Milk production of selected States
(Statistical Abstract of the U.S., 1978)

State	Whole milk sold (10^9 lbs/year)	Total area (mi^2)	Milk per unit area 10^5 lbs/mi^2
WI	20.5	56,154	3.52
VT	2.06	9,609	2.14
NY	9.92	49,576	2.00
PA	7.37	45,333	1.64
IA	4.07	56,290	1.38
CT	.595	5,009	1.19
MN	9.27	84,068	1.10
OH	4.43	41,222	1.08
MI	4.63	58,216	0.80
CA	11.53	158,693	0.73
MA	0.55	8,257	0.67
NJ	0.52	7,836	0.66

Another important factor in assessing the economic impact of protective actions in the milk pathway is the length of time that such actions will be necessary. During most of the year in northern parts of the U.S., cattle will already be on stored feed and there will be no additional costs for the stored feed protective action. For other situations and the Emergency PAG, the time over which protective actions will be necessary is a function of a number of parameters unique to each site and the causative accident. Thus, what are intended as conservative assumptions will be selected. The time behavior of I-131 on pasture grass is controlled by the 8-day radioactive half-life and the 14-day weathering half-life (yielding an effective half-life of about 5 days). Milk which contains I-131 at the Emergency PAG of 0.15 μCi/l will be reduced to the cost = benefit concentration (farm value) of 0.0023 μCi/l about 30 days later. Obviously in most cases of an atmospheric release, those areas closer to the release point will have higher levels of contamination and longer times of protective action. The NRC and EPA in the Planning Basis Report NUREG-0396 (NRC, 1978) assume that radiation doses from the airborne plume decrease according to the $r^{1.5}$ factor. Use of this factor for contamination of pasture results in milk concentrations at 2 miles that are about 100 times that at 50 miles. For an effective half-life of 5 days this would require an additional 30-35 days of protective action at 2 miles over that at 50 miles to yield the same milk levels upon ceasing action. Although these models cannot be assumed to be rigorously accurate in a specific accident situation, they do indicate that action might be required for 1 or 2 months.

NUREG-0396 notes that the dose from milk pathway is of the order of 300 times the thyroid dose from inhalation (74). Under this assumption (and above models), the food PAG's would be exceeded at hundreds of miles if protective action because of inhalation were required at 10 miles. It should be noted that the meteorological models that are empirically derived are not likely to be valid for such long distances. Further changes in wind direction and meteorological dispersion conditions may reduce the levels of pasture contamination and the downwind distance. For assessing ecomomic impact, contamination of a 22.5° Sector out to a distance of 50 miles will be arbitrarily assumed, even though actual contamination patterns are not likely to be similar.

Under these assumptions we then have:

Area (Circle - 50 mile radius) - 7850 mi^2

Area (22.5° Sector - 50 mile) - 491 mi^2

Milk Production - 3.52×10^5 lbs/mi^2 per year - 2.93×10^4 lbs/mi^2 per month

Production (22.5°/50 mi Sector) - 1.44×10^7 lbs./month

Cost of Stored Feed - \$0.017 per lb. milk

Cost Impact (22.5°/50 mile Sector) - \$$2.46 \times 10^5$ per month

Thus, the direct cost of placing cows on stored feed within a 22.5° Sector out to 50 miles based on farm value, would be about \$0.25 million per month. The cost would be zero during that portion of the year and in geographical areas where cattle are already on stored feed. While such protective actions might be required for periods up to 2 months at areas near the accident site, such would not be the case at the greater distances which involve the major portion of the area. Condemnation of milk is the protective action of last resort for areas of very high contamination. As noted above, the farm value of milk is \$0.30 per liter. Thus, the condemnation of milk at the Emergency PAG for a 22.5°/50 mile sector would have a cost of about \$2 million for a month of protective action. Where I-131 is the only significant contaminant, whole milk can be diverted to manufactured products, such as powdered milk, which can be stored to allow disappearance of the radioactivity. We do not have cost figures for this action.

It is of interest to compare the arbitrary assumptions on land area used above to the contamination resulting from the Windscale accident. (NB: This was not a power reactor of the type presently used in the United States). According to Booker, the Windscale accident resulted in milk values exceeding 0.015 μCi/l at about 200 kilometers or 125 miles downwind (75). Milk contamination was estimated as exceeding 0.01 μCi/l over 16,700 km^2 or 6400 mi^2. Thus, the Windscale accident resulted in contamination exceeding the Preventive PAG over an area about 10 times greater than that assumed above. Protective actions were taken at Windscale at milk concentrations of 0.1 μCi/l (approximately the Emergency PAG) and involved an area of about 520 km^2 or 200 mi^2 for periods of 3-6 weeks.

5.3 COST-EFFECTIVENESS ANALYSIS

Cost-effectiveness analysis is defined as the economy with which a particular task may be carried out.

The data available for this analysis was obtained with the cooperation of the Nuclear Regulatory Commission. Briefly, the NRC employed the models cited in the Reactor Safety Study (7) to evaluate the agricultural costs and cumulative dose commitment that could occur under two accident conditions - a design basis for siting purposes and a PWR 7 accident (7). A typical reactor site in the northeastern United States was envisioned. Unfortunately, the parameters employed are not directly comparable with the pathways and dosimetric parameters associated with the PAG's. Nevertheless, a good indication of the effects of taking a protective action (in this case the condemnation of milk) at specific interdiction levels can be ascertained.

Figure 5 presents the agricultural costs at specific interdiction criteria. The costs are, for the most part, associated with the market value of milk. The interdiction criteria are in terms of rem to an infant thyroid. From Figure 3, it can be seen that costs drop rapidly between 0.5 and 10 rem and more gradually after 20 rem. The ratio of costs for a design basis accident (siting) to a PWR 7 remains constant.

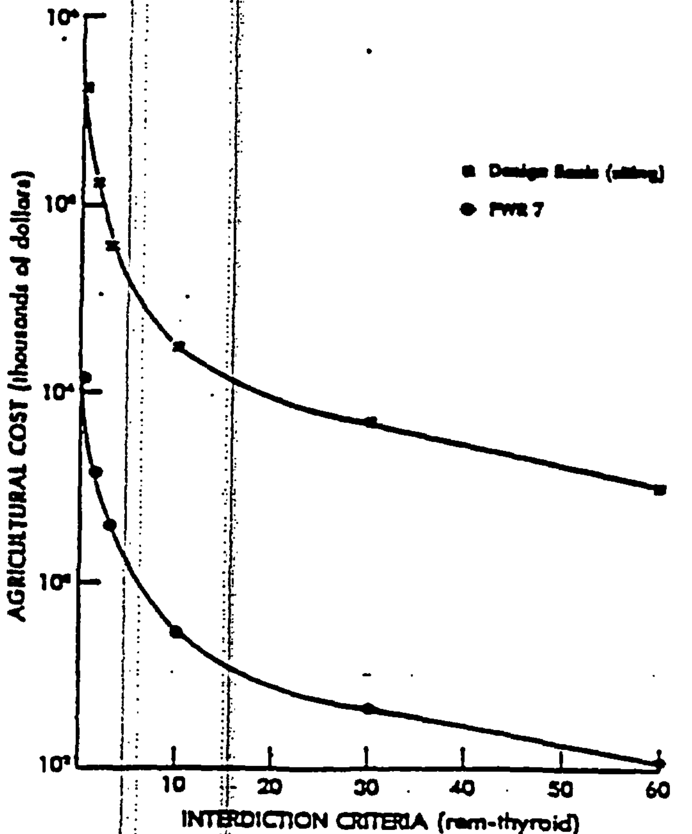


Figure 3. Agricultural cost model accident.

Figure 4 is a graph of the dose commitment for a design basis accident and a PWR 7 assuming protective action is initiated at specific interdiction levels. The dose commitments are accrued via external as well as internal exposure (inhalation and ingestion). Therefore, they do not exactly fit the situation described in the PAG's under consideration. The dose commitment rises rapidly when the interdiction criteria are between 0.5 and 20 rem. The increase in dose commitment for a design basis accident is less rapid than for a PWR 7. Hence, at or above an interdiction criteria of 20 rem, savings in radiation dose are minimal compared to the savings accrued below 10 rem.

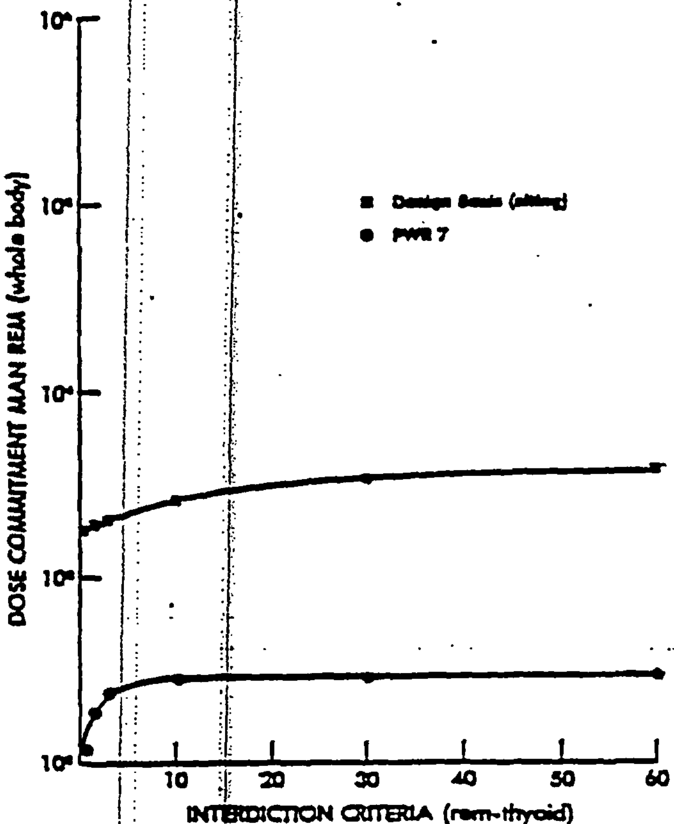


Figure 4. Dose commitment model accident.

5.4 SUMMARY AND CONCLUSION

The milk concentration at which the population benefits (from dose avoided) equals the direct costs of stored feed is equivalent to about one-third of the Preventive PAG for strontium and to one-fiftieth or less for iodine-131 and cesium. If condemnation is based on retail milk value, then the respective concentrations are about one-half and one-fiftieth of the Emergency PAG. Unless the indirect costs of implementing protective actions are significantly greater than the direct costs, it appears feasible to take protective actions at the respective PAG level and to continue such action to avoid about 90 percent of the projected dose for iodine and cesium. In the case of strontium contamination of milk, such action is only cost beneficial until the concentration is about 30 percent of the PAG response level.

Estimated costs of taking protective action within the Emergency Planning Zone (EPZ) for a 22.5° Sector to 50 miles (about 500 mi²) is \$2 million per month for condemnation (farm milk value) and \$0.26 million per month for use of stored feed. In the case of atmospheric dispersed contamination, protective action may have to continue for 2 months near the site.

The recommended approach is to place all cows on stored feed to prevent the contamination of milk at significant levels, to divert iodine contaminated milk to manufactured products that have a long shelf life to allow radioactive decay, and only consider condemnation of milk exceeding the Emergency PAG. It appears that doses to the public can be limited to less than 10 percent of the Preventive PAG (or less then 0.15 rem thyroid) by actions having direct costs of a few millon dollars for a significant accident.

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CHAPTER 7

Implementing the Protective Action Guides for the Intermediate Phase: Exposure to Deposited Materials

7.1 Introduction

This chapter provides guidance for implementing the PAGs set forth in Chapter 4. It is for use by State and local officials in developing their radiological emergency response plans to protect the public from exposure to radiation from deposited radioactive materials. Due to the wide variety in types of nuclear incidents and radionuclide releases that could occur, it is not practical to provide implementing guidance for all situations. The guidance in this chapter applies primarily to radionuclides that would be involved in incidents at nuclear power plants. It may be useful for radionuclides from incidents at other types of nuclear facilities or from incidents not involving fixed facilities (e.g., transportation accidents).However, specific implementation procedures for incidents other than those at nuclear power plants should be developed by planners on a case-by-case basis.

Contrary to the situation during the early phase of a nuclear incident, when decisions usually must be made and implemented quickly by State and local officials before Federal assistance is available, many decisions and actions during the intermediate phase can be delayed until Federal resources are present, as described in the Federal Radiological Monitoring and Assessment Plan (FE-85). Because of the reduced level of urgency for immediate implementation of these protective actions, somewhat less detail may be needed in State radiological emergency response plans than is required for the early phase.

At the time of decisions on relocation and early decontamination, sheltering and evacuation should have already been completed to protect the public from exposure to the airborne plume and from high exposure rates from deposited materials. These protective actions may have been implemented prior to verification of the path of the plume and therefore some persons may have been unnecessarily evacuated from areas where actual doses are much lower than were projected. Others who were in the path of the plume may have been sheltered or not protected at all. During the intermediate phase of the response, persons must be relocated from areas where the projected dose exceeds the PAG for relocation, and other actions taken to reduce doses to persons who are not relocated from contaminated areas. Persons evacuated from areas outside the relocation zone may return.

7.1.1 Protective Actions

The main protective actions for reducing exposure of the public to deposited radioactivity are relocation, decontami- nation, shielding, time limits on exposure, and control of the spread of surface contamination. Relocation is the most effective, and, usually, the most costly and disruptive. It is therefore only applied when the dose is sufficiently high to warrant it. The others are generally applied to reduce exposure of persons who are not relocated, or who return from evacuation status to areas that received lower. levels of deposited radioactivity. This chapter provides guidance for translating radiological conditions in the environment to projected dose, to provide the basis for decisions on the appropriate protective actions.

7.1.2 Areas Involved

Figure 7-1 provides a generalized example of the different areas and population groups to be dealt with. The path of the plume is assumed to be represented by area 1. In reality, variations in meteorological conditions would almost certainly produce a more complicated shape, but the same principles would apply.

Because of plant conditions and other considerations prior to or after the release, persons will already have been evacuated from area 2 and sheltered in area 3. Persons who have been evacuated from or sheltered in areas 2 and 3, respectively, as precautionary actions for protection from the plume, but whose homes are outside the plume deposition area (area 1), may return to their homes as soon as environmental monitoring verifies the boundary of the area that received deposition (area 1).

Area 4 is designated a restricted zone and is defined as the area where projected doses are equal to or greater than the relocation PAG. Persons residing just outside the boundary of the restricted zone may receive a dose near the PAG for relocation if decontamination or other dose reduction efforts are not implemented.

Area 1, with the exception of the restricted zone, represents the area of contamination that may continue to be occupied by the general public. Nevertheless, there will be contamination levels in this area that will require continued monitoring and dose reduction efforts other than relocation.

The relative positions of the boundaries shown in Figure 7-1 depend on areas evacuated and sheltered, and the radiological characteristics of the release. For example, area 4 (the restricted zone) could fall entirely inside area 2 (area evacuated), so that the only persons to be relocated would be those residing in area 4 who were either missed in the evacuation process or who, because of the high risk for their evacuation, had remained sheltered during plume passage.

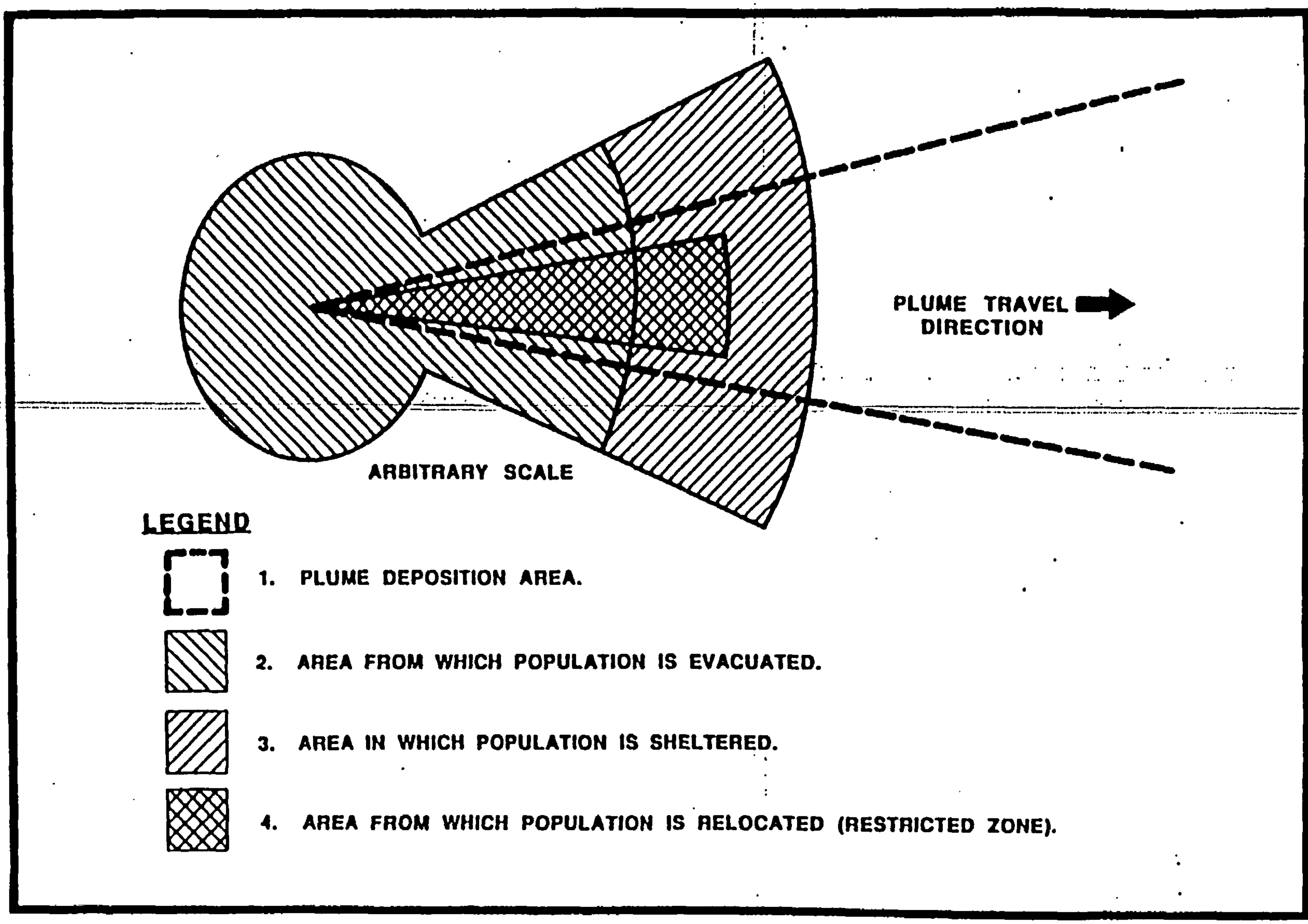

FIGURE 7-1. RESPONSE AREAS.

At the time the restricted zone is established, a temporary buffer zone (not shown in Figure 7-1) may be needed outside portions of the restricted zone in which occupants will not be allowed to return until monitoring confirms the stability of deposited contamination. Such zones would be near highly contaminated areas in the restricted zone where deposited radionuclides might be resuspended and then redeposited outside the restricted zone. This could be especially important at locations close to the incident site where the radioactivity levels are high and the restricted zone may be narrow. The extent of the buffer zone will depend on local conditions. Similarly, a buffer zone encompassing the most highly contaminated areas in which persons are allowed to reside may be needed. This area should be monitored routinely to assure acceptability for continued occupancy.

7.1.3 Sequence of Events

Following passage of the airborne plume, several tasks, as shown in Figure 7-2, must be accomplished simultaneously to provide for timely protection of the public. The decisions on the early task of relocating persons from high exposure rate areas must be based on exposure rate measurements and dose analyses. It is expected that monitoring and dose assessment will be an on-going process, with priority given to the areas with the highest exposure rate. The general sequence of events is itemized below, but the time frames will overlap, as demonstrated in Figure 7-2.

1. Based on environmental data, determine the areas where the projected one-year dose will exceed 2 rem and relocate persons from those areas, with priority given persons in the highest exposure rate areas.

2. Allow persons who were evacuated to return immediately to their residences if they are in areas where field gamma measurements indicate that exposure rates are near normal background levels (not in excess of twice the normal background in the area before the incident). If, however, areas of high deposition are found to be near areas with low deposition such that resuspended activity could drift into the occupied areas, a buffer zone should be established to restrict occupancy until the situation is analyzed and dose projections are confirmed.

3. Determine the location of the isodose line corresponding to the relocation PAG, establish the boundary of the restricted zone, and relocate any persons who still reside within the zone. Also, convert any evacuees who reside within the restricted zone to relocation status. Evacuated persons whose residence is in the area between the boundary of the plume deposition and the boundary to the restricted zone may return gradually as confidence is gained regarding the projected dose in the area.

4. Evaluate the dose reduction effectiveness of simple decontamination

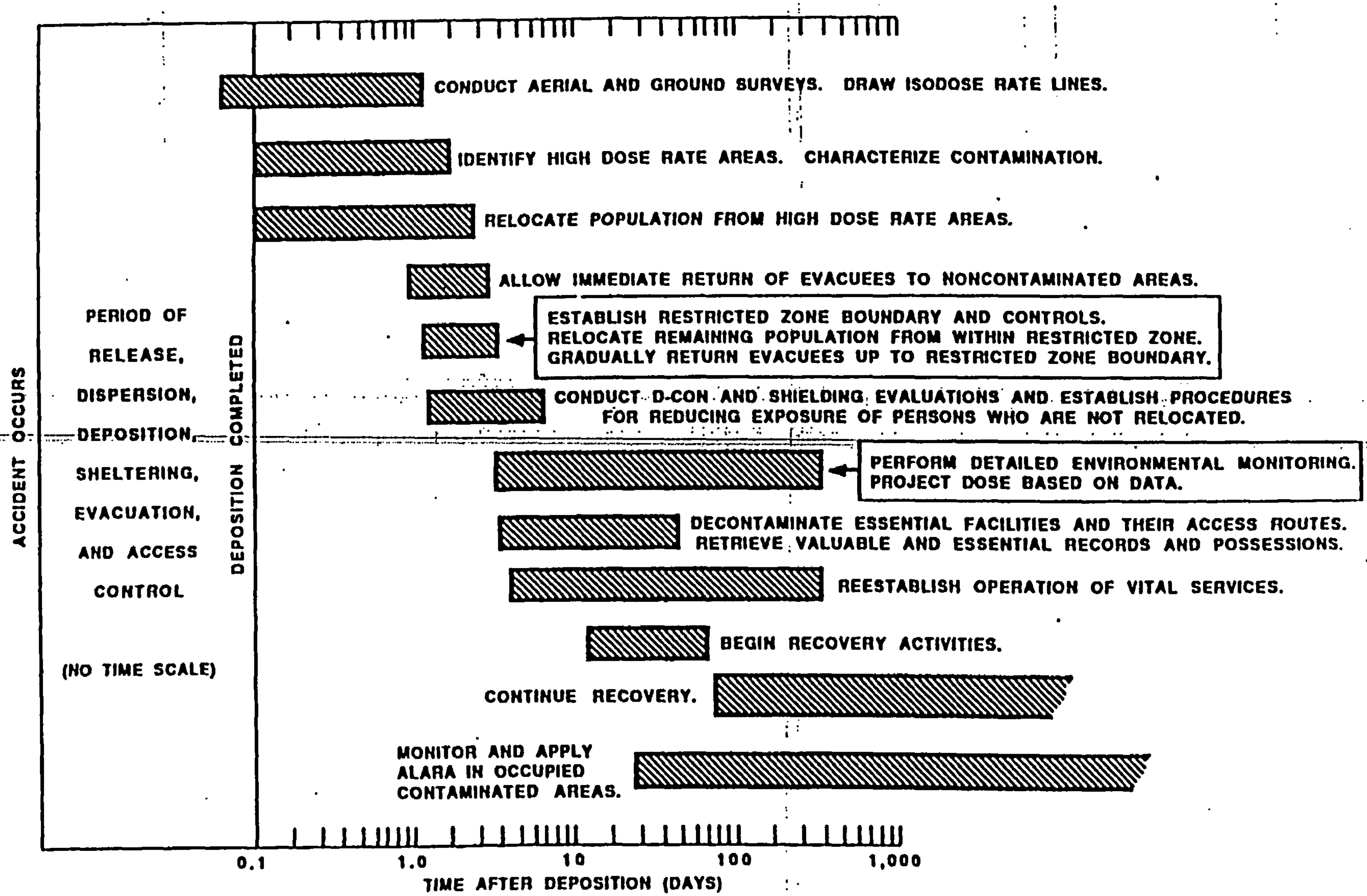

FIGURE 7-2. POTENTIAL TIME FRAME OF RESPONSE TO A NUCLEAR INCIDENT.

techniques and of sheltering due to partial occupancy of residences and workplaces. Results of these evaluations may influence recommendations for reducing exposure rates for persons who are not relocated from areas near, but outside, the restricted zone.

5. Establish a mechanism for controlling access to and egress from the restricted zone. Typically this would be accomplished through control points at roadway accesses to the restricted zone.

6. Establish monitoring and decontamination stations to support control of the restricted zone.

7. Implement simple decontamination techniques in contaminated areas outside the restricted zone, with priorities for areas with higher exposure rates and for residences of pregnant women.

8. Collect data needed to establish long-term radiation protection criteria for recovery and data to determine the effectiveness of various decontamination or other recovery techniques.

9. Begin operations to recover contaminated property in the restricted zone.

7.2 Establishment of Isodose-rate Lines

As soon as Federal or other assistance is available for aerial and ground monitoring, a concentrated effort should begin to establish isodose-rate lines on maps and the identification of boundaries to the restricted zone. Planning for this effort should include the development of standard maps that can be used by all of the involved monitoring and dose assessment organizations to record monitoring data.

Aerial monitoring (e.g., the Department of Energy Aerial Monitoring Service) can be used to collect data for establishing general patterns of radiation exposure rates from deposited radioactive material. These data, after translation to readings at 1 meter above ground, may form the primary basis for the development of isodose lines out to a distance where aerial monitoring shows no radiation above twice natural background levels. Air sample measurements will also be needed to verify the contribution to dose from inhalation of resuspended materials.

Gamma exposure rates measured at 1 meter will no doubt vary as a function of the location of the measurement within a very small area. This could be caused by different deposition rates for different types of surfaces (e.g., smooth surfaces versus heavy vegetation). Rinsing or precipitation could also reduce levels in some areas and raise levels in others where runoff settles. In general, where exposure rates vary within designated areas, the higher values should be used for dose projection for persons within these areas unless judgment can be used to estimate an appropriate average exposure rate.

Measurements made at 1 meter to project whole body dose from gamma radiation should be made with instruments. of the "closed window" type so as to avoid the detection of beta radiation. Although beta exposure will contribute to skin dose, its contribution to the overall risk of health effects from the radionuclides expected to be associated with reactor incidents should not be controlling in comparison to the whole body gamma dose (AR-89). Special beta dose analyses may be appropriate when time permits to determine its contribution to skin dose. Since beta dose rate measurements require sophisticated equipment that is generally not available for field use, beta dose to the skin should be limited based on measured concentrations of radionuclides per unit area.

7.3 Dose Projection

The primary dose of interest for reactor incidents is the sum of the effective gamma dose equivalent from external exposure and the committed effective dose equivalent from inhalation. The exposure periods of interest are first year, second year, and up to 50 years after the incident.

Calculation of the projected gamma dose from measurements will require knowledge of the principal radionuclides contributing to exposure and their relative abundances. Information on these radiological characteristics can be compiled either through the use of portable gamma spectrometers or by radionuclide analysis of environmental samples. Several measurement locations may be required to determine whether any selective radionuclide deposition occurred as a function of weather, surface type, distance from the point of release, or other factors. As part of the Federal Radio- logical Monitoring and Assessment Plan (FE-85), the U. S. Department of Energy and the U. S. Environmental Protection Agency have equipment and procedures to assist State officials in performing environmental measurements, including determination of the radiological characteristics of deposited materials.

The gamma exposure rate may decrease rapidly if deposited material includes a significant fraction of short-lived radionuclides. Therefore, the relationship between instantaneous exposure rate and projected first- and second-year annual or the 50-year doses will change as a function of time, and these relationships must be established for the particular mix of deposited radioactive materials present at the time of the gamma exposure rate measurement.

For incidents involving releases from nuclear power plants, gamma radiation from deposited radioactive materials is expected to be the principal exposure pathway, as noted above. Other pathways should also be evaluated, and their contributions considered, if significant. These may include inhalation of resuspended material and beta dose to the skin.

Exposure from ingestion of food and water is normally limited independently of decisions for relocation and decontamination (see Chapters 3 and 6). In rare instances, however, where withdrawal of food and/or water from use would, in itself, create a health risk, relocation may be an appropriate protective action for protection from exposure via ingestion. In this case, the committed effective dose equivalent from ingestion should be added to the projected dose from other exposure pathways for decisions on relocation.

The following sections provide methods for evaluating the projected dose from whole body external exposure and from inhalation of resuspended particulate material, based on environmental information.

7.3.1 Projected External Gamma Dose

Projected whole body external gamma doses at 1 meter height at particular locations during the first year, second year, and over the 50-year period after the incident are the parameters of interest. The environmental information available for calculating these doses is expected to be the current gamma exposure rate at 1 meter height and the relative abundance of each radionuclide contributing significantly to that exposure rate. Calculational models are available for predicting future exposure rates as a function of time due to radioactive decay and weathering. Weathering is discussed in WASH-1400, Appendix VI (NR-75), and information on the relationship between surface concentrations and gamma exposure rate at 1 meter is addressed in reference (DO-88).

Following the incident, experiments should be conducted to determine the dose reduction factors associated with part-time occupancy of dwellings and workplaces, and with simple, rapid, decontamination techniques, so that these factors can also be applied to the calculation of dose to persons who are not relocated. However, these factors should not be included in the calculation of projected dose for decisions on relocation.

Relocation decisions can generally be made on the basis of the first year projected dose. However, projected doses during the second year and over 50 years are needed for decisions on the need for other protective actions for persons who are not relocated. Dose conversion factors are therefore needed for converting environmental measurements to projected dose during the first year, second year, and over 50 years following the incident. Of the two types of environmental measurements that can be made to project whole body external gamma dose, gamma exposure rate in air is the easiest to make and is the most directly linked to gamma dose rate. However, a few measurements of the second type (radionuclide concentrations on surfaces) will also be needed to properly project decreasing dose rates.

Tables 7-1 and 7-2 provide information to simplify development

Table 7-1 Gamma Exposure Rate and Effective Dose Equivalent (Corrected for Radioactive Decay and Weathering) due to an Initial Uniform Concentration of 1 pCi/m² on Ground Surface

Radionuclide	Half-life (hours)	Initial exposure[a] rate at 1 m (mR/h per pCi/m²)	Integrated dose (weathering factor included)[b] year one (mrem per pCi/m²)	year two (mrem per pCi/m²)	0-50 years (mrem per pCi/m²)
Zr-95	1.54E+03	1.2E-08	3.3E-05	4.0E-07	3.4E-05
Nb-95	8.41E+02	1.3E-08	(b)	(b)	(b)
Ru-103	9.44E+02	8.2E-09	7.1E-06	0	7.1E-06
Ru-106	8.84E+03	3.4E-09	1.2E-05	3.7E-06	1.8E-05
Te-132	7.82E+01	4.0E-09	3.2E-06	0	3.2E-06
I-131	1.93E+02	6.6E-09	1.3E-06	0	1.3E-06
I-132	2.30E+00	3.7E-08	(b)	(b)	(b)
I-133	2.08E+01	1.0E-08	2.1E-07	0	2.1E-07
I-135	6.61E+00	2.4E-08	1.6E-07	0	1.6E-07
Cs-134	1.81E+04	2.6E-08	1.0E-04	4.7E-05	2.4E-04
Cs-137	2.65E+05	1.0E-08	4.5E-05	2.9E-05	6.1E-04
Ba-140	3.07E+02	3.2E-09	1.1E-05	0	1.1E-05
La-140	4.02E+01	3.5E-08	(b)	(b)	(b)

[a]Estimated exposure rate at 1 meter above contaminated ground surface. Based on data from reference (DO-88).

[b]Radionuclides that have short-lived daughters (Zr/Nb-95, Te/I-132, Ru/Rh-106, Cs-137/Ba-137m, Ba/La-140) are assumed to quickly reach equilibrium. The integrated dose factors listed are the effective gamma dose due to the parent and the daughter. Based on data from reference (DO-88).

Table 7-2 Exposure Rate and Effective Dose Equivalent (Corrected for Radioactive Decay) due to an Initial Concentration of 1 pCi/m^2 on Ground Surface

Radionuclide	Half-life (hours)	Initial exposure[a] rate at 1 m (mR/h per pCi/m^2)	Integrated dose (weathering factor not included)[b]		
			year one (mrem per pCi/m^2)	year two (mrem per pCi/m^2)	0-50 years (mrem per pCi/m^2)
Zr-95	1.54E+03	1.2E-08	3.8E-05	8.0E-07	3.9E-05
Nb-95	8.41E+02	1.3E-08	(b)	(b)	(b)
Ru-103	9.44E+02	8.2E-09	7.8E-06	0	7.8E-06
Ru-106	8.84E+03	3.4E-09	1.5E-05	7.6E-06	3.0E-05
Te-132	7.82E+01	4.0E-09	3.3E-06	0	3.3E-06
I-131	1.93E+02	6.6E-09	1.3E-06	0	1.3E-06
I-132	2.30E+00	3.7E-08	(b)	(b)	(b)
I-133	2.08E+01	1.0E-08	2.1E-07	0	2.1E-07
I-135	6.61E+00	2.4E-08	1.6E-07	0	1.6E-07
Cs-134	1.81E+04	2.6E-08	1.3E-04	9.6E-05	4.7E-04
Cs-137	2.65E+05	1.0E-08	6.0E-05	5.9E-05	1.8E-03
Ba-140	3.07E+02	3.2E-09	1.2E-05	0	1.2E-05
La-140	4.02E+01	3.5E-08	(b)	(b)	(b)

[a]Estimated exposure rate at 1 meter above contaminated ground surface. Based on data from reference (DO-88).

[b]Radionuclides that have short-lived daughters (Zr/Nb-95, Ru/Rh-106, Te/I-132, Cs-137/Ba-137m, Ba/La-140) are assumed to quickly reach equilibrium. The integrated dose factors listed are the effective gamma dose due to the parent and the daughter. Based on data from reference (DO-88).

of dose conversion factors through the use of data on the radionuclide mix, as determined from environmental measurements. These tables list the deposited radionuclides most likely to be the major contributors to dose from incidents at nuclear power facilities. In addition to providing integrated, effective doses per unit of surface concentration, they provide, in column three, the exposure rate (mR/h) in air per unit of surface contamination. All exposure rate values are based on those given in reference (DO-88). They were estimated from the total body photon dose rate conversion factors for exposure at 1 m above the ground surface. Since the ratio of effective dose to air exposure is about 0.7, dividing the effective dose rate by 0.7 results in an estimate of the exposure rate in air. The integrated effective doses are based on dose conversion factors also listed in reference (DO-88). Table 7-1 takes into account both radioactive decay and weathering, whereas the values in Table 7-2 include only radioactive decay. The effect of weathering is uncertain and will vary depending on the type of weather, type of surface, and the chemical form of the radionuclides. The user may choose either table depending on the confidence accorded the assumed weathering factors.

The following steps can be used to develop dose conversion factors to calculate projected future doses from gamma exposure rate measurements for specific mixes of radionuclides:

1. Using spectral analysis of gamma emissions from an environmental sample of deposited radioactivity, determine the relative abundance of the principal gamma emitting radionuclides. Analyses of uniform samples from several different locations may be necessary to determine whether the relative concentrations of radionuclides are constant. The results may be expressed as the activity (pCi) of each radionuclide in the sample.

2. Multiply each activity from step 1 by the corresponding values in column 3 of Table 7-1 or Table 7-2 (depending on whether or not weathering is to be considered) to determine the relative contribution to the gamma exposure rate (mR/h) at 1-meter height for each radionuclide. Sum the results for each sample.

3. Similarly, multiply each activity from step 1 by the corresponding values in columns 4, 5, and 6 to determine the 1st-year, 2nd-year, and 50-year relative integrated doses contributed by each radionuclide. Sum these results for each sample. Radionuclides listed in Tables 7-1 and 7-2 that have short-lived daughters (Zr/Nb-95, Te/I-132, Ru/Rh-106, Cs-137/Ba-137m, Ba/La-140) were assumed to be in equilibrium with their daughters when the tabulated values for integrated dose were calculated. Since the values for the parents include the total dose from the parent and the daughter, do not double count these daughters in the sum. (In the cases of Cs-137/Ba-137m, and Ru-106/Rh-106, the parents are not gamma emitters, so the listed exposure rates and doses are actually those from the daughters alone.)

4. Using the results from steps 2 and 3, the relevant dose conversion factors, *DCF*, for each sample are then given by:

$$DCF = \frac{\sum_{1}^{n} H_i}{\sum_{1}^{n} X_i}$$

where H_i = effective dose equivalent for radionuclide i (mrem),
X_i = gamma exposure rate for radionuclide i (mR/h)
n = the number of radionuclides in the sample.

Since the samples represented in the numerator and denominator are identical, the effect of the size of the sample cancels.

These dose conversion factors may be applied to any measured gamma exposure rate for which the relative concentrations of radionuclides are the same as those in the sample that was analyzed.

The following example demonstrates the use of the above procedures for calculating a DCF. For purposes of the example it is assumed that environmental measurements revealed a mix of radionuclides as shown in column 3 of Table 7-3. The (relative) exposure rate conversion factors in column 4 of Table 7-3 are taken from column 3 of Table 7-1. The (relative) exposure rates in column 5 are the products of columns 3 and 4. The (relative) doses for individual radionuclides in columns 6, 7, and 8 were calculated by multiplying the concentrations in column 3 by the dose conversion factors in columns 4, 5, and 6 of Table 7-1, respectively. (Columns 4, 5, and 6 of Table 7-2, which do not include weathering, could have been used instead of those in Table 7-1.)

For this example, the conversion factor for dose in the first year was obtained for the assumed radionuclide mix from the totals of columns 5 and 6 of Table 7-3, which indicate that a calculated dose of 0.023 mrem in the first year corresponds to an initial exposure rate of 1.5E-4 mR/h. Therefore, the first year dose conversion factor (DCF_1) for this example is 150 mrem for each mR/h measured at the beginning of the period.

This DCF may be multiplied by any gamma exposure rate measurement to estimate the dose in the first year for locations where the exposure rate is produced by a radionuclide mix the same as assumed for calculating the DCF, and where weathering affects the exposure rate in the same manner as assumed. For example, if a gamma exposure rate measurement were taken at the location where the contamination sample in Table 7-3 was taken, this exposure rate could be multiplied by the DCF calculated in the above example to obtain the projected first year dose at that point. Based on the example analysis and a relocation PAG of 2 rem, for this case the exposure rate at the boundary of the restricted zone should be no greater than

Table 7-3 Example[a] Calculation of Dose Conversion Factors for Gamma Exposure Rate Measurements Based on Measured Isotopic Concentrations[b]

Radionuclide	Half-life (hours)	Measured concentration (pCi/sample[d])	mR/h[c] / pCi/m^2	Calculated Exposure rate at 1 m (mR/hr)	Calculated effective dose at 1 meter		
					year one (mrem)	year two (mrem)	50 years (mrem)
I-131	1.93E+2	2.6E+2	6.6E-9	1.7E-6	3.3E-4	0	3.3E-4
Te-132	7.8E+1	3.6E+3	4.0E-9	1.4E-5	1.2E-2	0	1.2E-2
I-132	2.3	3.6E+3	3.7E-8	1.3E-4	(e)	(e)	(e)
Ru-103	9.44E+2	2.2E+2	8.2E-9	1.8E-6	1.6E-3	0	1.6E-3
Rh-106[f]	8.84E+3	5.0E+1	3.4E-9	1.7E-7	5.8E-4	1.9E-4	9.2E-4
Cs-134	1.81E+4	6.8E+1	2.6E-8	1.8E-6	6.9E-3	3.2E-3	1.6E-2
Ba-137m[f]	2.65E+5	4.2E+1	1.0E-8	4.2E-7	1.9E-3	1.2E-3	2.6E-2
Totals				1.5E-4	2.3E-2	4.6E-3	5.6E-2

[a]The data in this table are only examples to demonstrate a calculational process. The results should not be used in the prediction of relationships that would exist following a nuclear incident.

[b]Calculations are based on data in Table 7-1, which includes consideration of both radioactive decay and weathering.

[c]External exposure rate factors at 1 meter above ground for a person standing on contaminated ground, based on data in Table 7-1.

[d]The size of the sample is not important for this analysis because only the relative concentrations are needed to calculate the ratio of integrated dose to exposure rate.

[e]The integrated dose from I-132 is not calculated separately because it is the short-lived daughter of Te-132, and is assumed to be in equilibrium with it. The assumed quantity present is that for a daughter in equilibrium with the parent.

[f]This is a short lived daughter of a parent that has no gamma emissions and the halflife given is that of the parent.

$$\frac{2000\ mrem}{150\ mrem/mR/h} = 13\ mR/h,$$

if the contribution to effective dose from inhalation of resuspended radioactive materials is zero (See Section 7.3.2). The example DCF for the second year and 50 years are obtained by a similar process, yielding DCFs of 31 and 370 mrem per mR/h, respectively.

The ratio of the second year to first year dose is 31/150 = 0.21. If this is the case, persons not relocated on the basis of a 2 rem PAG should, for this example, receive no more than 0.21 x 2 = 0.4 rem in year 2. Similarly, the dose in fifty years should be no more than 4.9 rem. Actual doses should be less than these values to the extent that exposure rates are reduced by shielding from structures and by decontamination.

Prior to reaching conclusions regarding the gamma exposure rate that would correspond to the relocation PAG, one would need to verify by multiple sampling the consistency of the relative abundance of specific radionuclides as well as the relative importance of the inhalation pathway.

Dose conversion factors will change as a function of the radiological makeup of the deposited material. Therefore, dose conversion factors must be calculated based on the best current information following the incident. Since the relative concentrations will change as a function of time due to different decay rates, dose conversion factors must be calculated for specific measurement times of interest. By calculating the decay of the original sample(s), a plot of dose conversion factors (mrem per mR/h) as a function of time after the incident can be developed. As weathering changes the radionuclide mix, and as more is learned about other dose reduction mechanisms, such predictions of dose conversion factors may require adjustment.

7.3.2 Inhalation Dose Projection

It can be shown, for the mixture of radionuclides assumed to be deposited from postulated reactor incidents, and an assumed average resuspension factor of $10^{-6}\ m^{-1}$, that the effective dose from inhalation is small compared to the corresponding effective dose from external exposure to gamma radiation. However, air sample analyses should be performed for specific situations (e.g., areas of average and high dynamic activity) to determine the magnitude of possible inhalation exposure. The 50-year committed effective dose equivalent (H_{50}) resulting from the inhalation of resuspended airborne radioactive materials is calculated as follows:

$$H_{50} = I \times DCF \qquad (1)$$

where

I = total intake (µCi), and

DCF = committed effective dose equivalent per unit intake (rem/µCi).

It is assumed that the intake rate will decrease with time due to radioactive decay and weathering. No model is available to calculate the effect of weathering on resuspension of deposited materials, so the model developed for calculating its effect on gamma exposure rate (NR-75) is assumed to be valid. This should provide conservative results. The total intake (I) from inhalation over time t may be calculated for each radionuclide, using the following equation:

$$I = BC_0 \left[\frac{0.63}{\lambda_1+\lambda_2} \left(1-e^{-(\lambda_1+\lambda_2)t}\right) + \frac{0.37}{\lambda_1+\lambda_3} \left(1-e^{-(\lambda_1+\lambda_3)t}\right) \right] \quad (2)$$

where

B = average breathing rate for adults
= 1.05E+4 m^3/a (EP-88),

C_o = initial measured concentration of the resuspended radionuclide in air (pCi/m^3),

t = time during which radionuclides are inhaled (a),

λ_1 = radioactive decay constant (a^{-1}),

λ_2 = assumed weathering decay constant for 63 percent of the deposited activity, taken as 1.13 a^{-1} (NR-75), and

λ_3 = assumed weathering decay constant for 37 percent of the deposited activity, taken as 7.48 E-3 a^{-1} (NR-75).

Table 7-4 tabulates results calculated using the above assumptions for weathering. The table contains factors relating the committed effective dose from exposure during the first and second years after the incident to an initial air concentration of 1 pCi/m^3 for each of the principal radionuclides expected to be of concern from reactor incidents. The dose conversion factors are taken from FGR-11 (EP-88). Parent radionuclides and their short lived daughters are grouped together because these dose conversion factors are based on the assumption that both parents and daughters will occur in equal concentrations and will decay with the half life of the parent. Therefore, measured concentrations of the short lived daughters should be ignored and only the parent concentrations should be used in calculating long term projected doses.

Table 7-4 lists factors which include the effects of both weathering and radioactive decay, as well as those that include only the effects of radioactive decay. Users of these data should decide which factors to use based on their confidence on the applicability of the weathering models used (NR-75) to their environment.

The committed effective dose equivalent is calculated by multiplying the measured initial air concentration (pCi/m^3) for each radionuclide of concern by the appropriate factor from the table and summing the results. This sum may then be added to the corresponding external whole body gamma dose to yield the total com-

Table 7-4 Dose Conversion Factors for Inhalation of Resuspended Material[a]

Radionuclide[b]	Lung class[c]	Committed effective dose equivalent per unit air concentration at the beginning of year one (mrem per pCi/m³): Considering radioactive decay and weathering		Considering radioactive decay only	
		Year one	Year two	Year one	Year two
Sr-90/Y-90	Y/Y	1.0E+1	5.5E 0	1.4E+1	1.3E+1
Zr-95/Nb-95	Y/Y	6.5E-2	-	7.9E-2	-
Ru-103	Y	1.3E-2	-	1.5E-2	-
Ru-106/Rh-106	Y/D	2.8E 0	1.0E 0	3.7E 0	1.9E 0
Te-132/I-132	W/D	1.3E-3	-	1.3E-3	-
I-131	D	1.1E-2	-	1.1E-2	-
Cs-134	D	3.1E-1	1.5E-1	4.1E-1	3.0E-1
Cs-137/Ba-137m	D/D	2.5E-1	1.4E-1	3.3E-1	3.2E-1
Ba-140/La-140	D/W	4.4E-3	-	4.7E-3	-
Ce-144/Pr-144	Y/Y	2.0E 0	4.2E-1	2.7E 0	9.8E-1

[a]Calculated using the dose factors in EP-88, Table 2.1.

[b]Short lived daughters are not listed separately because the entries include the dose from both the daughter and the parent. These factors are based on the concentration of the parent only, at the beginning of the exposure period.

[c]The lung clearance class chosen is that which results in the highest dose conversion factor.

mitted effective dose equivalent from these two pathways.

The PAGs include a guide for dose to skin which is 50 times the magnitude of the PAG for effective dose. Analysis (AR-89) indicates that this guide is not likely to be controlling for radionuclide mixes expected to be associated with nuclear power plant incidents. Dose conversion factors are provided in Table 7-5 for use in case of incidents where the source term consists primarily of pure beta emitters. The skin dose from each radionuclide may be calculated by multiplying the measured concentration (pCi/m^2) by the corresponding dose conversion factor in the table. This will yield the first year beta dose to the skin at one meter height from exposure to deposited materials plus the estimated dose to the skin from materials deposited on the skin as a result of being in the contaminated area. These factors are calculated based on information in Reference AR-89, which used weathering factors that apply for gamma radiation and would, therefore, be conservative for application to beta radiation. Calculated doses based on these factors should be higher than the doses that would be received.

7.4 Priorities

In most cases protective actions during the intermediate phase will be carried out over a period of many days. It is therefore useful to consider what priorities are appropriate. Further, for situations where the affected area is so large that it is impractical to relocate all of the public, especially from areas exceeding the PAGs by only a small amount, priorities are needed for protective actions. The following priorities are appropriate:

1. As a first priority, assure that all persons are protected from doses that could cause acute health effects from all exposure pathways, including previous exposure to the plume.

2. Recommend the application of simple decontamination techniques and that persons remain indoors as much as possible to reduce exposure rates.

3. Establish priorities for relocation with emphasis on high exposure rate areas and pregnant women (especially those in the 8th to 15th week of pregnancy).

7.5 Reentry

After the restricted zone is established, persons will need to reenter for a variety of reasons, including recovery activities, retrieval of property, security patrol, operation of vital services, and, in some cases, care and feeding of farm and other animals. It may be possible to quickly decontaminate access ways to vital institutions and businesses in certain areas so that they can be occupied by adults either for living (e.g., institutions such as nursing homes, and hospitals) or for employment. Clearance of these areas for such occupancy will require dose reduction to comply with occupational exposure

Table 7-5 Skin Beta Dose Conversion Factors for Deposited Radionuclides[a]

Radionuclide	Dose conversion factor[b] (mrem per pCi/m^2)	
	Radioactive decay plus weathering	Radioactive decay only
Co-58	1.2E-7	1.4E-7
Co-60	4.2E-7	5.6E-7
Rb-86	6.3E-5	6.7E-5
Sr-89	1.5E-4	1.6E-4
Sr-90	1.2E-5	1.7E-5
Y-90	2.2E-4	2.9E-4
Y-91	1.6E-4	1.9E-4
Zr-95	7.2E-7	8.3E-7
Nb-95	6.1E-7	7.4E-7
Mo-99	4.4E-6	4.6E-6
Tc-99m	7.7E-9	7.7E-9
Ru-103	6.8E-7	7.8E-7
Ru-106[c]	6.4E-7	8.7E-7
Rh-105	6.5E-8	6.6E-8
Sb-127	3.4E-6	3.4E-6
Te-127	1.0E-6	1.0E-6
Te-127m	7.8E-7	9.5E-7
Te-129	5.0E-7	5.0E-7
Te-129m	3.4E-5	3.6E-5
Te-131m	2.9E-7	2.9E-7
Te-132	5.4E-9	5.4E-9
I-131	8.5E-7	8.7E-7
I-132	5.0E-5	5.0E-5
Cs-134	2.6E-5	3.3E-5
Cs-136[c]	1.4E-7	3.7E-7
Cs-137[c]	2.1E-5	2.9E-5
Ba-140	9.1E-6	9.6E-6
La-140	1.2E-5	1.3E-5
Ce-141	6.6E-7	7.1E-7

Table 7-5, Continued

Radionuclide	Dose conversion factors[b] (mrem per pCi/m^2)	
	Radioactive decay plus weathering	Radioactive decay only
Ce-143	2.3E-6	2.3E-6
Ce-144[c]	8.7E-7	1.1E-6
Pr-143	1.3E-5	1.4E-5
Nd-147	4.3E-6	4.5E-6
Np-239	3.4E-8	3.4E-8
Am-241	4.6E-8	6.4E-8

[a]Based on data from reference AR-89.

[b]Dose equivalent integrated for a one-year exposure at one meter height plus the estimated dose to the skin from materials deposited on the skin as a result of being in the contaminated area.

[c]Contributions from short-lived (one hour or less) decay products are included in dose factors for the parent radionuclides (i.e., Rh-106, Ba-136, Ba-137, and Pr-144).

limits (EP-87). Dose projections for individuals should take into account the maximum expected duration of exposure.

Persons working in areas inside the restricted zone should operate under the controlled conditions normally established for occupational exposure (EP-87).

7.6 Surface Contamination Control

Areas under the plume can be expected to contain deposited radioactive materials if aerosols or particulate materials were released during the incident. In extreme cases, individuals and equipment may be highly contaminated, and screening stations will be required for emergency monitoring and decontamination of individuals and to evaluate the need for medical evaluation. Equipment should be checked at this point and decontaminated as necessary to avoid the spread of contamination to other locations. This screening service would be required for only a few days following plume passage until all such persons have been evacuated or relocated.

After the restricted zone is established, based on the PAGs for relocation, adults may reenter the restricted zone under controlled conditions in accordance with occupational exposure standards. Monitoring stations will be required along roadways to control surface contamination at exits from the restricted zone. Because of the possibly high background radiation levels at control points near exits, significant levels of surface contamination on persons and equipment may be undetectable at these locations. Therefore, additional monitoring and decontamination stations may be needed at nearby low background locations. Decontamination and other measures should be implemented to maintain low exposure rates at monitoring stations.

7.6.1 Considerations and Constraints

Surface contamination limits to control routine operations at nuclear facilities and to transport radioactive material are generally set at levels lower than are practical for situations involving high-level, widespread contamination of the environment.

The principal exposure pathways for loose surface contamination on persons, clothing, and equipment are (a) internal doses from ingestion by direct transfer, (b) internal doses from inhalation of resuspended materials, (c) beta dose to skin from contaminated skin or clothing or from nearby surfaces, and (d) dose to the whole body from external gamma radiation.

Because of the difficulties in predicting the destiny of uncontrolled surface contamination, a contaminated individual or item should not be released to an unrestricted area unless contamination levels are low enough that they produce only a small increment of risk to health (e.g., less than 20 percent), compared to the risk to health from the principle exposure pathway (e.g., whole body gamma dose) in areas immediately outside the restricted zone. On the other hand, a level of contamination comparable to that existing on surfaces immediately outside the restricted zone may be acceptable on materials leaving the restricted zone. Otherwise, persons and equipment occupying areas immediately outside the restricted zone would not meet the surface contamination limits. These two constraints are used to set permissible surface contamination limits.

The contamination limit should also be influenced by the potential for the contamination to be ingested, inhaled, or transferred to other locations. Therefore, it is reasonable to establish lower limits for surfaces where contamination is loose than for surfaces where the contamination is fixed except for skin. The expected period of fixed contamination on skin would be longer so a lower limit would be justified.

For routine (nonincident) situations, measurement of gross beta-gamma surface contamination levels is commonly performed with a thin-window geiger counter (such as a CDV-700). Since beta-gamma measurements made with such field instruments cannot be interpreted in terms of dose or exposure rate, the guidance set forth below is related to the background radiation level in the area where the measurement is being made. Supplementary levels are provided for gamma exposure rates measured with the beta shield closed. Guidance levels expressed in this form should be easily detectable and should satisfy the above considerations. Corresponding or lower levels expressed in units related to instrument designations may be adopted for convenience or for ALARA determinations. Smears may also be used to detect loose surface contamination at very low levels. However, they are not considered necessary for emergency response and, therefore, such guidance is not provided.

7.6.2 Numerical Relationships

As discussed in Section 7.3.1, a relationship can be established between projected first year doses and instantaneous gamma exposure rates from properly characterized surface contamination. Based on assumed radiological characteristics of releases from fuel melt accidents, gamma exposure rates in areas where the projected dose is equal to the relocation PAG of 2 rem in the first year may be in the range of 2 to 5 mR/h during the first few days following the deposition from a type SST-2 accident (See Section E.1.2). (This relationship must be determined for each specific release mixture.) Based on relationships in reference (DO-88) and a mixture of radionuclides expected to be typical of an SST-2 type accident, surface contamination levels of $2x10^{k}$ pCi/m^2 would correspond approximately to a gamma exposure rate of 1 mR/h at 1 meter height.

7.6.3 Recommended Surface Contamination Limits

Surface contamination must be controlled both before and after relocation PAGs are implemented. Therefore, this section deals with the control of surface contamination on persons and equipment being protected during both the early and intermediate phases of a nuclear incident.

For emergency situations, the following general guidance regarding surface contamination is recommended:

1. Do not delay urgent medical care for decontamination efforts or for time-consuming protection of attendants.

2. Do not waste effort trying to contain contaminated wash water.

3. Do not allow monitoring and decontamination to delay evacuation from high or potentially high exposure rate areas.

4. (Optional provision, for use only if a major contaminating event occurs, and rapid early screening is needed.)

After plume passage, it may be necessary to establish emergency contamination screening stations in areas not qualifying as low background areas. Such areas should be less than 5 mR/h gamma exposure rate. These screening stations should be used only during the early phase and for major releases of particulate materials to the atmosphere to monitor persons emerging from possible high exposure areas, provide simple (rapid) decontamination if needed, and make decisions on whether to send them for special care or to a monitoring and decontamination station in a lower background area. Table 7-6 provides guidance on surface contamination levels for use if such centers are needed.

5. Establish monitoring and personnel decontamination (bathing) facilities at evacuation centers or other locations in low background areas (less than 0.1 mR/h). Encourage evacuated persons who were exposed in areas where inhalation of particulate materials would have warranted evacuation to bathe, change clothes, wash clothes, and wash other exposed surfaces such as cars and trucks and their contents and then report to these centers for monitoring. Table 7-7 provides surface contamination guidance for use at these centers. These screening levels are examples derived primarily on the basis of easily measurable concentrations using portable instruments.

6. After the restricted zone is established, set up monitoring and decontamination stations at exits from the restricted zone. Because of the probably high background radiation levels at these locations, low levels of contamination may be undetectable. If contamination levels are undetectable, then they probably do not exceed those in some unrestricted areas occupied by the exposed population and no decontamination is required. Nevertheless, these individuals should be advised to bathe and change clothes at their first opportunity and certainly within the next 24 hours. If, after decontamination at the boundary of the restricted zone station, persons still exceed the limits for this station, they should be sent for further decontamination or for medical or other special attention. As an alternative to decontamination, contaminated items other than persons or animals may be retained in the restricted zone for radioactive decay.

7. Establish auxiliary monitoring and decontamination stations in low background areas (background less than 0.1 mR/h). These stations should be used to achieve ALARA surface contamination levels. Table 7-7 provides surface contamination screening levels for use at those stations.

Table 7-6 Recommended Surface Contamination Screening Levels for Emergency Screening of Persons and Other Surfaces at Screening or Monitoring Stations in High Background Radiation Areas (0.1 mR/h to 5 mR/h Gamma Exposure)[a]

Condition	Geiger-counter shielded-window reading	Recommended action
Before decontamination	<2X bkgd and <0.5 mR/h above background	Unconditional release
	>2X bkgd or >0.5 mR/h above background	Decontaminate Equipment may be stored or disposed of as appropriate.
After decontamination	<2X bkgd and <0.5 mR/h above background	Unconditional release
	>2X bkgd or >0.5 mR/h above background	Continue to decontaminate or refer to low back-ground monitoring and d-con station. Equipment may also be stored for decay or disposed of as appropriate.

[a]Monitoring stations in such high exposure rate areas are for use only during the early phase of an incident involving major atmospheric releases of particulates. Otherwise use Table 7-7.

Table 7-7 Recommended Surface Contamination Screening Levels for Persons and Other Surfaces at Monitoring Stations in Low Background Radiation Areas (<0.1 mR/h Gamma Exposure Rate)

Condition	Geiger-counter thin window[a] reading	Recommended action
Before decontamination	<2X bkgd	Unconditional release
	>2X bkgd	Decontaminate
After simple[b] decontamination effort	<2X bkgd	Unconditional release
	>2X bkgd	Full decontamination
After full[c] decontamination effort	<2X bkgd	Unconditional release
	>2X bkgd	Continue to decontaminate persons
	<0.5 mR/h[d]	Release animals and equipment
After additional full decontamination effort	<2X bkgd	Unconditional full release
	>2X bkgd	Send persons for special evaluation
	<0.5 mR/h[d]	Release animals and equipment
	>0.5 mR/h[d]	Refer, or use informed judgment on further control of animals and equipment

[a]Window thickness of approximately 30 mg/cm^2 is acceptable. Recommended limits for open window readings are expressed as twice the existing background (including background) in the area where measurements are being made. Corresponding levels, expressed in units related to instrument designations, may be adopted for convenience. Levels higher than twice background

References

AR-89 Aaberg, Rosanne. Evaluation of Skin and Ingestion Exposure Pathways. EPA 520/1-89-016, U.S. Environmental Protection Agency, Washington (1989).

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EP-87 U.S. Environmental Protection Agency. Radiation Protection Guidance to Federal Agencies for Occupational Exposure. Federal Register, 52, 2822; January, 1987.

EP-88 U.S. Environmental Protection Agency. Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion. EPA 520/1-88-020, U.S. Environmental Protection Agency, Washington (1988).

FE-85 Federal Emergency Management Agency. Federal Radiological Emergency Response Plan (FRERP). Federal Register, 50, 46542; November 8, 1985.

NR-75 U. S. Nuclear Regulatory Commission. Reactor Safety Study. An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants. WASH-1400, NUREG-75/014, U.S. Nuclear Regulatory Commission, Washington, (1975).

(footnote continued)

(not to exceed the meter reading corresponding to 0.1 mR/h) may be used to speed the monitoring of evacuees in very low background areas.

[b] Flushing with water and wiping is an example of a simple decontamination effort.

[c] Washing or scrubbing with soap or solvent followed by flushing is an example of a full decontamination effort.

[d] Closed shield reading including background.

CHAPTER 8

Radiation Protection Guides for the Late Phase (Recovery)

(Reserved)

APPENDIX A

Glossary

APPENDIX A

Glossary

The following definitions apply specifically to terms used in this manual.

Acute health effects: Prompt radiation effects (those that would be observable within a short period of time) for which the severity of the effect varies with the dose, and for which a practical threshold exist.

Ablation: The functional destruction of an organ through surgery or exposure to large doses of radiation.

Buffer zone: An expanded portion of the restricted zone selected for temporary radiation protection controls until the stability of radioactivity levels in the area is confirmed.

Cloudshine: Gamma radiation from radioactive materials in an airborne plume.

Committed dose: The radiation dose due to radionuclides in the body over a 50-year period following their inhalation or ingestion.

Delayed health effects: Radiation effects which are manifested long after the relevant exposure. The vast majority are stochastic, that is, the severity is independent of dose and the probability is assumed to be proportional to the dose, without threshold.

Derived response level (DRL): A level of radioactivity in an environmental medium that would be expected to produce a dose equal to its corresponding Protective Action Guide.

Dose conversion factor: Any factor that is used to change an environmental measurement to dose in the units of concern.

Dose equivalent: The product of the absorbed dose in rad, a quality factor related to the biological effectiveness of the radiation involved and any other modifying factors.

Effective dose equivalent: The sum of the products of the dose equivalent to each organ and a weighting factor, where the weighting factor is the ratio of the risk of mortality from delayed health effects arising from irradiation of a particular organ or tissue to the total risk of mortality from delayed health effects when the whole body is irradiated uniformly to the same dose.

Evacuation: The urgent removal of people from an area to avoid or reduce high-level, short-term exposure, usually from the plume or from deposited activity. Evacuation may be a preemptive action taken in response to a facility condition rather than an actual release.

Genetic effect: An effect in a descendant resulting from the modification of genetic material in a parent.

Groundshine: Gamma radiation from radioactive materials deposited on the ground.

Incident phase: This guidance distinguishes three phases of an incident (or accident): (a) early phase, (b) intermediate phase, and (c) late phase.

(a) Early phase: The period at the beginning of a nuclear incident when immediate decisions for effective use of protective actions are required, and must be based primarily on predictions of radiological conditions in the environment. This phase may last from hours to days. For the purpose of dose projection, it is assumed to last for four days.

(b) Intermediate phase: The period beginning after the incident source and releases have been brought under control and reliable environmental measurements are available for use as a basis for decisions on additional protective actions and extending until these protective actions are terminated. This phase may overlap the early and late phases and may last from weeks to many months. For the purpose of dose projection, it is assumed to last for one year.

(c) Late phase: The period beginning when recovery action designed to reduce radiation levels in the environment to permanently acceptable levels are commenced, and ending when all recovery actions have been completed. This period may extend from months to years (also referred to as the recovery phase).

Linear Energy Transfer (LET): A measure of the ability of biological material to absorb ionizing radiation; specifically, for charged particles traversing a medium, the energy lost per unit length of path as a result of those collisions with elections in which the energy loss is less than a specified maximum value. A similar quantity may be defined for photons.

Nuclear incident: An event or series of events, either deliberate or accidental, leading to the release, or potential release, into the environment of radioactive materials in sufficient quantity to warrant consideration of protective actions.

Prodromal effects: The forewarning symptoms of more serious health effects.

Projected dose: Future dose calculated for a specified time period on the basis of estimated or measured initial concentrations of radionuclides or exposure rates and in the absence of protective actions.

Protective action: An activity conducted in response to an incident or potential incident to avoid or reduce radiation dose to members of the public (sometimes called a protective measure).

Protective Action Guide (PAG): The projected dose to reference man, or other defined individual, from an accidental release of radioactive material at which a specific protective action to reduce or avoid that dose is warranted.

Recovery: The process of reducing radiation exposure rates and concentrations of radioactive material in the environment to levels acceptable for unconditional occupancy or use.

Reentry: Temporary entry into a restricted zone under controlled conditions.

Relocation: The removal or continued exclusion of people (households) from contaminated areas to avoid chronic radiation exposure.

Restricted zone: An area with controlled access from which the population has been relocated.

Return: The reoccupation of areas cleared for unrestricted residence or use.

Sheltering: The use of a structure for radiation protection from an airborne plume and/or deposited radioactive materials.

Short-lived daughters: Radioactive progeny of radioactive isotopes that have half-lives on the order of a few hours or less.

Weathering factor: The fraction of radioactivity *remaining* after being affected by average weather conditions for a specified period of time.

Weighting factor: A factor chosen to approximate the ratio of the risk of fatal cancer from the irradiation of a specific tissue to the risk when the whole body is irradiated uniformly to the same dose.

Whole body dose: Dose resulting from uniform exposure of the entire body to either internal or external sources of radiation.

APPENDIX B

Risks To Health From Radiation Doses That May Result From Nuclear Incidents

Contents

Figures

Tables

APPENDIX B

Risks To Health From Radiation Doses That May Result From Nuclear Incidents

B.1 Introduction

This appendix reviews the risks from radiation that form the basis for the choice of Protective Action Guides (PAGs) for the response to a nuclear incident, as well as the choice of limits for occupational exposure during a nuclear incident.

B.1.1 Units of Dose

The objective of protective action is to reduce the risk to health from exposure to radiation. Ideally, one would like to assure the same level of protection for each member of the population. However, protective actions cannot take into account individual variations in radiosensitivity, since these are not known. Therefore, these PAGs are based on assumed average values of risk. We further assume that these risks are proportional to the dose, for any level of dose below the threshold for acute effects (see Section B.2.).

The dose from exposure to radioactive materials may be delivered during the period of environmental exposure only (e.g., external gamma radiation), or over a longer period (e.g., inhaled radionuclides which deposit in body organs). In the latter case, dose is delivered not only at the time of intake from the environment, but continues until all of the radioactive material has decayed or is eliminated from the body. Because of the variable time over which such doses may be delivered, the PAGs are expressed in terms of a quantity called the "committed dose." Conceptually, committed dose is the dose delivered over an individual's remaining lifetime following an intake of radioactive material. However, due to differences in physiology and remaining years of life, the committed dose to a child from internal radioactivity may differ from that to an adult. For simplicity, adult physiology and a remaining lifetime of 50 years are assumed for the purpose of calculating committed doses.

Another important consideration is that different parts of the body are at different risk from the same dose. Since the objective of protective actions is the reduction of health risk, it is appropriate to use a quantity called "effective dose." Effective dose is the sum of the products of the dose to each organ or tissue of the body and a weighting factor representing the relative risk. These weighting factors (IC-77) are chosen as the ratio of mortality (from delayed health effects) from irradiation of particular organs or tissues to the total risk of such mortality when the whole body is irradiated uniformly at the same dose.

Finally, doses from different types of radiation (e.g. alpha, beta, gamma, and neutron radiation) have different biological effectiveness. These differences are customarily accounted for, for purposes of radiation protection, by multiplicative modifying factors. A dose modified by these factors is designated the "dose equivalent." The PAGs are therefore expressed in terms of <u>committed effective dose equivalent</u>. The PAGs are augmented by limits for a few specific organs (skin and thyroid) which exhibit special sensitivity. These are expressed in terms of <u>committed dose equivalent</u> (rem). In the process of developing PAG values, it is necessary to evaluate the threshold dose levels for acute health effects. These levels are generally expressed in terms of absorbed dose (rad) to the whole body from short term (one month or less) exposure. Other units (Roentgens, rem, and rets) are also used in information cited from various references. They are all approximately numerically equivalent to rads in terms of the risk of acute health effects from beta and gamma radiation.

PAGs are intended to apply to all individuals in a population other than workers performing emergency services. However, there may be identifiable groups that have different average sensitivity to radiation or, because of their living situation, will receive higher or lower doses. In addition, some groups may be at greater risk from taking a given protective action. These factors are separately considered, when it is appropriate, in establishing values for the PAGs.

B.1.2 Principles for Establishing Protective Action Guides

The following four principles provide the basis for establishing values for Protective Action Guides:

1. Acute effects on health (those that would be observable within a short period of time and which have a dose threshold below which they are not likely to occur) should be avoided.

2. The risk of delayed effects on health (primarily cancer and genetic effects, for which linear nonthreshold relationships to dose are assumed) should not exceed upper bounds that are judged to be adequately protective of public health, under emergency conditions, and are reasonably achievable.

3. PAGs should not be higher than justified on the basis of optimization of cost and the collective risk of effects on health. That is, any reduction of risk to public health achievable at acceptable cost should be carried out.

4. Regardless of the above principles, the risk to health from a protective action should not itself exceed the risk to health from the dose that would be avoided.

With the exception of the second, these principles are similar to those set forth by the International Commission on Radiological Protection (IC-84b) as the basis for establishing intervention levels for nuclear accidents. We examine, below, the basis for estimating effects on health for use in applying the first two of these principles.

B.2 Acute Effects

This section provides information relevant to the first principle: avoidance of acute effects on health from radiation.

Acute radiation health effects are those clinically observable effects on health which are manifested within two or three months after exposure. Their severity depends on the amount of radiation dose that is received. Acute effects do not occur unless the dose is relatively large, and there is generally a level of dose (i.e., threshold) below which an effect is not expected to occur. Acute effects may be classified as severe or nonsevere clinical pathophysiological effects. Severe pathophysiological effects are those which have clinically observable symptoms and may lead to serious disease and death. Other pathophysiological effects, such as hematologic deficiencies, temporary infertility, and chromosome changes, are not considered to be severe, but may be detrimental in varying degrees. Some pathophysiological effects, such as erythema, nonmalignant skin damage, loss of appetite, nausea, fatigue, and diarrhea, when associated with whole body gamma or neutron exposure, are prodromal (forewarning of more serious pathophysiological effects, including death).

B.2.1 Review of Acute Effects

This section summarizes the results of a literature survey of reports of acute effects from short-term (arbitrarily taken as received in one month or less) radiation exposure in some detail. Many reports of observed effects at lower doses differ, and some are contradictory; however, most have been included for the sake of completeness. The results of the detailed review described in this Section are summarized in Section B.2.2.

The biological response to the rapid delivery of large radiation doses to man has been studied since the end of World War II. Dose-response relationships for prodromal (forewarning) symptoms and for death within 60 days have been developed from data on the Japanese A-bomb survivors, Marshall Island natives exposed to fallout, and patients undergoing radiotherapy. This work has been supplemented by a number of animal studies under controlled conditions.

The animal studies, usually using lethality as the end point, show that many factors can influence the degree of response. The rate at which the dose is delivered can affect the median lethal dose (LD_{50}) in many species, particularly at dose rates less than 5 R/min (PA-68a; BA-68). However, in primates there is less than a 50 percent increase in the LD_{50} as dose rates are decreased from 50 R/min to about 0.01 R/min (PA-68a). There is good evidence of species specificity (PA-68a; BO-69). The LD_{50} ranges from about 100 rad for burros to over 1000 rad for lagomorphs (e.g., rabbits). Response is modulated by: age (CA-68), extent of shielding (partial body irradiation) (BO-65), radiation quality (PA-68a; BO-69), diet, and state of health (CA-68).

While animal studies provide support and supplemental information, they cannot be used to infer the response for man. This lack of comparability of man and animals had already been noted by a review committee for the National Academy of Sciences as early as 1956, in considering the length of time over which acute effects might be expressed (NA-56): "Thus, an LD_{50}, 30-day consideration is inadequate to characterize the acute lethal dose response of man, and an LD_{50}, 60 days would be preferable."[1]

Several estimates of the levels at which acute effects of radiation occur in man have been published. For example, an early estimate of the dose-response curves for prodromal (forewarning) symptoms and for lethality was made in the first edition of "The Effects of Nuclear Weapons" (1957) (GL-57), and a more recent and well documented estimate is given in a NASA publication, "Radiobiological Factors in Manned Space Flight" (LA-67).

B.2.1.1 The Median Dose for Lethality

The radiation dose that would cause 50 percent mortality in 60 days was estimated as 450 Roentgens in early reports (NA-56; GL-57; RD-51). The National Commission on Radiation Protection and Measurements (NCRP) calculated that this would correspond to a midline absorbed dose of 315 rad (NC-74). The ratio of 315 rad to 450 Roentgens is 0.70, which is about the estimated ratio of the active marrow dose, in rads, to the tissue kerma in air, in rads (KE-80). The BEAR Committee noted that the customary reference to LD_{50} in animal studies, as if it were a specific property, independent of age, was not justifiable (NA-56): "...it is evident, now, that the susceptibility of a whole population is not describable by a single LD_{50}. The published values are usually obtained for young adults and are therefore maximal or nearly maximal for the strain. In attempts to estimate LD_{50} in man, this age dependence should be taken into consideration" (NA-56, pp.4-5). They observed that the LD_{50} approximately doubled as rats went from neonates to young adults and then decreased as the animals aged further. Finally, the BEAR

[1]The committee (known as the BEAR Committee) also noted "The reservation must be made here that the exposed Japanese population was heterogeneous with respect to age, sex, physical condition and degree of added trauma from burns or blast. The extent to which these factors affected the survival time has not been determined. In studies on laboratory animals the converse is true--homogeneous populations are studied" (NA-56, p.I-6).

Committee concluded: "The situation is complex, and it became evident that it is not possible to extrapolate with confidence from one condition of radiation exposure to another, or from animal data to man" (NA-56, p.I-8). Nevertheless, results from animal studies can aid in interpreting the human data that are available.

The NCRP suggested the $LD_{50/60}$ might be 10 to 20 percent lower for the old, very young, or sick, and somewhat greater for healthy adults of intermediate age (RD-51). Other estimates of adult $LD_{50/60}$ have ranged from about 300 rad to 243 $\pm$ 22 rad. These lower estimates are apparently based on a ratio of air to tissue dose similar to those calculated for midline organs in the body; 0.54 to 0.66 (KE-80; OB-76; KO-81).

A NASA panel examined all patient and accident studies, tried to remove confounding factors, and concluded, "On this basis, it may be assumed that the LD_{50} value of 286 rad obtained by a normal fit to the patient data is the preferred value for healthy man" (LA-67).

An $LD_{50/60}$ of 286 $\pm$ 25 rad (standard deviation) midline absorbed dose and an absorbed dose/air dose ratio of 0.66, suggested by the National Academy of Science (LA-67), is probably a reasonable value for healthy males. In the absence of more complete information, we assume that a value of 300 rad $\pm$ 30 rad is a reasonable reflection of current uncertainties for average members of the population.

B.2.1.2 Variation of Response for Lethality

Uncertainty in the dose-response function for acute effects has been expressed in various ways. The slope of the estimated dose-response function has most commonly been estimated on the basis of the percent difference in the LD_{50} and the $LD_{15.9}$ or $LD_{84.1}$ (one standard deviation from the LD_{50}), as was done by NASA (GL-57). These and other parameters derived in a similar manner describe the uncertainty in the central risk estimate for the dose-response function.

Another means is to use an estimate of upper and lower bounds for the central risk estimate, e.g., the 95 percent fiducial limits. At any given response point on the dose-response function, for example, the LD_{10}, the dose causing that response has a 95 percent probability of lying between the lower and upper bounds of the 95 percent fiducial limit for that point. To estimate this value, probit analyses were run for each species using data in published reports (KO-81; TA-71). This provided estimates for each species for comparability analyses. The 95 percent fiducial limits at the LD_{50} response for $LD_{50/30}$ studies averaged $\pm$9 percent (range -9 to +26 percent) and for $LD_{50/60}$ studies $\pm$17 percent (range -20 to +45 percent). At the LD_{15} response, values were $\pm$16 percent (range -12 to +50 percent) for $LD_{15/30}$ data and $\pm$26 percent (range -20 to +65 percent) for $LD_{15/60}$ data. For the LD_{85} response, values were $\pm$17 percent (range -36 to +36 percent) for the $LD_{85/30}$ data and

±24 percent (range -46 to +31 percent) for $LD_{85/60}$ data.

The differences in the magnitude of the fiducial limits are a function of the differences in age, sex, radiation quality, degree of homogeneity of the experimental animals, husbandry, and other factors. The estimates show that the fiducial limits, expressed as a percent of the dose at any response, get greater the farther from the LD_{50} the estimate is made. For the purpose of estimating fiducial limits for humans, the 95 percent fiducial limits will be considered to be LD_{15} ±15 percent, LD_{50} ±10 percent, and LD_{85} ±15 percent. Beyond these response levels, the fiducial limits are too uncertain and should not be used.

If the median lethal dose, $LD_{50/60}$, is taken as 300 ±30 rad midline absorbed dose, the response to higher and lower doses depends on the degree of biological variation in the exposed population. The NASA panel decided the wide variation in the sensitivity of patients was a reflection of the heterogeneity of the sample; and that the variation in sensitivity, the slope of the central estimate of the response function, would be stated in the form of one standard deviation calculated as 58 percent of the LD_{50}. They further decided the deviation in the patients (58 percent) was too great, and the standard deviation for "normal" man should be closer to that of dogs and monkeys (18 percent) (LA-67). (The rationale for selecting these species was not given.)

Jones attempted to evaluate the hematologic syndrome from mammalian lethality studies using the ratio of dose to LD_{50} dose as an indicator of the steepness of the slope of the dose-response function (JO-81). However, he evaluated LD_{50} studies only of species having a rather steep slope, i.e., dogs, monkeys, mice, and swine. He also looked at several different statistical models for dose-response functions and pointed out the problems caused by different models and assumptions, particularly in evaluating the tails of the dose-response function (less than LD_{10} and greater than LD_{90}). Jones recommended using a log-log model, which he felt provided a better fit at low doses (JO-81).

Scott and Hahn also evaluated acute effects from mammalian lethality, but suggested using a Weibull model (SC-80). One of the advantages of the Weibull model is that in addition to developing the dose-response function, it can also be used to develop hazard functions. These hazard functions, if developed using the same model, can be summed to find the joint hazard of several different types of exposure (SC-83). This would allow estimation of the total hazard from multiple organ exposures to different types of radiation.

As mentioned earlier, the human median lethal dose is commonly reported in terms of the $LD_{50/60}$. Most laboratory animal median lethal doses are reported in terms of the $LD_{50/30}$. In those cases where estimates of both $LD_{50/30}$ and $LD_{50/60}$ are available, i.e., the burro (ST-69), the variation (that is, the slope of the dose-response curve) is greater in the $LD_{50/60}$ study than in the $LD_{50/30}$ study. Both the dog and the monkey data are for $LD_{50/30}$, and so are not appropriate for direct comparison to man.

If an estimate of the deviation is made for data from other studies and species, those where most of the fatalities occur within 30 days (like dogs and monkeys) have standard deviations of from around 20 percent [swine (x-ray) (ST-69), dogs (NA-66), hamsters (AI-65), primates (Macaca) (DA-65)] to 30 percent [swine (^{60}Co) (HO-68)]. Those in which most deaths occur in 60 days, like man, have deviations from around 20 percent [sheep (CH-64)] to 40 percent [goats (PA-68b), burros (TA-71)]. Nachtwey, et al. (NA-66) suggested the steepness of the slope of the exposure response curve depends on the inherent variability of the subjects exposed and any variation induced by uncontrolled factors, e.g., temperature, diurnal rhythm, and state of stimulation or arousal. So, while the slope of the response curve for the patients studied by the NASA panel may be unrealistically shallow for normal human populations, there is no reason to think it should be as steep as those for dogs and monkeys.

The average deviation for those species (burros, sheep, and goats) for which the standard deviation of the $LD_{50/60}$ is available has been used as an estimator for man. The mean value is 34 ± 13 percent. This is only slightly greater than the average value for all physically large animals (swine, burros, sheep, and goats), 32 ± 12 percent.

B.2.1.3 Estimated Lethality vs Dose for Man

As noted in Section B.2.1.1, dose-response estimates vary for a number of reasons. Some factors affecting estimates for humans are:

1. Age:
Studies on rats indicate the LD_{50} is minimal for perinatal exposure, rises to maximum around puberty, and then decreases again with increasing age (CA-68). The perinatal LD_{50} is about one-third of that for the healthy young adult rats; that for the geriatric rat is about one-half of that for the young adult rat.

2. Sex:
Females are slightly more sensitive than males in most species (CA-68).

3. Health:
Animals in poor health are usually more sensitive than healthy animals (CA-68), unless elevated hematopoietic activity is occurring in healthy animals (SU-69).

While these and other factors will affect the $LD_{50/60}$ and the response curve for man, there are no numerical data available.

The variation in response at a given dose level increases as the population at risk becomes more heterogeneous and as the length of time over which mortality is expressed increases. In general, larger species show greater variance and longer periods of expression than do small mammals, e.g., rodents. It is likely that the human population would show at least the same amount of variation as do the larger animals, i.e., a coefficient of variation of about one-third.

The degree of variation exhibited in animal studies follows a Gaussian distribution as well as or better than a log normal distribution over that range of mortality where there are reasonable statistics. We have assumed here that the functional form of human response is Gaussian. Generally, sample sizes for extreme values (the upper and lower tails of the distribution) are too small to give meaningful results. Therefore, we have not projected risks for doses more than two standard deviations from the $LD_{50/60}$. We recognize that estimates of acute effects may not be reliable even beyond one standard deviation for a population containing persons of all ages and states of health. However, in spite of these uncertainties, previous estimates have been made of the acute effects caused by total body exposure to ionizing radiation as a function of the magnitude of the exposure (NC-71; LU-68; FA-73; NA-73).

Given the large uncertainties in the available data, a median lethal dose value of about 300 rad at the midline, with a standard deviation of 100 rad, may be assumed for planning purposes. Such risk estimates should be assumed to apply only in the interval from 5 percent to 95 percent fatality, as shown in Figure B-1. (See also section B.2.1.4.)

Figure B-1 is based on the following values:

Dose (rad)	Percent fatalities
<140	none[2]
140	5
200	15
300	50
400	85
460	95

For moderately severe prodromal (forewarning) effects, we believe the dose at which the same percentage of exposed would show effects would be approximately half of that causing fatality. This yields the following results (see also Figure B-1):

Dose (rad)	Percent affected
50	<2
100	15
150	50
200	85
250	98

Although some incidence of prodromal effects has been observed at doses in the range of 15 to 20 rads in patients (LU-68) and in the 0 to 10 rads range of dose in Japanese A-bomb survivors (SU-80a; GI-84),

[2]The risk of fatality below 140 rad is not necessarily zero; rather, it is indeterminate and likely to remain so. This also applies to prodromal effects below 50 rad.

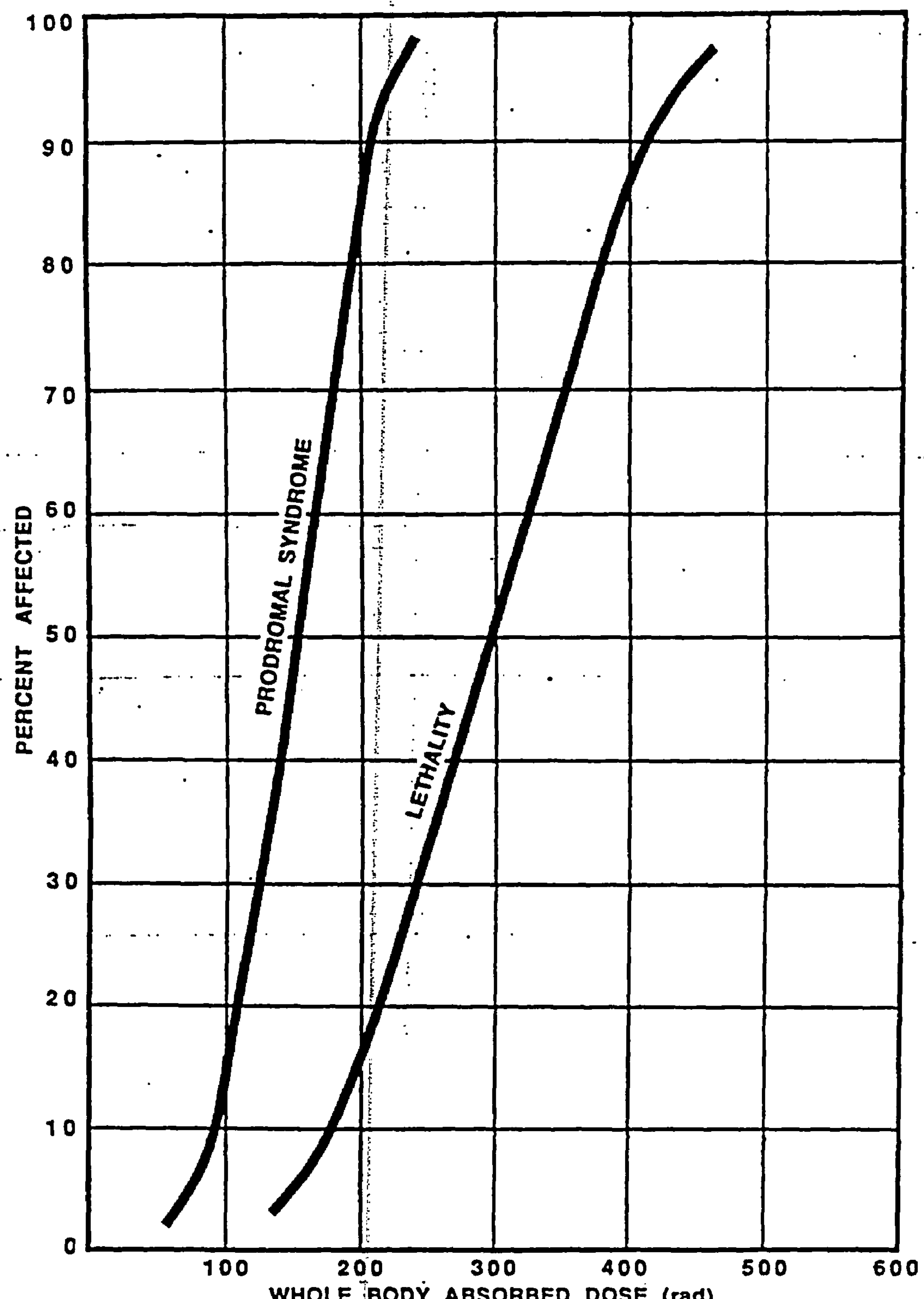

FIGURE B-1. ACUTE HEALTH EFFECTS AS A FUNCTION OF WHOLE BODY DOSE.

there is great uncertainty in interpreting the data. Patients may be abnormally sensitive, so that the dose-response function in patients may represent the lower bound of doses that would show a response in a healthy population (LU-67). The response of Japanese survivors in the low dose ranges is complicated by the blast and thermal exposure that occurred at the same time (SU-80b). For these reasons, care should be taken in applying estimates of prodromal effects. The prodomal dose-response function listed above is more likely to overestimate the proportion of persons affected than to underestimate it.

These estimated ranges and effects are in agreement with estimates made for manned space flights (LA-67; LU-67), which included consideration of the effect of abnormal physiology or sickness in the patients to which the data apply. Uncertainty in estimates of the biological effects of radiation exposure is great. It is probably due in part to variation in the health of individuals in exposed populations. These estimates assume a healthy young adult population and may not be a conservative estimate of risk for other population groups, such as children or the elderly. Lushbaugh, et al. (LU-68) found that prodromal effects probably occur in both healthy and ill persons in about the same dose range. However, Lushbaugh, et al. (LU-68) and NATO (NA-73) suggest that acute mortality in a population which is ill, injured, or in other ways debilitated will occur in 50 percent of that population at doses of 200-250 rad in about 60 days ($LD_{50/60}$), in contrast to an $LD_{50/60}$ from doses of 220-310 rad for a healthy young adult population. Thus, the ill or injured are assumed to have an increased risk of acute mortality at high doses.

The above estimates for $LD_{50/60}$ are also based on the assumption of minimal medical care following exposure. UNSCEAR (UN-88) estimates that the threshold for mortality would be about 50 percent higher in the presence of more intense medical care.

B.2.1.4 Threshold Dose Levels for Acute Effects

This section summarizes information available in the literature regarding thresholds for health effects. It also reviews actions that have been taken as a result of radiation exposure to provide insight on dose levels at which actions to avoid dose may be appropriate.

Some acute effects, such as cellular changes, may occur at low doses with no dose threshold. Most such effects have a minimum threshold of detectability; for example, five rad is about the lower limit of whole body dose which causes a cellular effect detectable by chromosome or other special analyses (NC-71; FA-73). This value is recommended by UNSCEAR as the starting point for biological dosimetry (UN-69). Purrott, et al. have reported a lower limit of detection of chromosome aberrations of 4 rad for x-rays and 10 rad for gamma rays (PU-75).

More recent advanced chromosome banding techniques permit detection of increased incidence of chromosome abnormalities from continuous exposure to systematically deposited radioisotopes or radioisotopes deposited in the lung at very low levels, e.g., body burdens of 100 to 1200 pCi of plutonium-239 (BR-77). While the exact dose associated with such burdens is not known, it is probably on the order of 10 to 100 millirem per year. Lymphocytes exposed to 5 rem <u>in vitro</u> show severe metabolic dysfunction and interphase cell death (ST-64). The extent to which similar effects occur after <u>in vivo</u> exposure is unknown. While chromosome abnormalities in circulating lymphocytes are reported to persist for long periods (UN-69), the significance of such abnormalities is not known (BR-77).

Hug has suggested 5 rem as the lower limit of exposure which might produce acute effects (WH-65). Five rad is also in the low dose, short-term exposure range defined by Cronkite and Haley, and is below the 10 rad which they thought would cause only a slight detectable physiological effect of unknown clinical significance (CR-71).

Although the ICRP has suggested that annual doses of 15 rad would not impair the fertility of normal fertile men (IC-69), an acute dose of 15 rad causes "moderate" oligospermia (approximately 70 percent reduction in sperm count) which lasts for some months (LA-67). Popescu and Lancranjan reported alterations of spermatogenesis and impaired fertility in men exposed to from 500 millirad to 3 rad per year for periods varying from 2 to 22 years (PO-75). The shortest exposure period in which abnormal spermatogenesis was reported was 31 to 41 months (PO-75); at the highest dose rate reported (3 rad/a), this is a cumulative dose of 8 to 10 rem. While more study is required, these results suggest the need to restrict acute doses to below 10 rem to avoid this effect, because a given acute dose is anticipated to be more effective than the same cumulative dose given over a longer period of time (NA-56; UN-58).

Many observations have indicated that doses of 10 rem or more to the pregnant woman are hazardous to the fetus. Mental retardation due to exposure of the fetus is discussed in Section B.3; this discussion is restricted to acute effects. The World Health Organization (WHO) indicates that there is no evidence of teratogenic effects from short term exposure of the fetus to a dose less than 10 rad during the early phase of gestation, the period when the fetus is most sensitive to these effects (WH-84).

A number of authorities have recommended that exposures of 10 roentgens or higher be considered as an indication for carrying out induced abortion (HA-59, DE-70, BR-72, NE-76). Brent and Gorson also suggest that 10 rad is a "practical" threshold for induction of fetal abnormalities (BR-72). The Swedish Government Committee on Urban Siting of Nuclear Power Stations stated the situation as follows: "What we have called unconditional indication of abortion involves the exposure of pregnant women where radiation dose to the fetus is higher than 10 rad. When such doses are received in connection with medical treatment, it has hitherto been assumed that the probability of damage to the fetus is so high that an abortion is recommended. The probability for such injury is still moderate compared with the normal frequency of similar fetal injuries, and the probability is particularly reduced when the dose is received late in the pregnancy" (NA-74).

B.2.1.5 Acute Effects in the Thyroid

Acute effects are produced in the thyroid by doses from radioiodine on the order of 3,000 to 100,000 rad. Ablation of the thyroid requires doses of 100,000 rad (BE-68). The thyroid can be rendered hypothyroid by doses of about 3,000 to 10,000 rad (IC-71). A thyroid dose from radioiodines of 1000 rad in adults and 400 rad in children implies an associated whole body dose of about 1 rad due to radioiodines circulating in the blood. Following inhalation of ^{131}I, the committed thyroid dose is about one rad/µCi intake of ^{131}I in adults. In the developing fetus, the thyroid dose ranges from one to six rad per µCi of ^{131}I entering the mother's body (IL-74).

Although acute clinical effects are only observed at high doses, subclinical acute thyroid radiation effects may occur at lower doses (DO-72). Impaired thyroid capability may occur above a threshold of about 200 rad (DO-72).

Effects of radiation exposure of the thyroid have been shown in animal experiments. Walinder and Sjoden found that doses in excess of 3,000 rad from ^{131}I caused noticeable depression of fetal and juvenile mouse thyroid development (WA-69). Moore and Calvin, working with the Chinese hamster, showed that an exposure as low as 10 roentgens (x-rays) would give rise to 3 percent aberrant cells when the thyroid was cultured (MO-68). While the direct relationship of these results to human effects is not certain, mammalian thyroid cells can be injured at exposures as low as 10 roentgens.

B.2.1.6 Acute Effects in the Skin

The first stage of skin reaction to radiation exposure is erythema (reddening) with a threshold of from 300 to 800 rad. Acute exudative radiodermatitis results from doses of 1,200 to 2,000 rad (WH-84).

B.2.1.7 Clinical Pathophysiological Effects

A large amount of anecdotal information is available on the injury of organ tissues by high doses of radiation. Acute injury to tissue includes swelling and vacuolation of the cells which make up the blood vessels, increased permeability of vessels to fluids so that exudates form, formation of fibrin clots and thrombi, fibrinoid thickening in the walls of blood vessels, and swelling and vacuolization of parenchymal cells. In summary, there is an initial exudative reaction followed in time by fibrosis and sclerosis (WH-76, CA-76).

Estimates of radiation doses necessary to cause severe tissue response in various organs are given in Table B-1. These tissue dose estimates are based on response to radiotherapy treatment, which is normally given on a fractionated dose basis, but also may be given as a continuous exposure. Therefore, these estimates must be adjusted to the equivalent single radiation dose for use in the present analysis. The formalism of Kirk, et al. (KI-71) is used to estimate the equivalent dose for a single acute exposure in rad-equivalent therapy units (rets: the dose calculated from the fractionated exposure which is equivalent to a single acute exposure for a specific biological endpoint.) Table B-2 lists acute exposure equivalents in rets for various organs.

With the exception of bone marrow, the exposures required to cause 5 percent injury within 5 years (TD 5/5) in internal organs are in the range of 1,000 to 5,000 rad. Since, with this type of injury, the dose response is nonlinear and has a threshold (i.e., is nonstochastic), there is an exposure below which injury is not expected. If the shape of the injury dose-response curve is the same for all internal organs as it is for the lung, plotting the two acute exposure equivalents (TD 50/5 and 5/5) for each organ on log probability paper allows a crude estimation of the number of clinical pathophysiological effects per 1000 persons exposed as a function of dose level. If one acute effect per 1000 persons within 5 years (TD 0.1/5) is taken as the threshold for the initiation of clinical pathophysiological effects in organs other than thyroid, the equivalent dose level for most organs is 550 rets or more; testes 440 ± 150 rets, ovary 170 ± 70 rets, and bone marrow 165 rets.

The radiation exposure to organs in rad units that will cause clinical pathophysiological effects within 5 years to 0.1 percent of the exposed population as a function of the duration of a continuous level of exposure can then be estimated by using Goitein's modification of the Kirk methodology (GO-76). This relationship is shown in Table B-3.

Bone marrow is an organ of particular concern because radionuclides known to concentrate in this organ system occur in nuclear incidents. The acute lethality due to the hematologic syndrome (LA-67) is estimated to occur in the range of 200 to 1,000 rad, so that the difference is small between exposure levels that might cause acute lethality and exposure levels that might cause only "severe clinical pathophysiology," as derived from radiotherapy data.

In summary, organ systems are not expected to show symptoms of severe clinical pathophysiology for projected short-term exposure doses less than a few hundred rad. Projected doses to bone marrow at this high level are relatively more serious and more likely to result in injury than doses to other organ systems.

Table B-1 Radiation Doses Causing Acute Injury to Organs (RU-72, RU-73)

Organ	Volume or area of exposure[a]	Risk of injury in five years: 5 percent (rad)	Risk of injury in five years: 50 percent (rad)	Type of injury
Bone marrow	whole	250	450	aplasia and pancytopenia
Liver	segment whole	3000 2500	4000 4000	acute and chronic hepatitis
Stomach	100 cm^2	4500	5500	ulcer, perforation, hemorrhage
Intestine	400 cm^2 100 cm^2	4500 5000	5500 6500	ulcer, perforation, hemorrhage
Lung	whole 100 cm^2	1500 3000	2500 3500	acute and chronic pneumonitis
Kidney	whole	2000	2500	acute and chronic nephrosclerosis
Brain	whole	6000	7000	infarction, necrosis
Spinal cord	10 cm	4500	5500	infarction, necrosis
Heart	60 percent	4500	5500	pericarditis and pancarditis
Skin	---	5500	7000	ulcers, fibrosis
Fetus	whole	200	400	death
Lens of eye	whole	500	1200	cataracts
Ovary	whole	200-300	625-1200	permanent sterilization
Testes	whole	500-1500	2000	permanent sterilization

[a]Dose delivered in 200-rad fractions, 5 fractions/week.

— Unspecified.

Table B-2 Acute Radiation Exposure as a Function of Rad Equivalent Therapy Units (rets)

Organ	Volume or area of exposure	Risk of injury in five years: 5 percent (rets)	Risk of injury in five years: 50 percent (rets)
Bone marrow	whole	230	340
	segment	1135	1360
Liver	whole	1000	1360
Stomach	100 cm^2	1465	1665
Intestine	400 cm^2	1465	1665
	100 cm^2	1570	1855
Lung	whole	720	1000
	100 cm^2	1135	1245
	75 percent	770[b]	---
Kidney	whole	875	1000
Brain	whole	1770	1950
Spinal cord	10 cm	1465	1665
Heart	60 percent	1465	1665
Skin	---	1665	1950
Fetus	whole	200	315
Lens of eye	whole	355	620
Ovary	whole	200-430[a]	410-875[a]
Testes	whole (sterilization)	340-720[a]	410-875[a]

[a]For a 200-rad/treatment, 5 treatments/week schedule (LU-76).

[b]Reference WA-73.

--- Unspecified.

Table B-3 Radiation Exposure to Organs Estimated to Cause Clinical Pathophysiological Effects within 5 Years to 0.1 Percent of the Exposed Population (GO-76)

Duration of exposure (days)	Ovary (rad)	Bone marrow (rad)	Testes (rad)	Other organs (rad)
(acute)	(170 rets)[a]	(165 rets)	(440 rets)	(550 rets)
1	315	300	810	1020
2	390	380	1010	1260
4	470	450	1210	1510
7	550	540	1430	1790
30	840	820	2190	2740
365[b]	1740	1690	4510	5640

[a]The dose in rets is numerically equal to the dose in rads.

[b]Assuming tissue recovery can continue at the same rate as observed during 30- to 60-day therapeutic exposure courses.

Even if severe clinical pathophysiological effects can be avoided, there is still a possibility of clinical pathophysiological effects of a less severe or transitory nature. The 1982 UNSCEAR report (UN-82) reviewed much of the data on animals and man. In the animal studies, there were reports of: changes in stomach acid secretion and stomach emptying at 50 to 130 rad; stunting in growing animals at the rate of 3 to 5 percent per 100 rad; degeneration of some cells or functions in the brain at 100 rad, particularly in growing animals; temporary changes in weight of hematopoietic tissues at 40 rad; and more damage in ovaries and testes caused by fractionated doses rather than acute doses. Some of the effects are transitory, others are long-lasting, but with only minor reductions in functional capacity.

Human data are limited and are reported primarily in the radiotherapy literature. The data suggest most tissues in man are more radiation resistant than those in animals. However, the human hematopoietic system shows a transient response, reflected by decreased circulating white cells and platelets, at about 50 rad. Temporary sterility has been observed after doses of 150 rad to the ovaries and 10 rad to the testes, when given as fractionated doses.

There is not sufficient data to determine dose-response functions nor to describe the duration and severity of dysfunction expected.

B.2.2 Summary and Conclusions Regarding Acute Effects

Based on the foregoing review of acute health effects and other biological effects from large doses delivered over short periods of time, the following whole body doses from acute exposure provide useful reference levels for decisionmaking for PAGs:

- 50 rad - Less than 2 percent of the exposed population would be expected to exhibit prodromal (forewarning) symptoms.
- 25 rad - Below the dose where prodromal symptoms have been observed.
- 10 rad - The dose level below which a fetus would not be expected to suffer teratogenesis (but see Section B.3, Mental Retardation.).
- 5 rad - The approximate minimum level of detectability for acute cellular effects using the most sensitive methods. Although these are not severe pathophysiological effects, they may be detrimental.

Based on the first principle to be satisfied by PAGs (paragraph B.1.6), which calls for avoiding acute health effects, values of 50 rem for adults and 10 rem for fetuses appear to represent upper bounds.

B.3 Mental Retardation

Brain damage to the unborn is a class of injury reported in atomic bomb survivors which does not fall into either an acute or delayed effect category, but exhibits elements of both. What has been observed is a significant, dose-related increase in the incidence and severity of mental retardation, microencephaly (small head size), and microcephaly (small brain size) in Japanese exposed to radiation in utero during the 8th to 15th week after conception (BL-73; MI-76). While the actual injury may be acute, it is not identified until some time after birth.

In an early study Mole (MO-82) suggested that, although radiation may not be the sole cause of these conditions, it is prudent to treat the phenomenon as radiation-related. More recently, Otake and Schull (OT-83) have concluded: (1) there is no risk to live-born due to doses delivered up to 8 weeks after conception, (2) most damage occurs at the time when rapid proliferation of neuronal elements occurs, i.e., 8 to 15 weeks of gestational age, (3) the dose-response function for incidence during this period appears to fit a linear model, (4) the risk of occurrence is about five times greater during the period 8-15 weeks of gestation than in subsequent weeks, and (5) in later stages of gestation, e.g., after the 15th week, a threshold for damage may exist.

In their published reports, Otake and Schull (OT-83) evaluated the incidence of severe mental retardation using the T-65 dosimetry and the dosimetry estimates developed in the ongoing dose reassessment program for the atomic bomb survivors, and using two tissue dose models. Their estimated ranges of risk were:

8 to 15 weeks after gestation: $3\text{-}4 \times 10^{-3}$ cases/rad;

16 or more weeks after gestation: $5\text{-}7 \times 10^{-4}$ cases/rad.

The higher values are based on the T-65 dosimetry and the Oak Ridge National Laboratory estimate of tissue dose. The lower values are based on Oak Ridge National Laboratory dosimetry and the Japanese National Institute of Radiological Sciences estimates of tissue dose. Later estimates based on the dose reassessment completed in 1986 are consistent with these published results (SC-87).

In view of the foregoing, the risk of mental retardation from exposure of a fetus in the 8th to 15th week of pregnancy is taken to be about 4×10^{-3} per rad. Because of this relatively high risk, special consideration should be given to protection of the fetus during this period. The risk to a fetus exposed after the 15th week is taken as 6×10^{-4} per rad. For the cases studied (OT-84), no increased incidence of mental retardation was observed for exposure during the 1st to the 7th week of pregnancy.

Federal Radiation Protection Guidance, adopted in 1987, recommends that dose to occupationally exposed pregnant women be controlled to keep the fetal dose below 0.5 rem over the entire term of pregnancy, and that no dose be delivered at more than the uniform monthly rate that would satisfy this limit (i.e., approximately 50-60 mrem/month)(EP-87). The NCRP has, for many years, recommended a limit of 0.5 rem (NC-71). ICRP recommends controlling exposure of the fetus to less than 0.5 rem in the first 2 months to provide appropriate protection during the essential period of organogenesis (IC-77).

B.4 Delayed Health Effects

This section addresses information relevant to the second principle (paragraph B.1.5) for establishing PAGs, the risk of delayed health effects in exposed individuals. The following subsections summarize the estimated risks of cancer and genetic effects, the two types of delayed effects caused by exposure to radiation.

B.4.1 Cancer

Because the effects of radiation on human health have been more extensively studied than the effects of many other environmental pollutants, it is possible to make numerical estimates of the risk as a result of a particular dose of radiation. Such estimates, may, however, give an unwarranted aura of certainty to estimated radiation risks. Compared to the baseline incidence of cancer and genetic defects, radiogenic cancer and genetic defects do not occur very frequently. Even in heavily irradiated populations, the number of cancers and genetic defects resulting from radiation is known with only limited accuracy. In addition, all members of existing exposed populations have not been followed for their full lifetimes, so data on the ultimate numbers of effects is not yet available. Moreover, when considered in light of information gained from experiments with animals and from various theories of carcinogenesis and mutagenesis, the observed data on the effects of human exposure are subject to a number of interpretations. This, in turn, leads to differing estimates of radiation risks by individual scientists and expert groups. In summary, the estimation of radiation risks is not a fully mature science and the evaluation of radiation hazards will continue to change as additional information becomes available.

Most of the observations of radiation-induced carcinogenesis in humans are on groups exposed to low-LET radiations. These groups include the Japanese A-bomb survivors and medical patients treated with x-rays for ankylosing spondylitis in England from 1935 to 1954 (SM-78). The National Academy of Science Committee on the Biological Effects of Ionizing Radiations (BEIR) (NA-80) and UNSCEAR (UN-77) have provided knowledgeable and exhaustive reviews of these and other data on the carcinogenic effects of human exposures. The most recent of the BEIR studies was published in 1980 and is here designated BEIR-3 to distinguish it from previous reports of the BEIR committee.

The most important epidemiological data on radiogenic cancer is that from the A-bomb survivors. The Japanese A-bomb survivors have been studied for more than 40 years, and most of them have been followed in a major, carefully planned and monitored epidemiological survey, the Life Span Study Sample, since 1950 (KA-82, WA-83). They were exposed to a wide range of doses and are the largest group that has been studied. They are virtually the only group providing extensive information on the response pattern at various levels of exposure to low-LET radiation.

The estimated cancer risk from low-LET, whole body, lifetime exposure presented here is based on a life table analysis using a linear response model. We use the arithmetic average of relative and absolute risk projections (the BEIR-3 L-L model) for solid cancers, and an absolute risk projection for leukemia and bone cancer (the BEIR-3 L-L model). For whole body dose, this yields an estimated 280 (with a possible range of 120 to 1200) fatalities per million person-rem for a population cohort representative of the 1970 U.S. population. We assume this estimate also applies to high-LET radiation (e.g. alpha emitters); no reduction has been applied for dose rate. (The rounded value, 3×10^{-4} fatalities[3] per person-rem, has been selected for this analysis.)

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[3]Preliminary reviews of new results from studies of populations exposed at Hiroshima

Whole body dose means a uniform dose to every organ in the body. In practice, such exposure situations seldom occur, particularly for ingested or inhaled radioactivity. Inhaled radioactive particulate materials may be either soluble or insoluble. Soluble particulate materials deposited in the lung will be rapidly absorbed, and the radionuclides associated with them distributed throughout the body by the bloodstream. As these radionuclides are transported in the blood, they irradiate the entire body. Usually, they then redeposit in one or more organs, causing increased irradiation of that organ. Insoluble particulate materials, on the other hand, are only partially absorbed into body fluids. (This fraction is typically assumed to be about 8 percent.) This absorption occurs over a period of years, with a portion entering the bloodstream and another retained in the pulmonary lymph nodes. The balance (92 percent) of inhaled insoluble particulate materials are removed from the lung within a few days by passing up the air passages to the pharynx where they are swallowed. Inhaled insoluble particulate materials thus irradiate both the lung and the gastrointestinal tract, with a small fraction being eventually absorbed into the bloodstream (TG-66). These nonuniform distributions of dose (and therefore risk) are taken into account through use of the weighting factors for calculating effective dose.

There is a latent period associated with the onset of radiation-induced cancers, so the increased risk is not immediately apparent. The increased risk is assumed to commence 2 to 10 years after the time of exposure and continue the remainder of the exposed individual's lifespan (NA-80).

For uniform exposure of the whole body, about 50 percent of radiation-induced cancers in women and about 65 percent in men are fatal (NA-80). Therefore, 1 rem of low-LET radiation would be expected to cause a total of about 500 cancer cases if delivered to a population of one million. (In the case of thyroid and skin, the ratio of nonfatal to fatal cancers are much higher. These are addressed separately below.) This corresponds to an average annual individual probability of developing cancer of about 7×10^{-6} per year. For perspective, the average annual risk of dying of cancer from all causes in the United States, in 1982, was 1.9×10^{-3}.

B.4.1.1 Cancer Risk Due to Radiation Exposure of the Thyroid

Exposure of the thyroid to extremely high levels of radiation may cause it to degenerate. At moderate levels of exposure some loss of thyroid function will occur. At lower levels of exposure, there are delayed health

(footnote continued)

and Nagasaki indicate that these risk estimates may be revised upwards significantly in the near future, particularly for acute exposure situations. EPA has recently used a slightly higher value, 4×10^{-4} fatalities in standards for air emissions under the Clean Air Act. We will revise these risk estimates to reflect new results following appropriate review.

effects, which take the form of both thyroid nodules and thyroid malignancies (NA-72; NA-80). Doses as low as 14 rad to the thyroid have been associated with thyroid malignancy in the Marshall Islanders (CO-70). The increased risk of radiation-induced cancer is assumed to commence about 10 years after initial exposure and to continue for the remaining lifespan of an exposed individual.

The true nature of thyroid nodules cannot be established until they are surgically removed and examined histologically, and those that are malignant can lead to death if not surgically removed (SA-68; DE-73; PA-74). Although thyroid malignancies are not necessarily fatal, effects requiring surgical removal of the thyroid cannot be considered benign. In this analysis, all thyroid cancers, both fatal and nonfatal, are counted for the purpose of estimating the severity of thyroid exposures.

Based on findings in BEIR-3, we estimate that 1 rem of thyroid exposure carries a risk of producing a thyroid cancer of 3.6×10^{-4}, of which a small fraction (on the order of 1 in 10) will be fatal (NA-80). Since the calculation of effective dose equivalent does not include consideration of nonfatal thyroid cancers and the severity of the medical procedures for their cure, it is appropriate to limit the dose to the thyroid by an additional factor beyond that provided by the PAG expressed in terms of effective dose equivalent. Protective action to limit dose to thyroid is therefore recommended at a thyroid dose 5 times the numerical value of the PAG for effective dose.

B.4.1.2 Cancer Risk Due to Radiation Exposure of the Skin

The risk of fatal skin cancer is estimated to be on the order of one percent of the total risk of fatal cancer for uniform irradiation of the entire body (IC-78). However, since the weighting scheme for calculating effective dose equivalent does not include skin, the PAG expressed in terms of effective dose does not provide protection against radionuclides which primarily expose skin. As in the case of the thyroid, the ratio of nonfatal to fatal cancers from irradiation of the skin is high (on the order of 100 to 1). It would not be appropriate to ignore this high incidence of nonfatal skin cancers by allowing 100 times as much dose to the skin as to the whole body. For this reason, evacuation is recommended at a skin dose 50 times the numerical value of the PAG for effective dose.

B.4.1.3 Cancer Risk Due to Radiation Exposure of the Fetus

The fetus is estimated to be 5 to 10 times as sensitive to radiogenic cancer as an adult (FA-73; WH-65). Stewart reports increased relative incidence of childhood cancers following prenatal x-ray doses as low as 0.20 to 0.25 rem and doubling of childhood cancers between 1-4 rem (ST-73). She concluded that the fetus is about equally sensitive to cancer induction in each trimester. Her findings are supported by similar results reported by MacMahon and Hutchinson (MA-64), Kaplan (KA-58), Polhemus and Kock (PO-59), MacMahon (MA-63), Ford, et al. (FO-59), Stewart and Kneale (ST-70b), and an AEC report (AE-61). MacMahon reported that although there were both positive and negative findings, the combination of weighted data indicates a 40 percent increase in childhood cancer mortality after in vivo exposure to diagnostic x rays (1.0 to 5.0 rad): about 1 cancer per 2,000 exposed children in the first 10 years after birth (MA-63). He concluded that although the range of dose within which these effects are observed is wide, effects will be fewer at 1 rad than at 5 rad.

Graham, et al., investigating diagnostic x-ray exposure, found a significantly increased relative risk of leukemia in children: by a factor of 1.6 following preconception irradiation of mothers or in utero exposure of the fetus; by a factor of 2 following postnatal irradiation of the children; and by a factor of 2 following preconception irradiation of the mother and in utero exposure of the child (GR-66).

B.4.1.4 Age Dependence of Doses

Almost all dose models are based on ICRP "Reference Man," which adopts the characteristics of male and female adults of working age. ICRP-30 dosimetric models, which use "Reference Man" as a basis, are therefore appropriate for only adult workers and do not take into account differences in dose resulting from the differences in physiological parameters between children and adults, e.g., intake rates, metabolism, and organ size. Although it is difficult to generalize for all radionuclides, in some cases these differences tend to counterbalance each other. For example, the ratio of volume of air breathed per unit time to lung mass is relatively constant with age, so that the ICRP adult model for inhaled materials provides a reasonably good estimate of the dose from a given air concentration of radioactive material throughout life.

The thyroid is an exception because the very young have a relatively high uptake of radioiodine into a gland that is much smaller than the adult thyroid (see Section B.4.2.2.). This results in a larger childhood dose and an increased risk which persists throughout life. We have examined this worst case situation. Age-specific risk coefficients for fatal thyroid cancer (See Table 6-8 of "Risk Assessment Methodology" (EP-89)) are about 1.9 higher per unit dose for persons exposed at ages 0 to 9 years than for the general population. Age-dependent dose factors (see NRPB-R162 (GR-85)) for inhalation of I-131, are a factor of about 1.7 higher for 10 year olds than for adults. Therefore, the net risk of fatal thyroid cancer from a given air concentration of I-131 is estimated to be a factor of about 3 higher for young children than for the remainder of the population. This difference is not considered large enough, given the uncertainties of exposure estimation for implementing protective actions, to warrant establishing age-dependent PAGs.

B.4.2 Genetic Risk

An average parental dose of 1 rem before conception has been estimated to produce 5 to 75 significant genetically-related disorders per million liveborn offspring (NA-80). For this analysis we use the geometric mean of this range, i.e. 1.9×10^{-5}. This estimate applies to effects in the first generation only, as a result of dose to parents of liveborn offspring. The sum of effects over all generations is estimated to be approximately twelve times greater; that is, 2.3×10^{-4}. In addition, since any radiation dose delivered after a parent's last conception has no genetic effect, and not all members of the population become parents, less than half of the entire dose in an average population is of genetic significance. Taking the above factors into account, we estimate that the risk of genetically-related disorders in all generations is 1×10^{-4} per person-rem to a typical population.

Although the overall severity of the genetic effects included as "significant" in the above estimates is not well known, rough judgements can be made. The 1980 BEIR report referred to "....disorders and traits that cause a serious handicap at some time during lifetime" (NA-80). From the types of defects reported by Stevenson (ST-59), it can be estimated that, of all radiation-induced genetic effects, 50 percent lead to minor to moderate medical problems (i.e., hair or ear anomalies, polydactyl, strabismus, etc.), 25 percent lead to severe medical problems (i.e., congenital cataracts, diabetes insipidus, deaf mutism, etc.), 23 percent would require extended hospitalization (i.e., mongolism, pernicious anemia, manic-depressive psychoses, etc.), and 2 percent would die before age 20 (i.e., anencephalus, hydrocephalus, pancreatic fibrocytic disease, etc.).

B.4.3 Summary of Risks of Delayed Effects

Table B-4 summarizes average lifetime risks of delayed health effects based on results from the above discussion. Because of the nature of the dose-effect relationships assumed for delayed health effects from radiation (linear, nonthreshold), there is no dose value below which no risk can be assumed to exist.

B.4.4 Risks Associated with Other Radiation Standards

A review of radiation standards for protection of members of the general population from radiation shows a range of values spanning several orders of magnitude. This occurs because of the variety of bases (risk, cost, practicability of implementation, and the situations to which they apply) that influenced the choice of these standards. Some source-specific standards are relatively protective, e.g., the EPA standard limiting exposure of the public from nuclear power operations (25 mrem/y) from all path-

Table B-4 Average Risk of Delayed Health Effects in a Population[a]

	Effects per person-rem		
	Whole Body	Thyroid[c]	Skin
Fatal cancers	2.8E-4[b]	3.6E-5	3.0E-6
Nonfatal cancers	2.4E-4[b]	3.2E-4	3.0E-4
Genetic disorders (all generations)	1.0E-4		

[a] We assume a population with the same age distribution as that of the U.S. population in 1970.

[b] Risk to the fetus is estimated to be 5 to 10 times higher.

[c] Risk to young children is estimated to be about two to three times as high.

ways combined corresponds to a risk (for cancer death) of $5x10^{-4}$ for lifetime exposure. Similarly, regulations under the Clean Air Act limit the dose due to emissions of radionuclides to air alone from all DOE and NRC facilities to 0.01 rem per year, which corresponds to a cancer risk of $2x10^{-4}$ for lifetime exposure. Other guides permit much higher risks. For example, the level at which the EPA recommends action to reduce exposure to indoor radon (0.02 working levels) corresponds to a risk of about $2x10^{-2}$ (for fatal lung cancer) for lifetime exposure. All of these standards and guides apply to nonemergency situations and were based on considerations beyond a simple judgement of acceptable risk.

Federal Radiation Protection Guidance for nonemergency situations recommends that the dose from all sources combined (except from exposure to medical and natural background radiation) to individuals in the population not exceed 0.5 rem in a single year (FR-60) and that the dose to the fetus of occupationally-exposed mothers not exceed 0.5 rem during the 9-month gestation period (EP-87). This dose corresponds to an annual incremental risk of fatal cancer to members of the general population of about $1.4x10^{-4}$. If exposure of the fetus is limited to one ninth of 0.5 rem per month over a 9-month gestation period, as recommended, the risk of severe mental retardation in liveborn is limited to about $7x10^{-4}$.

The International Commission on Radiation Protection recommends that the dose to members of the public not exceed 0.5 rem per year due to nonrecurring exposure to all sources of radiation combined, other than natural sources or beneficial medical uses of radiation (IC-77). They also recommend a limiting dose to members of the public of 0.1 rem per year from all such sources combined for chronic (i.e., planned) exposure (IC-84a). These upper bounds may be taken as representative of acceptable values for the situations to which they apply. That is, these are upper bounds of individual risk that are acceptable for the sum of all sources and exposure pathways under international recommendations, for circumstances that are justified on the basis of public benefit, and when actual doses from individual sources are "as low as reasonably achievable" (ALARA) within these upper bounds.

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APPENDIX C

Protective Action Guides for the Early Phase:

Supporting Information

Contents

Figures

Tables

Tables (continued)

APPENDIX C

Protective Action Guides for the Early Phase: Supporting Information

C.1 Introduction

This appendix sets forth supporting information for the choice of Protective Action Guides (PAGs) for the early phase of the response to a nuclear incident involving the release of airborne radioactive material. It then describes application of the basic principles for selection of response levels set forth in Chapter 1 to the guidance on evacuation and sheltering in Chapters 2 and 5.

Response to a radiological emergency will normally be carried out in three phases, as discussed in Chapter 1. Decisions during the first (early) phase will usually be based on predicted or potential radiological conditions in the environment, rather than on actual measurements. The principal protective action is evacuation, with sheltering serving as a suitable alternative under some conditions. This appendix examines the potential magnitudes and consequences of predicted exposures of populations during the early phase, for selected nuclear reactor accident scenarios, in relation to the benefits and detrimental consequences of evacuation and sheltering. Nuclear reactor facilities are chosen for evaluation because, due to their number, size of source, and energy available to drive a release, they are most likely to provide an upper bound on the magnitude of the variety of possible sources of nuclear incidents. Although atmospheric releases from other types of nuclear incidents are likely to involve smaller consequences, the affected populations, and therefore the costs and benefits of protective action are each expected to scale in roughly the same proportion for lesser magnitude incidents. Thus, basic conclusions developed for responses to reactor facilities are assumed to remain valid for other types of nuclear incidents. Supplementary protective actions, such as washing and change of clothing to reduce exposure of the skin and use of stable iodine to reduce uptake of radioiodine to the thyroid, are also considered, but in less detail.

C.1.1 Existing Federal Guidance

In the 1960's, the Federal Radiation Council (FRC) defined PAGs and established limiting guides for ingestion of strontium-89, strontium-90, cesium-137, and iodine-131 (FR-64; FR-65). That guidance applied to restricting the use of food products that had become contaminated as the result of release of radioactivity to the stratosphere from weapons testing. During the period immediately following an incident at any domestic nuclear facility, when the critical source of exposure is expected to be a nearby airborne plume, the principal protective actions are evacuation or sheltering. The PAGs developed here thus do not supersede previous guidance, but provide additional guidance for prompt exposure pathways specific to a domestic nuclear incident.

C.1.2 Principal Exposure Pathways

The immediate exposure pathway from a sub-stratospheric airborne release of radioactive materials is direct exposure from the cloud of radioactive material carried by prevailing winds. Such a plume can contain radioactive noble gases, iodines, and/or particulate materials, depending on the source involved and conditions of the incident. These materials emit gamma rays, which are not significantly absorbed by air, and will expose the entire bodies of nearby individuals.

Another immediate exposure pathway occurs when people are submerged in the cloud of radioactive materials. In this case radioactive materials are inhaled, and the skin and clothes may be contaminated. Inhaled radioactive materials, depending on their solubility in body fluids, may either remain in the lungs or move via the blood to other organs. Many radionuclides which enter the bloodstream tend to be predominantly concentrated in a single organ. For example, if radioiodines are inhaled, a significant fraction will tend to move rapidly from the lungs through the bloodstream to the thyroid gland where much of the iodine will be deposited and most of the dose[1] will be delivered. Although dose to skin from materials deposited on the skin and clothing could be significant, it will be less important in terms of risk of fatal cancer than dose from inhalation, if early protective actions include washing of exposed skin and changes of clothing.

As the plume passes over an area, radioactive materials may settle onto the ground and other surfaces. People remaining in the area will then continue to be exposed through ingestion and external radiation, and through inhalation of resuspended materials. The total dose from such deposited materials may be more significant than that due to direct exposure to the plume, because the term of exposure can be much longer. However, since the protective actions considered here (evacuation and/or sheltering) may not be appropriate or may not apply for this longer term exposure, doses from these exposures beyond the early phase are not included in the dose considered in the PAGs for the early phase. It is assumed that, within four days after an incident, the population will be

[1]In this and all subsequent references, the word "dose" means the committed dose equivalent to the specified organ, or; if no organ is specified, the sum of the committed effective dose equivalent from intake of radionuclides and the effective dose equivalent from external sources of radiation. (Section B.1.1 contains a more detailed discussion of units of dose for PAGs.)

protected from these subsequent doses on the basis of the PAGs for relocation and for contaminated food and water. (See Chapters 3 and 4.)

Based on the foregoing considerations, the PAGs for the early phase are expressed in terms of estimated doses from exposure due to external radiation, inhalation, and contamination of the skin only during the early phase following an incident.

C.2 Practicality of Implementation

Whereas Appendix B deals with the risk associated with the projected dose that could be avoided by any protective action, this section addresses the costs and risks associated with evacuation itself. That is, these analyses relate to Principles 3 and 4 for deriving PAGs, set forth in Chapter 1, which address the practicality of protective actions, rather than acceptability of risks under Principles 1 and 2, which is evaluated in Appendix B.

The principal relevant protective actions during the early phase are, as noted earlier, evacuation and sheltering. In some cases, washing and changing of clothing, or thyroid blocking may also be appropriate actions. The costs, risks, and degrees of protection associated with evacuation are generally higher than those for sheltering. Although there may be some costs and risks associated with the other protective actions, they are small and not readily quantifiable. Therefore, only the costs and risks associated with evacuation will be evaluated here. These factors are evaluated to determine whether the costs are low enough to justify lower PAGs than would be required to satisfy upper bounds of acceptable risk under Principles 1 and 2.

C.2.1 Cost of Evacuation

Costs incurred to reduce the radiation risk from nuclear incidents can be considered to fall into several major categories. The first category includes the design, construction, and operation of nuclear facilities in such a manner as to minimize the probability and consequences of radiological incidents. It is recognized that the probability and consequences of such incidents usually cannot be reduced to zero. Therefore, a second category is necessary: the development of emergency response plans to invoke actions which would reduce exposure of potentially exposed populations, and consequently their risks, if a major nuclear incident should occur.

Both of the above categories of cost are properly attributed to the cost of design and operation of a nuclear facility. A third category of costs is the actual expenses incurred by taking protective actions as the result of an incident. In general, the choice of levels for PAGs will affect only this third category of costs. That is, all costs in the first two categories are assumed to be unaffected by decisions on the levels of PAGs. (This will be the case unless the PAGs were to be set so high as to never require protective action, in which case response plans would be unnecessary.) Therefore, the costs associated with implementing the PAGs are evaluated only in terms of the actual cost of response. In a similar manner, the risk incurred by protective actions is compared only to the risk associated with the radiation dose that would be avoided by the action, and is unaffected by any other measures taken to reduce risks that fall in the first two categories of cost identified above.

C.2.1.1 Cost Assumptions

The analyses in this section are based on evaluation of the costs of evacuation and the doses that would be received in the absence of protective actions for nuclear reactor incidents. These were calculated as a function of offsite location, meteorological condition, and incident type (TA-87). Dose and cost data are based on the following assumptions:

1. Airborne releases are those associated with fuel melt accidents at nuclear reactor facilities followed by containment failure.

2. Meteorological conditions range from stable to unstable, and windspeeds are those typical of the stability class.

3. Plume dispersion follows a Gaussian distribution, with a 0.01 m/s dry deposition velocity for iodine and particulate materials.

4. Doses are those incurred from whole body gamma radiation from the plume, inhalation of radioactive material in the plume, and from four days exposure to deposited radioactive material.

5. Population distributions are the average values observed around 111 nuclear power reactor plants, based on 1970 data.

6. The cost of evacuation is $185 per person for a 4-day evacuation involving a 100-mile round trip, with an average of 3 persons per household. These evacuation costs include wages and salaries of personnel directing the evacuation, transportation costs of evacuees to and from the staging location, food and shelter for the evacuees during the evacuation period, loss of personal and corporate income during the evacuation period, and the costs of any special supplies (TA-87).

The estimated costs and doses avoided are based on the following idealized evacuation area model (see Figure C.1.):

1. All people within a 2-mile radius of the incident are evacuated for all scenarios.

2. People are also evacuated from a downwind area bounded by equivalent rays on either side of the center line of the plume, which define the angular spread (70, 90, or 180 degrees) of the area evacuated by an arc at the distance beyond which the evacuation dose would not be exceeded on the plume centerline.

Figure C-1 shows the relationship between the area in which the evacuation dose would be exceeded and the larger area that might be evacuated. The figure shows the plume centered in an idealized evacuation area.

C.2.1.2 Analysis

Evaluation of costs for evacuation and doses to populations as a function of the area evacuated depends on a variety of assumptions. Three fuel-melt accident categories, six meteorological stability classes, and the three evacuation area models discussed above were examined. Detailed assumptions and data are reported elsewhere (EP-87a). Selected data, including the cost per unit of collective dose to the population Figure C.1 (person-rem) avoided, are presented in Tables C-1, C-2, and C-3, for three stability classes, for the median nuclear accident category examined (SST-2). (SST accident categories are described in Section E.1.2).

The data are presented for both the total area and the incremental area evacuated for each change in dose level examined. When evaluating the cost per person-rem avoided for a specific set of circumstances, it is appropriate to assess the ratio of the total cost to the total dose avoided to calculate the average cost per person-rem avoided. However, when one is comparing the cost versus dose avoided to make a judgment between a variety of different limiting dose values, it is appropriate to compare the dose savings and costs at the margin, since the cost of evacuating the additional area is incurred to avoid the incremental collective dose. Therefore, the appropriate quantities are the cost and risk for the additional area evacuated. Results of analyses on both a total and incremental basis are presented in Tables C-1, C-2, and C-3 for accident category SST-2. This is the smallest category of fuel melt accident yielding effective dose equivalents during the first 4 days of exposure that are greater than 0.5 rem outside the assumed 2-mile evacuation circle for all stability classes. Data on costs versus dose saved for all three accident categories are summarized in Table C-4 in the next section.

Changes in population density would not affect the above results, since both cost and collective dose are proportional to the size of the population affected. Factors that could affect these results are different assumptions for cost of evacuation, accident scenarios, and evacuation area models. The results will be directly proportional to different assumptions for the cost of evacuation. Some data on the variation with accident scenario are presented in the next section. In situations where different widths of evacuation area are assumed, the change in cost per unit dose avoided will be approximately proportional to the change in width in degrees. This approximation is more accurate for the higher stability classes (E and F). Evacuation within a 2 mile radius circle and a 90 degree sector in the downwind direction is generally considered to be adequate for release

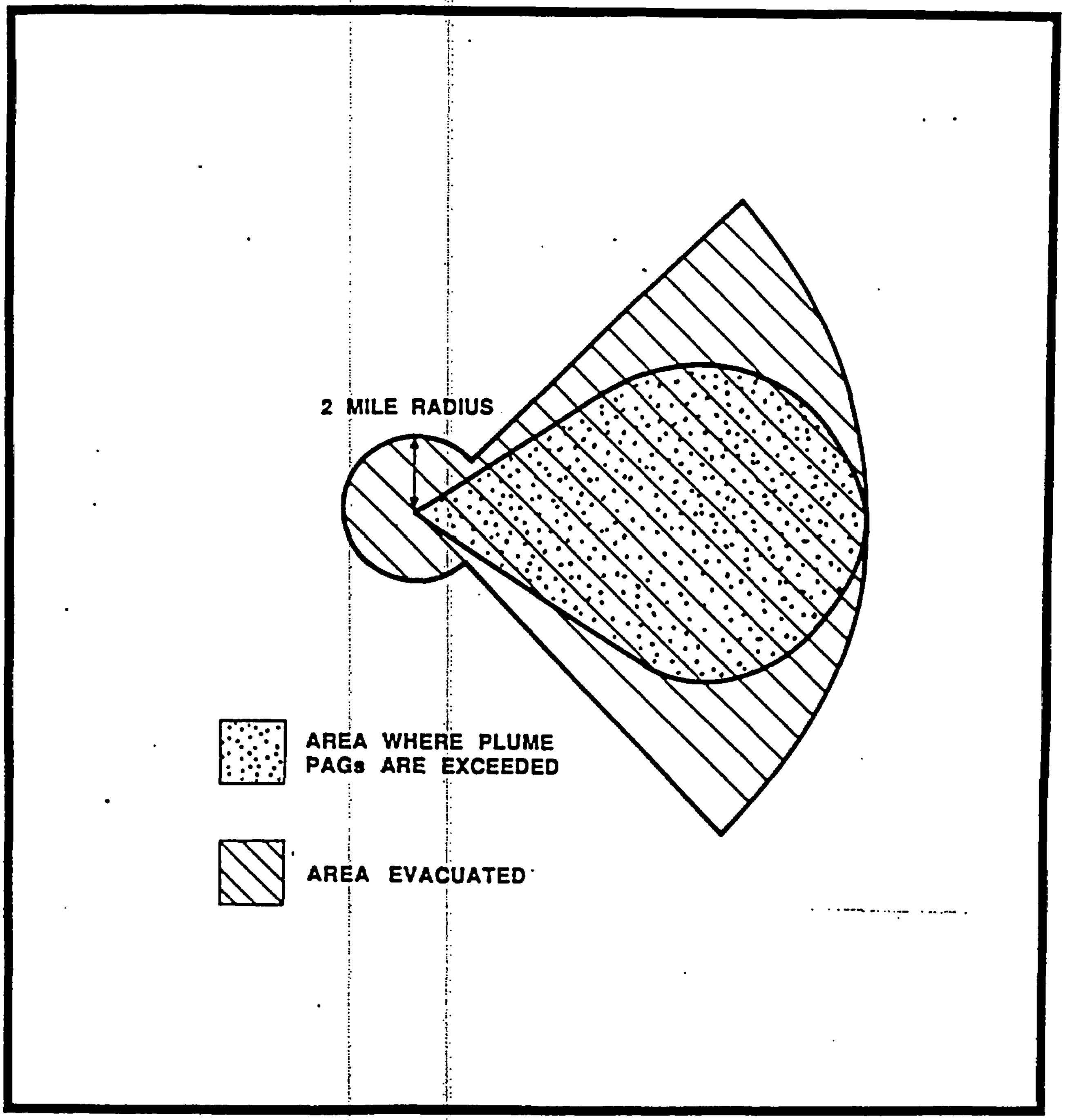

FIGURE C-1. EVACUATION MODEL.

Table C-1 Costs for Implementing Various PAGs for an SST-2 Type Accident (Stability Class A)

Evacuation angle (degrees)	PAG value (rem)	Total Area			Marginal Area		
		Cost (dollars)	Dose avoided (person-rem)	Dollars/ person-rem avoided	Δ Cost (dollars)	Δ Dose avoided (person-rem)	Δ Dollars/ Δperson-rem avoided
70	0.5	2.83E+7	8.97E+4	315	2.16E+7	4.91E+4	440
	1	6.68E+6	4.06E+4	164	5.19E+6	2.33E+4	223
	2	1.49E+6	1.73E+4	88	1.19E+6	1.21E+4	98
	5	2.99E+5	5.22E+3	57	9.70E+4	2.44E+3	40
	10	(a)	(a)	(a)			
90	0.5	3.63E+7	9.29E+4	391	2.78E+7	5.05E+4	550
	1	8.54E+6	4.24E+4	201	6.68E+6	2.42E+4	276
	2	1.86E+6	1.82E+4	102	1.54E+6	1.28E+4	120
	5	3.26E+5	5.41E+3	60	1.25E+5	2.63E+3	47
	10	(a)	(a)	(a)			
180	0.5	7.16E+7	9.33E+4	767	5.49E+7	5.06E+4	1080
	1	1.67E+7	4.27E+4	391	1.32E+7	2.43E+4	543
	2	3.48E+6	1.84E+4	190	3.04E+6	1.29E+4	235
	5	4.48E+5	5.46E+3	82	2.47E+5	2.68E+3	92
	10	(a)	(a)	(a)			

[a] The 4-day dose does not exceed the PAG outside the 2-mile radius of the accident site. The total cost of evacuation within this radius is 2.02E+5 dollars; the total dose avoided is 2.78E+3 person-rem; and the total cost per person-rem avoided is $73.

Table C-2 Costs for Implementing Various PAGs for an SST-2 Type Accident (Stability Class C)

Evacuation angle (degrees)	PAG value (rem)	Total Area: Cost (dollars)	Total Area: Dose avoided (person-rem)	Total Area: Dollars/ person-rem avoided	Marginal Area: Δ Cost (dollars)	Marginal Area: Δ Dose avoided (person-rem)	Marginal Area: Δ Dollars/ Δperson-rem avoided
70	0.5	4.95E+7	1.13E+5	439	3.71E+7	4.95E+4	750
	1	1.23E+7	6.31E+4	195	9.87E+6	2.58E+4	382
	2	2.46E+6	3.73E+4	66	1.68E+6	1.02E+4	165
	5	7.82E+5	2.71E+4	29	3.89E+5	6.15E+3	63
	10	3.93E+5	2.10E+4	19	1.32E+5	4.75E+3	28
	20	2.60E+5	1.62E+4	16	3.40E+4	2.50E+3	10
	50	(a)	(a)	(a)			
90	0.5	6.35E+7	1.13E+5	564	4.77E+7	4.95E+4	964
	1	1.58E+7	6.32E+4	250	1.27E+7	2.58E+4	491
	2	3.11E+6	3.74E+4	83	2.16E+6	1.02E+4	212
	5	9.48E+5	2.72E+4	35	5.00E+5	6.16E+3	81
	10	4.47E+5	2.10E+4	21	1.70E+5	4.76E+3	36
	20	2.77E+5	1.63E+4	17	3.40E+4	2.50E+3	14
	50	(a)	(a)	(a)			
180	0.5	1.25E+8	1.13E+5	1110	9.44E+7	4.95E+4	1910
	1	3.10E+7	6.32E+4	491	2.51E+7	2.58E+4	971
	2	5.95E+6	3.74E+4	159	4.28E+6	1.02E+4	419
	5	1.68E+6	2.72E+4	62	9.90E+5	6.16E+3	161
	10	6.87E+5	2.10E+4	33	3.36E+5	4.77E+3	70
	20	3.51E+5	1.63E+4	22	6.70E+4	2.50E+3	27
	50	(a)	(a)	(a)			

[a] The 4-day dose does not exceed the PAG outside the 2-mile radius of the accident site. The total cost of evacuation within this radius is 2.02E+5 dollars; the total dose avoided is 2.78E+3 person-rem; and the total cost per person-rem avoided is $73.

Table C-3 Costs for Implementing Various PAGs for an SST-2 Type Accident (Stability Class F)

Evacuation angle (degrees)	PAG value (rem)	Total Area: Cost (dollars)	Total Area: Dose avoided (person-rem)	Total Area: Dollars/ person-rem avoided	Marginal Area: Δ Cost (dollars)	Marginal Area: Δ Dose avoided (person-rem)	Marginal Area: Δ Dollars/ Δ person-rem avoided
70	0.5	8.95E+7	4.61E+5	194	4.01E+7	1.98E+4	2020
	1	4.95E+7	4.41E+5	112	2.12E+7	2.17E+4	977
	2	2.83E+7	4.19E+5	67	1.59E+7	3.66E+4	436
	5	1.23E+7	3.83E+5	32	5.65E+6	2.93E+4	193
	10	6.68E+6	3.53E+5	19	3.03E+6	3.18E+4	95
	20	3.65E+6	3.22E+5	11	9.70E+5	3.10E+4	32
	50	1.49E+6	2.68E+5	5.6			
90	0.5	1.15E+8	4.61E+5	250	5.15E+7	1.98E+4	2600
	1	6.35E+7	4.41E+5	144	2.72E+7	2.17E+4	1260
	2	3.63E+7	4.19E+5	87	2.05E+7	3.66E+4	560
	5	1.58E+7	3.83E+5	41	7.26E+6	2.93E+4	248
	10	8.54E+6	3.53E+5	24	3.90E+6	3.18E+4	123
	20	4.64E+6	3.22E+5	14	1.30E+6	3.10E+4	41
	50	1.86E+6	2.68E+5	6.9			
180	0.5	2.27E+8	4.61E+5	493	1.02E+8	1.99E+4	5120
	1	1.25E+8	4.41E+5	285	5.39E+7	2.17E+4	2480
	2	7.16E+7	4.19E+5	171	4.05E+7	3.66E+4	1110
	5	3.10E+7	3.83E+5	81	1.44E+7	2.92E+4	492
	10	1.67E+7	3.53E+5	47	7.71E+6	3.18E+4	242
	20	8.98E+6	3.22E+5	28	2.40E+6	3.10E+4	80
	50	3.51E+6	2.68E+5	13			

durations not exceeding a few hours and where reliable wind direction forecasts are available.

C.2.1.3 Conclusions

As shown in Tables C-1, C-2, and C-3 for an SST-2 accident, the cost per unit dose avoided is greatest for wide angle evacuation and for the most stable conditions, class (F). Although a few emergency plans call for evacuation over wider angles (up to 360 degrees), the model shown in Figure C-1 with a 90 degree angle is most common.

To estimate an upper bound on dose for evacuation based on cost, we first consider common values placed on avoiding risk. As one input into its risk management decisions, EPA has used a range of $400,000 to $7,000,000 as an acceptable range of costs for avoiding a statistical death from pollutants other than radiation. For a risk of $3x10^{-4}$ cancer deaths per person-rem (see Appendix B), these dollar values are equivalent to a range of from about $120 to $2,000 per person-rem avoided. These values can be compared to the marginal cost-effectiveness (dollars per person-rem) of evacuation over an angle of 90 degrees. The resulting ranges of upper bounds on dose are shown in Table C-4 for SST-1, SST-2, and SST-3 accident scenarios. The maximum upper bounds (based on minimum costs for avoiding risk) range from 1 to 10 rem, with most values being approximately 5 rem. The minimum upper bounds (based on maximum costs for avoiding risk) range from 0.15 to 0.8 rem, with 0.5 rem being representative of most situations. From these data we conclude that, based on the cost of evacuation, a PAG larger than the range of values 0.5 to 5 rem would be incompatible with Principle 3.

C.2.2 Risk of Evacuation

Principle 4 requires that the risk of the protective action not exceed the risk associated with the dose that will be avoided. Risk from evacuation can come from several sources, including (1) transportation incidents for both pedestrians and vehicle passengers, (2) exposure to severe weather conditions or a competing disaster, and (3), in the case of immobile persons, anxiety, unusual activity, and separation from medical care or services. The first source, transportation incidents, is the only category for which the risk has been quantified. An EPA report (HA-75) evaluated the risk of transportation fatalities associated with emergency evacuations that have actually occurred and concluded that the risk of death per mile traveled is about the same as that for routine automobile travel. Using this as a basis, the risk of death from travel is about $9x10^{-8}$ deaths per person-mile, or $9x10^{-6}$ deaths per person for the 100-mile round trip assumed for evacuation. Assuming a risk of fatal cancer from radiation of approximately $3x10^{-4}$ per person-rem, such an evacuation risk is equivalent to a dose of about 0.03 rem.

Table C-4 Upper Bounds on Dose for Evacuation, Based on the Cost of Avoiding Fatalities[a]

Accident Category	Atmospheric Stability Class	Dose Upper Bounds[b,c] Maximum (rem)	Minimum (rem)
SST-1	A	5	0.4
	C	5	0.4
	F	10	0.8
SST-2	A	1	0.15
	C	3.5	0.25
	F	10	0.7
SST-3	A	(d)	(d)
	C	(d)	(d)
	F	5	0.45

[a] Based on data from EP-87a.

[b] Windspeeds typical of each stability class were chosen.

[c] Based on an assumed range of $400,000 to $7,000,000 per life saved.

[d] For stability classes A and C, the dose from an SST-3 accident is not predicted to exceed 0.5 rem outside a 2-mile radius. It is assumed that evacuation inside this radius would be carried out based on the emergency condition on the site. No differential evacuation costs were calculated within this area.

In comparing this risk (or, more exactly, its equivalent in dose) to the risk avoided by evacuation, it is important to note that protective action must be implemented over a larger population than will actually be exposed at the level of the PAG. Because of uncertainty or unpredictable changes in wind direction, the exact location of the plume will not be precisely known. Dose projections are made for the maximum exposed individuals - those at the assumed location of the plume centerline. To assure that these individuals will be protected, it is necessary that others on either side take protective action at exposures that are less than at the plume centerline, and, in some cases, are zero. Thus, the entire evacuated population could incur, on the average, a risk from the protective action which exceeds the risk of the radiation dose avoided. Although it is not possible to assure that no individuals incur risks from evacuation greater than their radiation risks, we can assure that this does not occur, on the average, at the outer margin of the evacuation area. For this reason, we also examined the average dose avoided for the incrementally evacuated population for various choices of evacuation levels. Table C-5 presents the results, which are derived from the data in Tables C-1, C-2, and C-3. For the levels analyzed, the average dose avoided is always significantly greater than 0.03 rem. We conclude, therefore, that the choice of PAGs will not be influenced by Principle 4, for persons in the general population whose risk from evacuation is primarily the normal risk of transportation, if the centerline dose avoided is at or above 0.5 rem.

As previously discussed, hazardous environmental conditions (e.g., severe weather or a competing disaster) could create transportation risks from evacuation that would be higher than normal. It is therefore appropriate to make an exception to allow higher projected doses for evacuation decisions under these circumstances. In the absence of any definitive information on such higher risks from evacuation, we have arbitrarily assumed that it would be appropriate to increase the recommended projected dose for evacuation of the general population under hazardous environmental conditions up to a factor of 5 higher than that used under normal environmental conditions.

Table C-5 Average Dose Avoided per Evacuated Individual for Incremental Dose Levels for Evacuation

Centerline dose (rem)	Average dose avoided (rem per individual) by stability class		
	A	C	F
0.5 to 1	0.34	0.19	0.07
1 to 2	0.67	0.38	0.15
2 to 5		0.87	0.33
5 to 10			0.75

It is also recognized that those persons who are not readily mobile are at higher risk from evacuation than are average members of the population. It would be appropriate to adopt higher PAGs for evacuation of individuals who would be at greater risk from evacuation itself than for the typically healthy members of the population, who are at low risk from evacuation. In the absence of definitive information on the higher risk associated with the evacuation of this group, we have arbitrarily assumed that it is appropriate to adopt PAGs a factor of five higher for evacuation of high risk groups under normal environmental conditions. If both conditions exist, (high risk groups and hazardous environmental conditions) projected doses up to 10 times higher than the PAGs for evacuation of the general population under normal conditions may be justified. These doses are expected to satisfy Principle 4 without violating Principles 1 and 3. Although they violate Principle 2, Principle 4 becomes, for such cases, the overriding consideration.

C.2.3 Thyroid Blocking

The ingestion of stable potassium iodide (KI) to block the uptake of radioiodine by the thyroid has been identified as an effective protective action. The Food and Drug Administration (FDA) analyzed available information on the risk of radioiodine-induced thyroid cancers and the incidence and severity of side effects from potassium iodide (FD-82). They concluded "...risks from the short-term use of relatively low doses of potassium iodide for thyroid blocking in a radiation emergency are outweighed by the risks of radioiodine-induced thyroid nodules or cancer at a projected dose to the thyroid gland of 25 rem. FDA recommends that potassium iodide in doses of 130 milligrams (mg) per day for adults and children above 1 year and 65 mg per day for children below 1 year of age be considered for thyroid blocking in radiation emergencies in those persons who are likely to receive a projected radiation dose of 25 rem or greater to the thyroid gland from radioiodines released into the environment. To have the greatest effect in decreasing the accumulation of radioiodine in the thyroid gland, these doses of potassium iodide should be administered immediately before or after exposure. If a person is exposed to radioiodine when circumstances do not permit the immediate administration of potassium iodide, the initial administration will still have substantial benefit even if it is taken 3 or 4 hours after acute exposure". Evacuation and sheltering are, however, preferred alternatives for most situations because they provide protection for the whole body and avoid the risk of misapplication of potassium iodide.

The Federal Emergency Management Agency has published a Federal policy developed by the Federal Radiological Preparedness Coordinating Committee regarding the use of KI as a protective action (FE-85). In summary, the policy recommends the stock-piling of KI and distribution during emergencies to emergency workers and institutionalized persons, but does not recommend requiring stockpiling or distribution to the general public. The policy recognizes, however, that options on the distribution and use of KI rests with the States and, hence, the policy statement permits State and local governments, within the limits of their authority, to take measures beyond those recommended or required nationally.

C.2.4 Sheltering

Sheltering means staying inside a structure with doors and windows closed and, generally, with exterior ventilation systems shut off. Sheltering in place (i.e. at or near the location of an individual when the incident occurs) is a low-cost, low-risk protective action that can provide protection with an efficiency ranging from almost 100 percent to zero, depending on the circumstances. It can also be particularly useful to assure that a population is positioned so that, if the need arises, communication with the population can be carried out expeditiously. The degree of protection provided by a structure is governed by attenuation of radiation by structural components (the mass of walls, ceilings, etc.) and by its outside/inside air-exchange rate. These two protective characteristics are considered separately.

The protection factor may be characterized by a dose reduction factor (DRF), defined as:

$$\text{DRF} = \frac{\text{dose with protective action}}{\text{dose without protective action}}$$

The shielding characteristics of most structures for gamma radiation can be categorized based on whether they are "small" or "large." Small structures are primarily single-family dwellings, and large structures include office, industrial, and commercial buildings. The typical attenuation factors given in Table C-6 show the importance of the type of structure for protection from external gamma radiation (EP-78a). If the structure is a wood frame house without a basement, then sheltering from gamma radiation would provide a DRF of 0.9; i.e., only 10 percent of the dose would be avoided. The DRFs shown in Table C-6 are initial values prior to infiltration of contaminated air, and therefore apply only to short duration plumes. The values will increase with increasing time of exposure to a plume because of the increasing importance of inside-outside air exchange. However, this reduction

Table C-6 Representative Dose Reduction Factors for External Radiation

Structure	DRF	Effectiveness (percent)
Wood frame house (first floor)	0.9	10
Wood frame house (basement)	0.6	40
Masonry house	0.6	40
Large office or industrial building	0.2 or less	80 or better

in efficiency is not dramatic for source terms involving primarily gamma radiation, because most of the dose arises from outside, not from the small volume of contaminated air inside a shelter. Therefore, most shelters will retain their efficiency as shields against gamma radiation, even if the concentration inside equals the concentration outside.

The second factor is the inside/outside air exchange rate. This factor primarily affects protection against exposure by inhalation of airborne radionuclides with half lives long compared to the air exchange rate. The factor is expressed as the number of air exchanges per hour, L (h^{-1}), or the volume of fresh air flowing into and out of the structure per hour divided by the volume of the structure. Virtually any structure that can be used for sheltering has some degree of outside/inside air exchange due to natural ventilation, forced ventilation, or uncontrollable outside forces, primarily wind.

Assuming constant atmospheric and source conditions and no effects from filtration, deposition, or radioactive decay, the following model can be used to estimate the buildup of indoor concentration of radioactivity, for a given outdoor concentration, as a function of time after appearance of the plume and of ventilation rate:

$$C_i = C_o(1 - e^{-Lt}),$$

where C_i = concentration inside,

C_o = concentration outside,

L = ventilation rate (h^{-1}), and

t = elapsed time (h).

Typical values for ventilation rates range from one-fifth to several air exchanges per hour. In the absence of measurements, an air exchange rate of 1.0/h may be assumed for structures with no special preparation except for closing the doors and windows. An air exchange rate of 0.3/h is appropriate for relatively air-tight structures, such as well-sealed residences, interior rooms with doors chinked and no windows, or large structures with ventilation shut off. Using the above model to calculate indoor concentration relative to outdoor concentration after one, two, and four complete air exchanges, the indoor concentration would be about 64 percent, 87 percent, and 98 percent of the outside concentration, respectively. It is apparent that staying in a shelter for more time than that required for one or two complete air exchanges is not very effective for reducing inhalation exposure.

The inhalation DRF is equal to the ratio of the average inside to outside air concentration over the period of sheltering. Studies have been conducted of typical ventilation rates for dwellings (EP-78a) and for large commercial structures (GR-86). In each case the rate varies according to the air tightness of the structure, windspeed, and the indoor-to-outdoor temperature difference. For the purpose of deriving PAGs, average ventilation rates were chosen for the two types of structures that are of greatest interest. Table C-7 shows calculated dose reduction factors for inhalation exposure as a function of plume duration, for beta-gamma source terms, assuming average ventilation rates for these structures.

A potential problem with sheltering is that persons may not leave the shelter as soon as the plume passes and, as a result, will receive exposure from radioactive gases trapped inside. The values for DRFs tabulated in Table C-7 ignore this potential additional contribution. This effect is generally minor for gamma dose (generally less than a 10 percent increase in the dose received during plume passage, (EP-78b)), but can be greater for inhalation dose.

Doses from inhalation during sheltering can be reduced in several ways, including reducing air exchange rates by sealing cracks and openings with cloth or weather stripping, tape, etc., and filtering the inhaled air with commonly available items like wet towels and handkerchiefs. Analyses for some hypothesized accidents, such as short-term transuranic releases, show that sheltering in residences and other

Table C-7 Dose Reduction Factors for Sheltering from Inhalation of Beta-Gamma Emitters

Ventilation rate (air changes/h)	Duration of plume exposure(h)	DRF
0.3[a]	0.5	0.07
	1	0.14
	2	0.25
	4	0.41
	6	0.54
1.0[b]	0.5	0.21
	1	0.36
	2	0.56
	4	0.75
	6	0.83

[a]Applicable to relatively "airtight" structures such as well- sealed residences, interior rooms with chinked doors and no windows, or large structures with outside ventilation shut off.

[b]Applicable to structures with no special preparation except for closing of doors and windows.

buildings can be more effective than for beta-gamma emitters, may provide adequate protection, and may be more effective than evacuation when evacuation cannot be completed before plume arrival (DO-90). However, sheltering effectiveness for the inhalation exposure pathway can be reduced drastically by open windows and doors or by forced air ventilation. Therefore, reliance on protection assumed to be afforded based on large dose reduction factors for sheltering should be accompanied by cautious examination of possible failure mechanisms, and, except in very unusual circumstances, should not be relied upon at projected doses greater than 10 rem. Such analysis should be based on realistic or "best estimate" dose models and include consideration of unavoidable dose if evacuation were carried out.

C.3 Recommended PAGs for Exposure to a Plume during the Early Phase

The four principles which form the basis for the selection of PAG values are presented in Chapter 1. The risks of health effects from radiation that are relevant to satisfying Principles 1 and 2 are presented in Appendix B and analyses of the costs and risks associated with evacuation relative to Principles 3 and 4 have been presented in this appendix. These results, for application to the early phase, are summarized in Table C-8.

The following describes how these results lead to the selection of the PAGs. Conformance to Principle 1 (avoidance of acute health effects) is assured by the low risk required to satisfy Principle 2, and thus requires no additional consideration. Principle 2 (acceptable risk of delayed health effects) leads to the choice of 0.5 rem as an upper bound on the avoided dose below which evacuation of the general population is justified under normal conditions. This represents a risk of about 2E-4 of fatal cancer. Maximum lifetime risk levels considered acceptable by EPA from routine operations of individual sources range from 1E-6 to 1E-4. Risk levels that are higher than this must be justified on the basis of the emergency nature of a situation. In this case, we judge that up to an order of magnitude higher combined risk from all phases of an incident may be justifiable. The choice of 0.5 rem avoided dose as an appropriate criterion for an acceptable level of risk during the early phase is a subjective judgment that includes consideration of possible contributions from exposure during other phases of the incident, as well as the possibility that risk estimates may increase moderately in the near future as a result of current reevaluations of radiation risk.

Principle 4 (risk from the protective action must be less than that from the radiation risk avoided) supplies a lower bound of 0.03 rem on the dose at which evacuation of most members of the public is justified. Finally, under Principle 3 (cost/risk considerations) evacuation is justified only at values equal to or greater than 0.5 rem. This will be limiting unless lower values are required for purely health-based

Table C-8 Summary of Considerations for Selecting the Evacuation PAGs.

Dose (rem)	Consideration	Principle	Section
50	Assumed threshold for acute health effects in adults.	1	B.2.1.4
10	Assumed threshold for acute health effects in the fetus.	1	B.2.1.4
5	Maximum acceptable dose for normal occupational exposure of adults.	2	C.5
5	Maximum dose justified to average members of the population, based on the cost of evacuation.	3	C.2.1.3
0.5	Maximum acceptable dose to the general population from all sources from nonrecurring, non-accidental exposure.	2	B.4.4
0.5	Minimum dose justified to average members of the population, based on the cost of evacuation.	3	C.2.1.3
0.5	Maximum acceptable dose[a] to the fetus from occupational exposure of the mother.	2	C.5
0.1	Maximum acceptable dose to the general population from all sources from routine (chronic), nonaccidental exposure.	2	B.4.4
0.03	Dose that carries a risk assumed to be equal to or less than that from evacuation.	4	C.2.2

[a]This is also the dose to the 8- to 15-week-old fetus at which the risk of mental retardation is assumed to be equal to the risk of fatal cancer to adults from a dose of 5 rem.

reasons (Principle 2). But this is not the case. The single lower purely health-based value, 0.1 rem, is only valid as a health-based criterion for chronic exposure.

In summary, we have selected the value 0.5 rem as the avoided dose which justifies evacuation, because 1) it limits the risk of delayed effects on health to levels adequately protective of public health under emergency conditions, 2) the cost of implementation of a lower value is not justified, and 3) it satisfies the two bounding requirements to avoid acute radiation effects and to avoid increasing risk through the protective action itself. We note that this choice also satisfies the criterion for acceptable risk to the fetus of occupationally exposed mothers (as well as falling well below dose values at which abortion is recommended).

As noted in Section C.2.4, we assume that the dose normally avoidable by evacuation (the dose that is not avoided by the assumed alternative of sheltering) is one half of the projected dose. The value of the PAG for evacuation of the general public under normal circumstances is therefore chosen as one rem projected sum of the committed effective dose equivalent from inhalation of radionuclides and effective dose equivalent from exposure to external radiation.

The above considerations apply to evacuation of typical members of the population under normal circumstances, and apply to effective doses (i.e. the weighted sum of doses to all organs). As discussed in previous sections, it may be appropriate to further limit dose to the thyroid and skin, to adjust the value for special groups of the population at unusually high risk from evacuation, and to provide for situations in which the general population may be at higher than normal risk from evacuation. These are addressed, in turn, below.

In the case of exposure of the thyroid to radioiodine, action based solely on effective dose would not occur until a thyroid dose about 33 times higher than the corresponding effective dose to the entire body. As noted in Section B.4.1.1, because the weighting factor for thyroid used to calculate effective dose does not reflect the high ratio of curable to fatal thyroid cancers, protective action to limit dose to the thyroid is recommended at a thyroid dose 5 times the numerical value of the PAG.

Similarly, since effective dose does not include dose to the skin, and for other reasons discussed in Section B.4.1.2, protective action to limit dose to skin is recommended at a skin dose 50 times the numerical value of the PAG. As in the case of the thyroid, this includes consideration of the risk of both curable and noncurable cancers.

Special risk groups include fetuses, and persons who are not readily mobile. As noted in Sections B.4.1.3 and B3, we assume that the risk of radiation-induced cancer is about 5 to 10 times higher for fetuses than for adults and that the risk of mental retardation in fetuses exposed during the 8th to 15th weeks of gestation is about 10 times higher than the risk of fatal cancer in equivalently exposed adults. However, due to the difficulty of rapidly evacuating only pregnant women in a population, and the assumed higher-than-average risk associated with their evacuation, it is not considered appropriate to establish separate PAGs for pregnant women. We note that the PAG is chosen sufficiently low to satisfy Federal guidance for limiting exposure of the fetus in pregnant workers.

Higher PAGs for situations involving higher risks from evacuation were discussed in Section C.2.2. Under normal, low-risk, environmental conditions, PAGs for evacuation of groups who present higher than average risks from evacuation (e.g., persons who are not readily mobile) are recommended at projected doses up to 5 rem. Evacuation of the general population under high-risk environmental conditions is also recommended at projected doses up to 5 rem. If evacuation of high risk groups under hazardous environmental conditions is being considered, projected doses up to 10 rem may, therefore, be justified.

Short-term sheltering is recognized as a low-cost, low-risk, protective action primarily suited for protection from exposure to an airborne plume. Sheltering will usually be clearly justified to avoid projected doses above 0.5 rem, on the basis of avoidance of health risks. However, data are not available to establish a lower level at which sheltering is no longer justified because of its cost or the risk associated with its implementation. Sheltering will usually have other benefits related to emergency communication with members of the public. It is expected that protective action planners and decision authorities will take into account the added benefits of sheltering (e.g., communication and established planning areas) for decisions on sheltering at levels below 0.5 rem.

Bathing and changing of clothing are effective for reducing beta dose to the skin of persons exposed to an airborne plume of radioactive materials. Since these are also low-cost, low-risk actions, no PAG is recommended for initiating their implementation. It is expected that any persons exposed in areas where evacuation is justified based on projected dose from inhalation will be routinely advised by emergency response officials to take these actions within 12 hours after exposure.

The use of stable iodine to protect against uptake of inhaled radioiodine by the thyroid is recognized as an effective alternative to evacuation for situations involving radioiodine releases where evacuation cannot be implemented. If procedures are included in the applicable emergency response plan, use of stable iodine should be considered for any such situation in which evacuation or sheltering will not be effective in preventing thyroid doses of 25 rem (see also C.2.3).

C.4 Comparison to Previous PAGs

This section compares the level of protection provided by the previously published PAGs for evacuation (one rem external gamma dose from the plume and 5 rem committed dose to the thyroid from inhalation, under normal evacuation circumstances) with this PAG. The effective dose addressed by this PAG, as well as skin, thyroid, and external gamma doses from the plume during the early phase from the three major exposure pathways for an airborne release were calculated for radionuclide mixes postulated for three nuclear power plant accident sequences. The doses were then normalized for each accident so that they represent a location in the environment where the controlling dose would be equal to the current PAG. These results are shown in Table C-9.

Based on the results shown in Table C-9, the following conclusions are

Table C-9. Comparison of Projected Doses for Various Reactor Accident Scenarios[a]

Accident category[b]	Effective dose equivalent[c] (rem)	Skin dose[d] (rem)	Thyroid dose[e] (rem)	External dose[f] (rem)
SST-1	0.7	6	5	0.02
SST-2	1	5	5	0.4
SST-3	0.4	6	5	0.1

[a]Doses are normalized to the limiting PAG.

[b]See Table E-1 for a description of these accident scenarios.

[c]The dose is the sum of doses from 4-day exposure to external gamma radiation from deposited materials, external exposure to the plume, and the committed effective dose equivalent from inhalation of the plume.

[d]The dose equivalent from external beta radiation from the plume and from 12 hours exposure to materials deposited on skin and clothing.

[e]Committed dose equivalent to the thyroid from inhalation.

[f]External gamma dose equivalent from the plume.

apparent, for the accident sequences analyzed:

1. The PAG for the thyroid is controlling for all three accident categories. For the SST-2 category, effective dose is also controlling. Thus,application of the previous PAG (5 rem) for thyroid would provide the same protection as the revised PAG for all three accident categories.

2. Skin doses will not be controlling for any of the accident sequences (if bathing and change of clothing is completed within 12 hours of plume passage, as assumed).

3. Gamma dose from direct exposure to the plume is small compared to the effective dose from the three major exposure pathways combined.

In summary, for the accident sequences analyzed, the old PAGs provide the same level of protection as the new PAGs. For releases that contain a smaller fraction radioiodines than these accident scenarios the new PAGs are slightly more protective.

C.5 Dose Limits for Workers Performing Emergency Services

Dose limits for workers during emergencies are based on avoiding acute health effects and limiting the risk of delayed health effects, in the context of the need to assure protection of the population and of valuable properties. It is assumed that most such workers are accustomed to accepting an element of risk as a condition of their employment. Examples of occupations that may be affected include law enforcement, firefighting, radiation protection, civil defense, traffic control, health services, environmental monitoring, animal care, and transportation services. In addition, selected workers at utility, industrial, and at farms and other agribusinesses may be required to protect others, or to protect valuable property during an emergency. The above are examples -- not designations -- of workers that may be exposed to radiation during emergencies.

Radiation exposure of workers during an emergency should normally be governed by the Federal Radiation Protection Guidance for Occupational Exposure (EP-87). This guidance specifies an upper bound of five rem committed effective dose equivalent per year for most workers. (Pregnant women, who, under this guidance should not normally engage in work situations that involve more than approximately 50 mrem/month, would normally be evacuated as part of the general population.) The guidance also specifies that doses to workers should be maintained as low as reasonably achievable; that doses should be monitored; and that workers should be informed of the risks involved and of basic principles for radiation protection.

There are some emergency situations, however, for which higher doses may be justified. These include lifesaving operations and the protection of valuable property. International guidance (IC-77) recognizes two additional dose levels for workers under specially justified circumstances: two times the annual limit for any single event, and five times the annual limit in a lifetime. The dose limits recommended here adopt the former value (10 rem) for operations limited to the protection of valuable property. The latter value (25 rem) may be permitted for situations involving lifesaving operations or activities that are essential to preventing substantial risks to populations. In this context "substantial risks" means collective doses that are significantly larger than those incurred through the protective activities engaged in by the workers. Workers should not operate under dose limits higher than five rem unless the following conditions are satisfied:

1. Lower doses through the rotation of workers or other commonly-used dose reduction methods are not possible, and

2. Instrumentation is available to measure their exposure.

In addition to the limitation on effective dose equivalent, the dose equivalent received in any year by workers under normal occupational conditions is limited to 15 rem to the lens of the eye and 50 rem to any other organ, tissue (including skin), or extremity of the body. (Extremity is defined as the forearms and hands or the lower legs and feet (EP-87).) By analogy to these dose limits for organs and extremities, the limits for workers performing the various categories of emergency services are established at numerical values that are 5 times the limits for effective dose to the lens of the eye and 10 times the limits for effective dose to any other organ, tissue (including skin), or extremity of the body.

Situations may occur in which a dose in excess of 25 rem would be required for lifesaving operations. It is not possible to prejudge the risk that one person should be allowed to take to save the life of another. However, persons undertaking an emergency mission in which the dose would exceed 25 rem to the whole body should do so only on a voluntary basis and with full awareness of the risks involved, including the numerical levels of dose at which acute effects of radiation will be incurred and numerical estimates of the risk of delayed effects.

The risk of acute health effects is discussed in B.2. Table C-10 presents estimated cancer mortality rates for a dose of 25 rem, as a function of age at the time of exposure. The risk of cancer from moderately higher doses will increase proportionately. These values were calculated using risk estimates from BEIR-3 (NA-80) as discussed in Section B.4, and life table analyses that assume the period of cancer risk lasts for the worker's lifetime (BU-81). The risk was calculated for the midpoint of each age range. Roughly equivalent risks of nonfatal cancer and serious genetic effects (if gonadal tissue is exposed) will also be incurred.

The dose limits of 75 rem to the whole body previously recommended by EPA and 100 rem that has been recommended by NCRP (GL-57) for lifesaving action represents a very high level of risk of acute and delayed health effects. A dose of 100 rem is expected to result in an approximately 15 percent risk of temporary incapacity from nonlethal acute effects and an indeterminate, but less than 5 percent, chance of death within 60 days. This is in addition to a risk of about 1 in 30 of incurring fatal cancer. Such high risk levels can only be accepted by a recipient who has been made aware of the risks involved. Therefore, no absolute dose limit for lifesaving activities is offered.

Table C-10 Cancer Risk to Emergency Workers Receiving 25 Rem Whole Body Dose

Age of the emergency worker at time of exposure (years)	Approximate risk of premature death (deaths per 1,000 persons exposed)	Average years of life lost if premature death occurs (years)
20 to 30	9.1	24
30 to 40	7.2	19
40 to 50	5.3	15
50 to 60	3.5	11

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APPENDIX D

Background for Protective Action Recommendations: Accidental Radioactive Contamination of Food and Animal Feeds*

*This background document concerning food and animal feeds was published by the Food and Drug Administration in 1982.

APPENDIX E

Protective Action Guides for the Intermediate Phase

(Relocation)

Background Information

Contents

Figures

Tables

Appendix E

Protective Action Guides for the Intermediate Phase (Relocation) Background Information

E.1 Introduction

This Appendix provides background information for the choice of Protective Action Guides (PAGs) for relocation and other protective actions to reduce exposure to deposited radioactive materials during the intermediate phase of the response to a nuclear incident. The resulting PAGs and associated implementing guidance are provided in Chapters 4 and 7, respectively.

This analysis is based on the assumption that an airborne plume of radioactive material has already passed over an area and left a deposit of radioactive material behind, or that such material exists from some other source, and that the public has already been either sheltered or evacuated, as necessary, on the basis of PAGs for the early phase of a nuclear incident, as discussed in Chapters 2 and 5. PAGs for subsequent relocation of the public and other protective actions, as well as dose limits for persons reentering the area from which the public is relocated, are addressed in this Appendix.

We first set forth the assumptions used to derive information pertinent to choosing the dose level at which relocation of the public is appropriate. This is followed by an examination of information relevant to this decision, and selection of the PAG for relocation. The Appendix concludes with a brief discussion of the basis for dose limits for persons temporarily reentering areas from which the public has been relocated.

E.1.1 Response Duration

In order to decide whether to initiate relocation of the public from specific areas it is necessary to predict the dose that would be avoided. One factor in this prediction is the duration of the exposure to be avoided. Relocation can begin as soon as patterns of exposure from deposited radioactivity permit restricted areas to be identified. For the purpose of this analysis, relocation of persons who have not already been evacuated from the restricted zone is assumed to take place on the fourth day after the incident. Return of evacuated persons to their residences outside the restricted zone and transition to relocation status of persons already evacuated is assumed to occur over a period of a week or more.

The period of exposure avoided by relocation ends when the relocated person either returns to his property or is permanently resettled in a new location. At the time of relocation decisions, it will usually not be possible to predict when either of these actions will occur. Therefore, for convenience of dose projection, it is assumed that the period of exposure avoided is one year and that any extension beyond this period will be determined on the basis of recovery criteria. This assumption corresponds to emergency response planning guidance by ICRP (IC-84) and IAEA (IA-85).

E.1.2 Source Term

The "source term" for this analysis is comprised of the quantities and types of particulate radioactive material found in the environment following a nuclear incident. Nuclear incidents can be postulated with a wide range of release characteristics. The characteristics of the source terms assumed for the development of these PAGs are those postulated for releases from various types of fuel-melt accidents at nuclear power plants (SN-82). Table E-1 provides brief descriptions of these accident types. Radionuclide releases have been estimated for the three most severe accident types (SST-1, SST-2, SST-3) based on postulated core inventories and release fractions (Table E-2). The other types (SST-4 and SST-5) would generally not produce offsite doses from exposure to deposited material sufficient to warrant consideration of relocation.

Table E-1 Brief Descriptions Characterizing Various Nuclear Power Plant Accident Types (SN-82)

Type	Description
SST-1	Severe core damage. Essentially involves loss of all installed safety features. Severe direct breach of containment.
SST-2	Severe core damage. Containment fails to isolate. Fission product release mitigating systems (e.g., sprays, suppression pool, fan coolers) operate to reduce release.
SST-3	Severe core damage. Containment fails by base-mat melt-through. All other release mitigation systems function as designed.
SST-4	Modest core damage. Containment systems operate in a degraded mode.
SST-5	Limited core damage. No failures of engineered safety features beyond those postulated by the various design basis accidents. Containment is assumed to function for even the most severe accidents in this group.

Table E-2 Release Quantities for Postulated Nuclear Reactor Accidents

Principal radionuclides contributing to dose from deposited materials	Half-life (days)	Estimated quantity released[a] (Curies)		
		SST-1	SST-2	SST-3
Zr-95	6.52E+1	1.4E+6	4.5E+4	1.5E+2
Nb-95	3.50E+1	1.3E+6	4.2E+4	1.4E+2
Ru-103	3.95E+1	6.0E+6	2.4E+5	2.4E+2
Ru-106	3.66E+2	1.5E+6	5.8E+4	5.8E+1
Te-132	3.25	8.3E+7	3.9E+6	2.6E+3
I-131	8.05	3.9E+7	2.6E+5	1.7E+4
CS-134	7.50E+2	8.7E+6	1.2E+5	1.3E+2
CS-137	1.10E+4	4.4E+6	5.9E+4	6.5E+1
Ba-140	1.28E+1	1.2E+7	1.7E+5	1.7E+2
La-140	1.67	1.5E+6	5.1E+4	1.7E+2

[a]Based on the product of reactor inventories of radionuclides and estimated fractions released for three accident categories (SN-82).

For other types of source terms, additional analysis may be necessary to assure adequate protection. For example, if the release includes a large proportion of long-lived radionuclides, doses will continue to be delivered over a long period of time, and, if no remedial actions are taken, the dose delivered in the first year may represent only a small portion of the total dose delivered over a lifetime. On the other hand, if the release consists primarily of short-lived radionuclides, almost the entire dose may be delivered within the first year.

From the data in Table E-2, it is apparent that, for the groups of accidents listed, both long and short lived radionuclides would be released. Consequently, doses due to deposited materials from such accidents would be relatively high during the first year followed by long term exposures at lower rates.

E.1.3 Exposure Pathways

The principal exposure pathway to members of the public occupying land contaminated by deposits of radioactive materials from reactor incidents is expected to be exposure of the whole body to external gamma radiation. Although it is normally expected to be of only minor importance, the inhalation pathway would contribute additional doses to internal organs. The health risks from other pathways, such as beta dose to the skin and direct ingestion of dirt, are also expected to be minor in comparison to the risks due to external gamma radiation (AR-89). Skin and inhalation dose would, however, be important exposure pathways for source terms with significant fractions of pure beta emitters, and inhalation dose would be important for source terms with significant fractions of alpha emitters.

Since relocation, in most cases, would not be an appropriate action to prevent radiation exposure from ingestion of food and water, these exposure pathways have not been included in this analysis. They are addressed in Chapters 3 and 6. In some instances, however, where withdrawal of food and/or water from use would, in itself, create a health risk, relocation may be an appropriate alternative protective action. In this case, the committed effective dose equivalent from ingestion should be added to the projected dose from deposited radionuclides via other pathways, for decisions on relocation.

E.1.4 Response Scenario

This section defines the response zones, population groups, and the activities assumed for implementation of protective actions during the intermediate phase.

After passage of the radioactive plume, the results of environmental monitoring will become available for use in making decisions to protect the public. Sheltering, evacuation, and other actions taken to protect the public from the plume will have already been implemented. The tasks immediately ahead will be to (1) define the extent and characteristics of deposited radioactive material and identify a restricted zone in accordance with the PAG for relocation, (2) relocate persons from and control access to the restricted zone, (3) allow persons to return to areas outside the restricted zone, (4) control the spread of and exposure to surface contamination, and (5) apply simple decontamination and other low-cost, low-risk techniques to reduce the dose to persons who are not relocated.

Because of the various source term characteristics and the different protective actions involved (evacuation, sheltering, relocation, decontamination, and other actions to reduce doses to "as low as reasonably achievable" levels), the response areas for different protective actions may be complex and may vary in size with respect to each other. Figure E-1 shows a generic example of some of the principal areas involved. The area covered by the plume is assumed to be represented by area 1. In reality, variations in meteorological conditions would almost certainly produce a more complicated shape.

Based on plant conditions or other considerations prior to or after the release, members of the public are assumed to have already been evacuated from area 2 and sheltered in area 3. Persons who were evacuated or

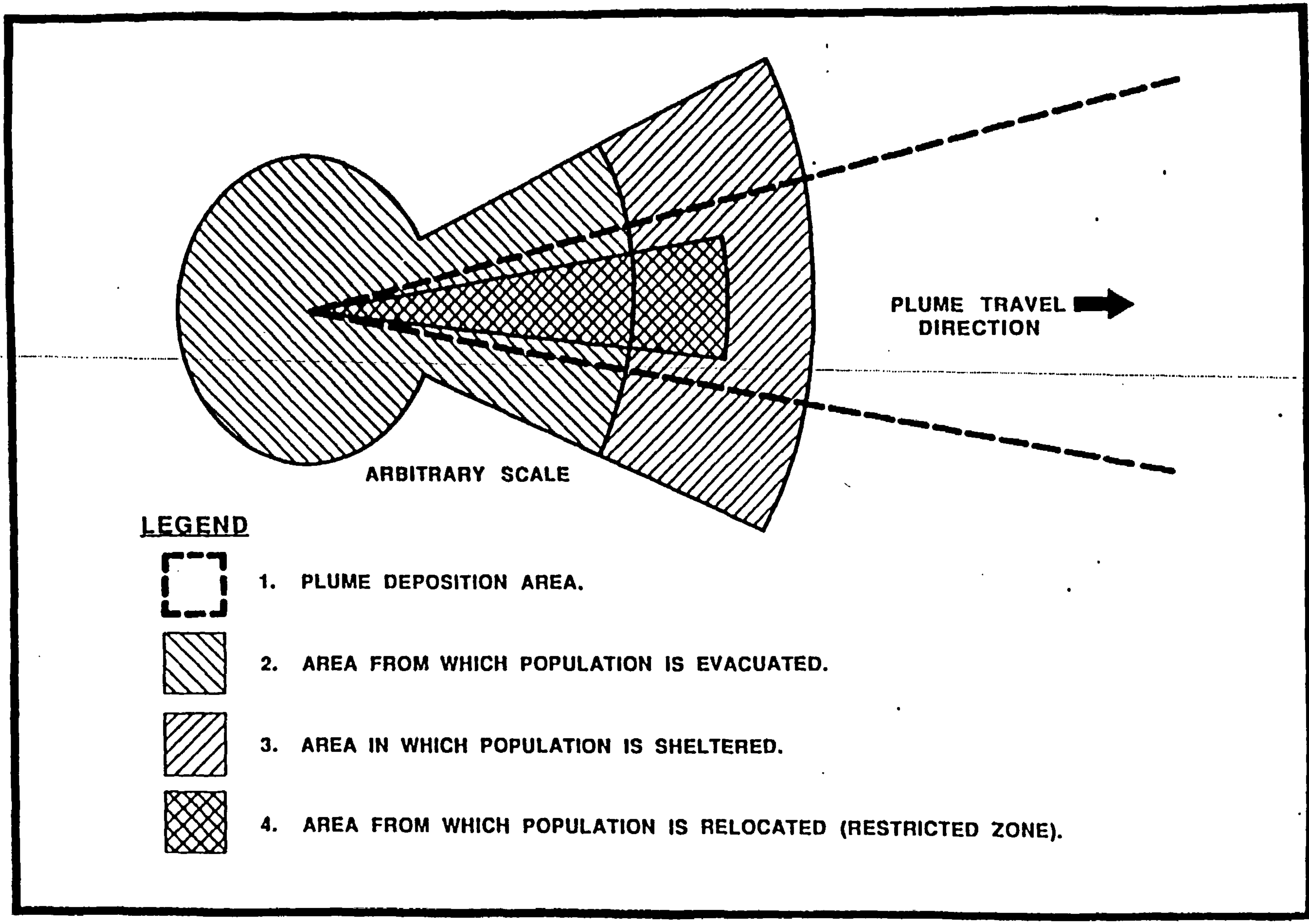

FIGURE E-1. RESPONSE AREAS.

sheltered as a precautionary action for protection from the plume but whose homes are outside the plume deposition area (area 1) are assumed to return to their homes or discontinue sheltering when environmental monitoring verifies the outer boundary of area 1.

Area 4 is the restricted zone and is defined as the area where projected doses are equal to or greater than the relocation PAG. The portion of area 1 outside of area 4 is designated as a study zone and is assumed to be occupied by the public. However, contamination levels may exist here that would be of concern for continued monitoring and decontamination to maintain radiation doses "as low as reasonably achievable" (ALARA).

The relative positions of the boundaries shown in Figure E-1 are dependent on areas evacuated and sheltered. For example, area 4 could fall entirely inside area 2 (the area evacuated) so that relocation of persons from additional areas would not be required. In this case, the relocation PAG would be used only to determine areas to which evacuees could return.

Figure E-2 provides, for perspective, a schematic representation of the response activities expected to be in progress in association with implementation of the PAGs during the intermediate phase of the response to a nuclear incident.

E.2 Considerations for Establishing PAGs for the Intermediate Phase

The major considerations in selecting values for these PAGs for relocation and other actions during the intermediate phase are the four principles that form the basis for selecting all PAGs. Those are discussed in Section E.2.1. Other considerations (Federal radiation protection guidance and risks commonly confronting the public) are discussed in Sections E.2.2 and E.5.

In addition, a planning group consisting of State, Federal, and industry officials provided recommendations in 1982 which EPA considered in the development of the format, nature, and applicability of PAGs for relocation. Abbreviated versions of these recommendations are as follows:

a. The PAGs should apply to commercial, light-water power reactors.

b. The PAGs should be based primarily on health effects.

c. Consideration should be given to establishing a range of PAG values.

d. The PAGs should be established as high as justifiable because at the time of the response, it would be possible to lower them, if justified, but it probably would not be possible to increase them.

e. Only two zones (restricted and unrestricted) should be established to simplify implementation of the PAGs.

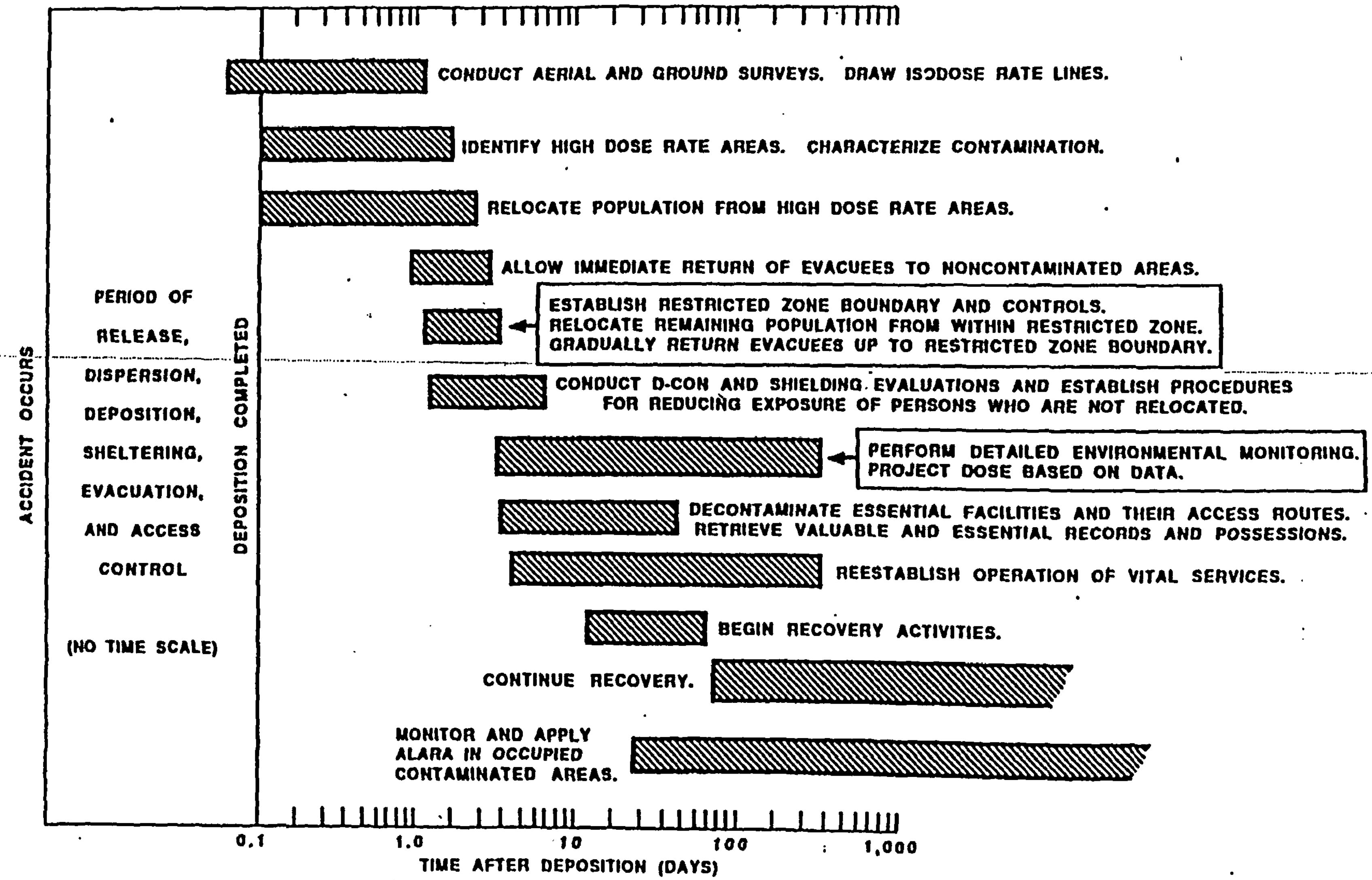

FIGURE E-2 POTENTIAL TIME FRAME OF RESPONSE TO A NUCLEAR INCIDENT.

f. The PAGs should not include past exposures.

g. Separate PAGs should be used for ingestion pathways.

h. PAGs should apply only to exposure during the first year after an incident.

Although these PAGs apply to any nuclear incident, primary consideration was given to the case of commercial U.S. reactors. In general, we have found it possible to accommodate most of the above recommendations.

E.2.1 Principles

In selecting values for these PAGs, EPA has been guided by the principles that were set forth in Chapter 1. They are repeated here for convenience:

1. Acute effects on health (those that would be observable within a short period of time and which have a dose threshold below which they are not likely to occur) should be avoided.

2. The risk of delayed effects on health (primarily cancer and genetic effects, for which linear nonthreshold relationships to dose are assumed) should not exceed upper bounds that are judged to be adequately protective of public health, under emergency conditions, and are reasonably achievable.

3. PAGs should not be higher than justified on the basis of optimization of cost and the collective risk of effects on health. That is, any reduction of risk to public health achievable at acceptable cost should be carried out.

4. Regardless of the above principles, the risk to health from a protective action should not itself exceed the risk to health from the dose that would be avoided.

Appendix B analyzed the risks of health effects as a function of dose (Principles 1 and 2). Considerations for selection of PAGs for the intermediate phase of a nuclear incident differ from those for selection of PAGs for the early phase primarily with regard to implementation factors (i.e., Principles 3 and 4). Specifically, they differ with regard to cost of avoiding dose, the practicability of leaving infirm persons and prisoners in the restricted zone, and avoiding dose to fetuses. Although sheltering is not generally a suitable alternative to relocation, other alternatives (e.g., decontamination and shielding) are suitable. These considerations are reviewed in the sections that follow.

E.2.1.1 Cost/Risk Considerations

The Environmental Protection Agency has issued guidelines for internal use in performing regulatory impact analyses (EP-83). These include consideration of the appropriate range of costs for avoiding a statistical death. The values are inferred from the additional compensation associated with employment carrying a higher than normal risk of mortality and are expressed as a range of $0.4 to $7 million per statistical death avoided. The following discussion compares these values to the cost of avoiding radiation-induced fatal cancers through relocation.

The basis for estimating the societal costs of relocation are analyzed in a report by Bunger (BU-89). Estimated incremental societal costs per day per person relocated are shown below. (Moving and loss of inventory costs are averaged over one year.)

Moving	$1.70
Loss of use of residence	2.96
Maintain and secure vacated property	0.74
Extra living costs	1.28
Lost business and inventories	14.10
Extra travel costs	4.48
Idle government facilities	1.29
Total	$26.55

The quantity of interest is the dose at which the value of the risk avoided is equal to the cost of relocation. Since the above costs are expressed in dollars/person-day, it is convenient to calculate the dose that must be avoided per-person day. The equation for this is:

$$H_E = \frac{C}{VR}$$

where:

H_E = dose
C = cost of relocation
V = value of avoiding a statistical death
R = statistical risk of death from radiation dose

Using the values cited above, and a value for R of 3×10^{-4} deaths/rem (See Appendix B), one obtains a range of doses of about 0.01 to 0.2 rem/day. Thus, over a period of one year the total dose that should be avoided to justify the cost of relocation would be about 5 to 80 rem.

These doses are based on exposure accumulated over a period of one year. However, exposure rates decrease with time due to radioactive decay and weathering. Thus, for any given cumulative dose in the first year, the daily exposure rate continually decreases, so that a relocated person will avoid dose more rapidly in the first part of the year than later. Figure E-3 shows the effect of changing exposure rate on the relationship between the cost of avoiding a statistical death and the time after an SST-2 accident (See Table E-1) for several assumed cumulative annual doses. The curves represent the cost per day divided by the risk of fatality avoided by relocation per day, at time t, for the annual dose under consideration, where t is the number of days after the accident. The right ordinate shows the gamma exposure rate (mR/h) as a function of time for the postulated radionuclide mix at one meter height.

The convex downward curvature results from the rapid decay of short-lived radionuclides during the first few weeks following the accident. Since the cost per day for relocation is assumed to be constant and the dose avoided per day decreases, the cost effectiveness of relocation decreases with time. For this reason it is cost

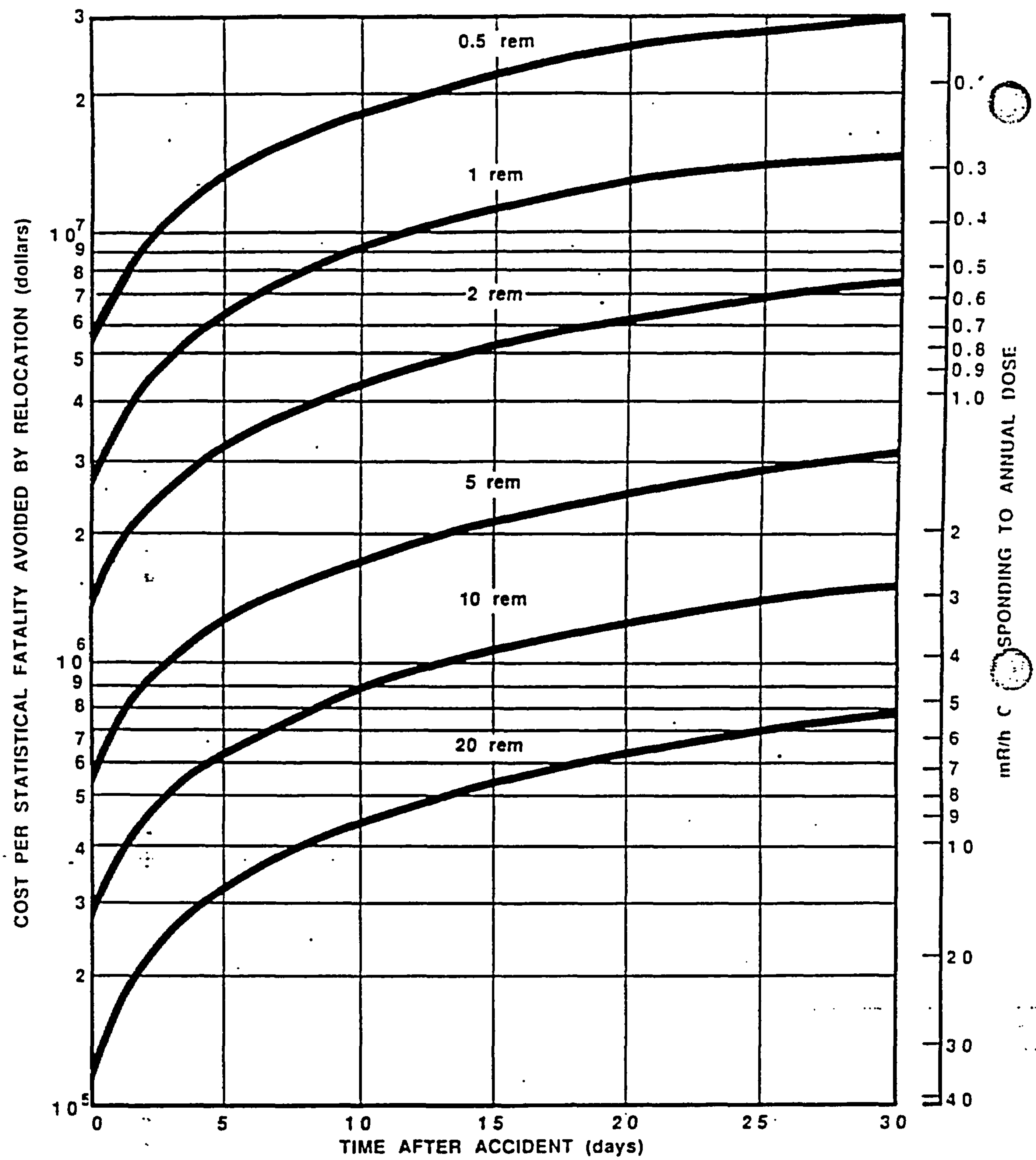

FIGURE E-3. COST OF AVOIDING STATISTICAL FATALITIES AND EXPOSURE RATES CORRESPONDING TO VARIOUS TOTAL FIRST YEAR DOSES (ASSUMES AN SST-2 ACCIDENT AND A $27 PER PERSON-DAY COST OF RELOCATION).

effective to quickly recover areas where the population has been relocated at projected doses only marginally greater than the PAG.

Only trends and general relationships can be inferred from Figure E-3 because it applies to a specific mix of radionuclides. However, for this radionuclide mix, cost analysis supports relocation at doses as low as one rem for the first week and two rem for up to 25 days after an accident.

E.2.1.2 Protection of Special Groups

Contrary to the situation for evacuation during the early phase of an incident, it is generally not practical to leave a few persons behind when most members of the general population have been relocated from a specified area for extended periods of time. Further, no data are available on differing risks of relocation for different population groups. In the absence of such data, we have assumed that these risks will be similar to those from evacuation. Those risks were taken as equivalent to the health risk from doses of 30 mrem for members of the general population and of 150 mrem for persons at high risk from evacuation (see Appendix C). Therefore, to satisfy Principle 4 for population groups at high risk, the PAG for relocation should not be lower than 150 millirem. Given the arbitrary nature of this derivation, it is fortunate that this value is much lower than the PAG selected, and is therefore not an important factor in its choice.

Fetuses are a special group at greater risk of health effects from radiation dose than is the general population, but not at significantly greater risk from relocation itself. The risk of mental retardation from fetal exposure (see Appendix B) is significant. It is affected by the stage of pregnancy relative to the assumed one-year exposure, because the 8th to 15th week critical period during which the risk is greatest, must be considered in relation to the rapidly changing dose rate. Taking these factors into account, it can be postulated that the risk of mental retardation due to exposure of the fetus during the intermediate phase will range from one to five times the cancer risk of an average member of the public, depending upon when conception occurs relative to the time of the incident. The elevated risk of radiation-induced cancer from exposure of fetuses is less significant, as discussed in Appendix B.

It will usually be practicable to reduce these risks by establishing a high priority for efforts other than relocation to reduce the dose in cases where pregnant women reside near the boundary of the restricted zone. However, women who are less than seven months pregnant may wish to relocate for the balance of their pregnancy if the projected dose during pregnancy cannot be reduced below 0.5 rem.

E.2.2 Federal Radiation Protection Guides

The choice of a PAG at which relocation should be implemented does not mean that persons outside the boundary of the restricted zone should not be the subject of other protective actions to reduce dose. Such actions are justified on the basis of existing Federal radiation protection guidance (FR-65) for protecting the public, including implementation of the principle of maintaining doses "as low as reasonably achievable" (ALARA).

The intended actions to protect the public from radiation doses on the basis of Radiation Protection Guides (RPGs) are those related to source control. Although it is reasonable for members of the public to receive higher exposure rates prior to the source term being brought under control, the establishment of acceptable values for relocation PAGs must include consideration of the total dose over the average remaining lifetime of exposed individuals (usually taken as 50 years).

The nationally and internationally recommended upper bound for dose in a single year from man-made sources, excluding medical radiation, is 500 mrem per year to the whole body of individuals in the general population (IC-77, FR-65). These recommendations were not developed for nuclear incidents. They are also not appropriate for chronic exposure. The ICRP recommends an upper bound of 100 mrem per year, from all sources combined, for chronic exposure (IC-77). The corresponding 50-year dose at 100 mrem/yr is 5 rem. We have chosen to limit: a) the projected first year dose to individuals from an incident to the Relocation PAG, b) the projected second year dose to 500 mrem, and c) the dose projected over a fifty-year period to 5 rem. Due to the extended duration of exposures and the short half-life of important radioiodines, no special limits for thyroid dose are needed.

E.3. Dose from Reactor Incidents

Doses from an environmental source will be reduced through the natural processes of weathering and radioactive decay, and from the shielding associated with part time occupancy in homes and other structures. Results of dose calculations based on the radiological characteristics of releases from three categories of postulated, fuel-melt, reactor accidents (SST-1, SST-2, and SST-3) (SN-82) and a weathering model from WASH-1400 (NR-75) are shown in Table E-3. This table shows the relationship between annual doses for the case where the sum, over fifty years, of the effective dose equivalent from gamma radiation and the committed effective dose equivalent from inhalation of resuspended materials is 5 rem. Radioactive decay and weathering reduces the second year dose from reactor incidents to 20 to 40 percent of the first year dose, depending on the radionuclide mix in the release.

Based on studies reported in WASH-1400 (NR-75), the most conservative dose reduction factor for

Table E-3 Annual Doses Corresponding to 5 Rem in 50 Years[a]

Year	Dose According to Accident Category[b] (rem)		
	SST-1	SST-2	SST-3
1	1.25	1.60	1.91
2	0.52	0.44	0.38
3	0.33	0.28	0.24
4	0.24	0.20	0.17
5	0.18	0.16	0.13
6	0.14	0.12	0.11
7	0.12	0.11	0.090
8	0.10	0.085	0.070
9	0.085	0.075	0.065
10	0.080	0.070	0.060
11	0.070	0.060	0.050
12	0.060	0.055	0.050
15	0.055	0.045	0.040
20	0.045	0.040	0.030
25	0.040	0.035	0.025
30	0.030	0.030	0.025
40	0.025	0.020	0.020
50	0.020	0.015	0.010

[a]Whole body dose equivalent from gamma radiation plus committed effective dose equivalent from inhalation assuming a resuspension factor of 10^{-6} m^{-1}. Weathering according to the WASH-1400 model (NR-75) and radioactive decay are assumed.

[b]Radionuclide abundance ratios are based on reactor inventories from WASH-1400 (NR-75). Release quantities for accident categories SST-1, SST-2 and SST-3 are shown in Table E-2. Initial concentrations are assumed to have decayed for 4 days after reactor shutdown.

structures (frame structures) is about 0.4 (dose inside divided by dose outside) and the average fraction of time spent in a home is about 0.7. Combining these factors yields a net dose reduction factor of about 0.6. In most cases, therefore, structural shielding would be expected to reduce the dose to persons who are not relocated to 60 percent (or less) of the values shown in Table E-3 before the application of decontamination.

E.4. Alternatives to Relocation

Persons who are not relocated, in addition to dose reduction provided by partial occupancy in homes and other structures, can reduce their dose by the application of various techniques. Dose reduction efforts can range from the simple processes of scrubbing and/or flushing surfaces, soaking or plowing of soil, removal and disposal of small spots of soil found to be highly contaminated (e.g., from settlement of water), and spending more time than usual in lower exposure rate areas (e.g., indoors), to the difficult and time consuming processes of removal, disposal, and replacement of contaminated surfaces. It is anticipated that simple processes would be most appropriate to reduce exposure rates for persons living in contaminated areas outside the restricted zone. Many of these can be carried out by the residents with support from officials for monitoring, guidance on appropriate actions, and disposal. The more difficult processes will usually be appropriate for recovery of areas from which the population is relocated.

Decontamination experiments involving radioactive fallout from nuclear weapons tests have shown reduction factors for simple decontamination methods in the vicinity of 0.1 (i.e., exposure rate reduced to 10 percent of original values). However, recent experiments at the Riso National Laboratory in Denmark (WA-82, WA-84), using firehoses to flush asphalt and concrete surfaces contaminated with radioactive material of the type that might be deposited from reactor accidents, show decontamination factors for radionuclides chemically similar to cesium that are in the range of 0.5 to 0.95, depending on the delay time after deposition before flushing is applied. The factor for ruthenium on asphalt was about 0.7 and was independent of the delay of flushing. The results of these experiments indicate that decontamination of the important reactor fission products from asphalt or concrete surfaces may be much more difficult than decontamination of nuclear weapons fallout. Other simple dose reduction methods listed above would be effective to varying degrees. The average dose reduction factor for gamma radiation from combinations of simple decontamination methods is estimated to be at least 0.7. Combining this with the 40 percent reduction estimated above for structural shielding indicates that the doses listed in Table E-3 may be more than twice as high as those which would actually be received by persons who are not relocated.

E.5 Risk Comparisons

Many hazardous conditions and their associated risks are routinely faced by the public. A lingering radiation dose will add to those risks, as opposed to substituting one risk for another, and, therefore, radiation protection criteria cannot be justified on the basis of the existence of other risks. It is, however, useful to review those risks to provide perspective. This section compares the risks associated with radiation doses to those associated with several other risks to which the public is commonly exposed.

Figure E-4 compares recent statistics for the average lifetime risk of accidental death in various occupations to the estimated lifetime risk of fatal cancer for members of the general population exposed to radiation doses ranging up to 25 rem. Non-radiation risk values are derived from information in reference (EP-81) and radiation risk values are from Appendix B. These comparisons show, for example, that the lifetime cancer risk associated with a dose of 5 rem is comparable to the lifetime risk of accidental death in some of the safest occupations, and is well below the average lifetime risk of accidental death for all industry.

Risks of health effects associated with radiation dose can also be compared to other risks facing individuals in the general population. The risks listed in Table E-4 are expressed as the number of premature deaths and the average reduction of life-span due to these deaths within a group of 100,000 persons. For purposes of comparison, a dose of 5 rem to each member of a population group of 100,000 persons representative of the average U.S. population carries an estimated lifetime risk of about 150 fatal cancers (see Appendix B). The number of deaths resulting from the various causes listed in Table E-4 is based on data from mortality records.

In summary, the risk of premature death normally confronting the public from specific types of accidents ranges from about 2 to 1000 per 100,000 population. The estimated radiation doses required to produce a similar risk of death from radiation-induced cancer range from about 0.07 to 33 rem.

E.6 Relocation PAG Recommendations

Previous sections have reviewed data, standards, and other information relevant to establishing PAGs for relocation. The results are summarized in Table E-5, in relation to the principles set forth in Section E.2.1.

Based on the avoidance of acute effects alone (Principle 1) 50 rem and 10 rem are upper bounds on the dose at which relocation of the general population and fetuses, respectively, is justified. However, on the basis of control of chronic risks (Principle 2) a lower upper bound is appropriate. Five rem is taken as an upper bound on acceptable risk for controllable <u>lifetime</u> exposure to radiation, including avoidable exposure to accidentally deposited radioactive materials. This corresponds to an average of 100 mrem per year for fifty years, a value commonly accepted as an upper bound for chronic <u>annual</u> exposure of members of the public from all sources of exposure combined, other than natural background and medical radiation (IC-77). In the case of projected doses from nuclear reactor accidents, a five rem lifetime dose corresponds to about 1.25 to 2 rem from exposure during the first year and 0.4 to 0.5 rem from exposure during the second year.

Analyses based on Principle 3 (cost/risk) indicate that considering cost

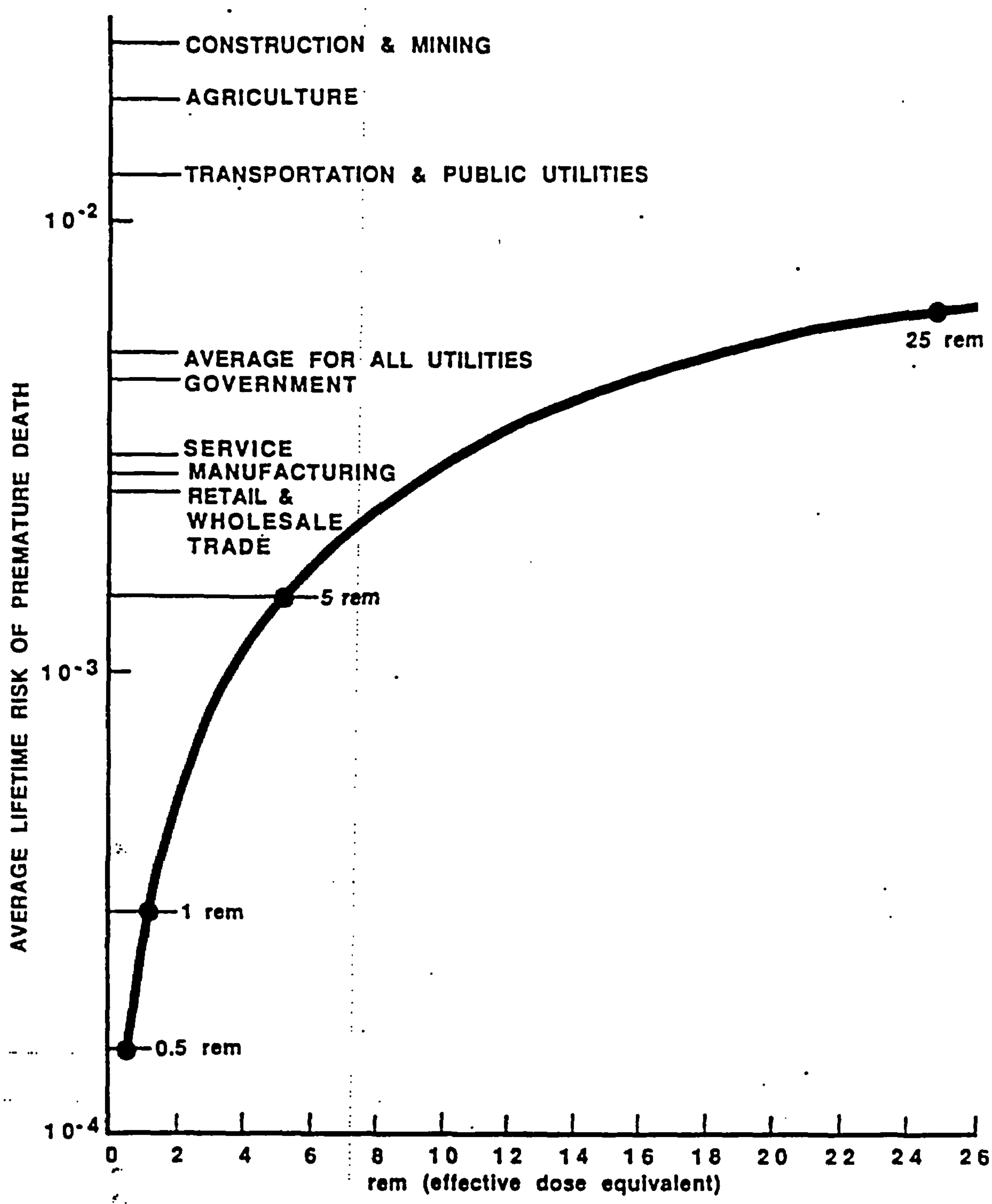

FIGURE E-4. AVERAGE LIFETIME RISK OF DEATH FROM WHOLE BODY RADIATION DOSE COMPARED TO THE AVERAGE RISK OF ACCIDENTAL DEATH FROM LIFETIME (47 YEARS) OCCUPATION IN VARIOUS INDUSTRIES.

Table E-4 Measure of Lifetime Risk of Mortality from a Variety of Causes[a] (Cohort Size = 100,000)

Nature of accident	Premature deaths	Aggregate years of life lost to cohort	Reduction of life expectancy at birth (years)	Average years of life lost to premature deaths
Falls	1,000	12,000	0.12	11
Fires	300	7,600	0.076	26
Drowning	190	8,700	0.087	45
Poisoning by drugs and medicaments	69	2,500	0.025	37
Cataclysm[b]	17	490	0.005	30
Bites and stings[c]	8	220	0.002	27
Electric current in homes[d]	8	290	0.003	37

[a]All mortality effects shown are calculated as changes from the U.S. Life Tables for 1970 to life tables with the cause of death under investigation removed. These effects also can be interpreted as changes in the opposite direction, from life tables with the cause of death removed to the 1970 Life Table. Therefore, the premature deaths and years of life lost are those that would be experienced in changing from an environment where the indicated cause of death is not present to one where it is present. All values are rounded to no more than two significant figures.

[b]Cataclysm is defined to include cloudburst, cyclone, earthquake, flood, hurricane, tidal waves, tornado, torrential rain, and volcanic eruption.

[c]Accidents by bite and sting of venomous animals and insects include bites by centipedes, venomous sea animals, snakes, and spiders; stings of bees, insects, scorpions, and wasps; and other venomous bites and stings. Other accidents caused by animals include bites by any animal and nonvenomous insect; fallen on by horse or other animal; gored; kicked or stepped on by animal; ant bites; and run over by horse or other animal. It excludes transport accidents involving ridden animals; and tripping, falling over an animal. Rabies is also excluded.

[d]Accidents caused by electric current from home wiring and appliances include burn by electric current, electric shock or electrocution from exposed wires, faulty appliances, high voltage cable, live rail, and open socket. It excludes burn by heat from electrical appliances and lighting.

Table E-5 Summary of Considerations for Selecting PAGs for Relocation

Dose (rem)	Consideration	Principle
50	Assumed threshold for acute health effects in adults.	1
10	Assumed threshold for acute health effects in the fetus.	1
6	Maximum projected dose in first year to meet 0.5 rem in the second year[a].	2
5	Maximum acceptable annual dose for normal occupational exposure of adults.	2
5	Minimum dose that must be avoided by one year relocation based on cost.	3
3	Minimum projected first-year dose corresponding to 5 rem in 50 years[a].	2
3	Minimum projected first-year dose corresponding to 0.5 rem in the second year[a].	2
2	Maximum dose in first year corresponding to 5 rem in 50 years from a reactor incident, based on radioactive decay and weathering only.	2
1.25	Minimum dose in first year corresponding to 5 rem in 50 years from a reactor incident based on radioactive decay and weathering only.	2
0.5	Maximum acceptable single-year dose to the general population from all sources from non-recurring, non-incident exposure.	2
0.5	Maximum acceptable dose to the fetus from occupational exposure of the mother.	2
0.1	Maximum acceptable annual dose to the general population from all sources due to routine (chronic), non-incident, exposure.	2
0.03	Dose that carries a risk assumed to be equal to or less than that from relocation.	4

[a]Assumes the source term is from a reactor incident and that simple dose reduction methods are applied during the first month after the incident to reduce the dose to persons not relocated from contaminated areas.

alone would not drive the PAG to values less than 5 rem. Analyses in support of Principle 4 (risk of the protective action itself) provide a lower bound for relocation PAGs of 0.15 rem.

Based on the above, <u>2 rem projected committed effective dose equivalent from exposure in the first year is selected as the PAG for relocation</u>. Implementation of relocation at this value will provide reasonable assurance that, for a reactor accident, a person relocated from the outer margin of the relocation zone will, by such action, avoid an exposure rate which, if continued over a period of one year, would result in a dose of about 1.2 rem. This assumes that 0.8 rem would be avoided without relocation through normal partial occupancy of homes and other structures. This PAG will provide reasonable assurance that persons outside the relocation zone, following a reactor accident, will not exceed 1.2 rem in the first year, 0.5 rem in the second year, and 5 rem in 50 years. The implementation of simple dose reduction techniques, as discussed in section E-4, will further reduce dose to persons who are not relocated from contaminated areas. Table E-6 summarizes the estimated maximum dose that would be received by these persons for various reactor accident categories with and without the application of simple dose reduction techniques. In the case of non-reactor accidents these doses will, in general, differ, and it may be necessary to apply more restrictive PAGs to the first year in order to assure conformance to the second year and lifetime objectives noted above.

Since effective dose does not include dose to the skin (and for other reasons discussed in Appendix B) protective action to limit dose to skin is recommended at a skin dose 50 times the numerical value of the PAG for effective dose. This includes consideration of the risk of both curable and fatal cancers.

E.7 Criteria for Reentry into the Restricted Zone

Persons may need to reenter the restricted zone for a variety of reasons, including radiation monitoring, recovery work, animal care, property maintenance, and factory or utility operation. Some persons outside the restricted zone, by nature of their employment or habits, may also receive higher than average radiation doses. Tasks that could cause such exposures include: 1) changing of filters on air handling equipment (including vehicles), 2) handling and disposal of contaminated vegetation (e.g., grass and leaves) and, 3) operation of control points for the restricted zone.

Individuals who reenter the restricted zone or who perform tasks involving exposure rates that would cause their radiation dose to exceed that permitted by the PAGs should do so in accordance with existing Federal radiation protection guidance for occupationally exposed workers (EP-87). The basis for that guidance has been provided elsewhere (EP-87).

Table E-6 Estimated Maximum Doses to Nonrelocated Persons From Areas Where the Projected Dose is 2 REM[a]

Accident Category	Dose (rem): No additional dose reduction, Year 1	Year 2	50 years	Early simple dose reduction[b], Year 1	Year 2	50 years
SST-1	1.2	0.5	5.0	0.9	0.35	3.5
SST-2	1.2	0.34	3.9	0.9	0.24	2.7
SST-3	1.2	0.20	3.3	0.9	0.14	2.3

[a]Based on relocation at a projected dose of 2 rem in the first year and 40 percent dose reduction to nonrelocated persons from normal, partial occupancy in structures. No dose reduction is assumed from decontamination, shielding, or special limitations on time spent in high exposure rate areas.

[b]The projected dose is assumed to be reduced 30 percent by the application of simple dose reduction techniques during the first month. If these techniques are completed later in the first year, the first year dose will be greater.

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EP-81 U.S. Environmental Protection Agency. Background Report. Proposed Federal Radiation Protection Guidance for Occupational Exposure. EPA 520/4-81-003, U.S. Environmental Protection Agency, Washington (1981).

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SN-82 Sandia National Laboratories. Technical Guidance for Siting Criteria Development. NUREG/CR-2239, U.S. Nuclear Regulatory Commission, Washington (1982).

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WA-84 Warming, L. Weathering and Decontamination of Radioactivity Deposited on on Concrete Surfaces. RISO-M-2473, Riso National Laboratory, DK-4000 Roskilde, Denmark. December (1984).

APPENDIX F

Radiation Protection Criteria for the Late Phase

Background Information

(Reserved)

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<table>
<tr><td colspan="2">WISCONSIN PUBLIC SERVICE CORP.

Kewaunee Nuclear Power Plant

Emergency Plan Implementing Procedure</td><td colspan="2">No. EPIP-AD-02</td><td>Rev. AK</td></tr>
<tr><td colspan="2"></td><td colspan="3">Title Emergency Class Determination</td></tr>
<tr><td colspan="2"></td><td colspan="2">Date OCT 5 2004</td><td>Page 1 of 22</td></tr>
<tr><td colspan="2">Reviewed By John Egdorf</td><td colspan="3">Approved By Jerrie Coleman</td></tr>
<tr><td>Nuclear Safety Related</td><td>☐ Yes
☑ No</td><td>PORC Review Required</td><td>☐ Yes
☑ No</td><td>SRO Approval Of Temporary Changes Required ☐ Yes ☑ No</td></tr>
</table>

1.0 Purpose

1.1 This procedure provides instruction for determining proper emergency classification listed in order to activate the appropriate level of response from the Kewaunee Nuclear Power Plant (KNPP) emergency response organization and off-site response organization.

2.0 General Notes

2.1 None

3.0 Precautions and Limitations

3.1 Plant monitors used to determine whether emergency classification levels are being exceeded should be checked for accuracy prior to declaring an emergency class (e.g., compare against redundant channels, determine if consistent with system status, or verification by sample analysis when required by Chart A(1).

3.2 This procedure is not written to facilitate de-escalation. Therefore, any decision to de-escalate must be based on a thorough review of procedures and plant conditions. If appropriate, it is preferable to terminate or enter recovery. However, there may be occasions where it is appropriate to de-escalate.

3.3 Once indication is available that an emergency action level has been met, classification must be made as soon as possible and must not exceed 15 minutes. A classification should not be made before an emergency action level has been met. Once a classification has been declared, notification must be initiated and in progress to the State and County agencies within 15 minutes of event classification using "Event Notice," Form EPIPF-AD-07-01. During the initial 15-minute classification assessment, there may be rapidly changing conditions. Classification during this initial period should be based on currently available plant status.

4.0 Initial Conditions

4.1 This procedure applies during any plant evolution that may result in an emergency declaration.

5.0 Procedure

5.1 Determine if a plant emergency exists during abnormal plant conditions by referring to Table 2-1, Emergency Action Level Charts.

5.2 IF a plant emergency exists, THEN perform the required actions of the appropriate emergency procedure listed below:

5.2.1 EPIP-AD-03, "KNPP Response to an Unusual Event"

5.2.2 EPIP-AD-04, "KNPP Response to Alert or Higher"

5.3 As plant conditions change, continue referring to the Emergency Action Level Charts.

5.4 Determine if the emergency should be reclassified.

5.5 IF the event is reclassified, THEN return to Step 5.2.

5.6 IF Final Conditions (Section 6.0) are not met, THEN return to Step 5.3.

5.7 IF Final Conditions (Section 6.0) are met, THEN use of this procedure may be suspended.

6.0 Final Conditions

6.1 Plant Emergency has been Terminated or Recovery actions have begun and the Responsible Director has suspended the use of EPIPs.

7.0 References

7.1 Kewaunee Nuclear Power Plant Emergency Plan

7.2 EPIP-AD-01, Personnel Response to the Plant Emergency Siren

7.3 EPIP-AD-03, KNPP Response to an Unusual Event

7.4 EPIP-AD-04, KNPP Response to Alert or Higher

7.5 COMTRAK 89-001, NRC Inspection Report 88-11, Improve Guidance for Fires Chart G

7.6 OEA 87-246, Report OE 2265, Improve Description of Unusual Aircraft Activity Chart P

7.7 NRC Letter 07-11-94, Branch Position on Acceptable Deviations to NUREG-0654

7.8 OEA 96-083, NRC IN 1997-045 Chart A(2)

8.0 Records

8.1 The following QA records and non-QA records are identified in this directive/procedure and are listed on the KNPP Records Retention Schedule. These records shall be maintained according to the KNPP Records Management Program.

8.1.1 QA Records

None

8.1.2 Non-QA Records

None

EMERGENCY ACTION LEVEL CHARTS

The following charts are separated into different abnormal operating conditions which may, depending upon their severity, be classified as an Unusual Event, Alert, Site Emergency, or General Emergency.

	CHART	PAGE
Abnormal Radiological Effluent	A (1)	5
Gaseous Effluent Action Levels	A (2)	6 - 8
Fuel Damage Indication	B	9
Primary Leak to LOCA	C	10
Primary to Secondary Leak	D	11
Loss of Power	E	12
Engineered Safety Feature Anomaly	F	13
Loss of Indication	G	14
DELETED	H	14
Secondary Side Anomaly	I	15
Miscellaneous Abnormal Plant Conditions	J	16
Fire and Fire Protection	K	18
DELETED	L	18
Earthquake	M	19
High Winds or Tornado	N	19
Flood, Low Water, or Seiche	O	20
External Events and Chemical Spills	P	21
Security Contingency	Q	22

CHART A(1)
ABNORMAL RADIOLOGICAL EFFLUENT

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
SEE CHART A(2)	Effluent monitors detect levels corresponding to greater than 1 rem/hr whole body or 5 rem/hr thyroid at the site boundary under "actual meteorological" conditions.	GENERAL EMERGENCY
Projected or measured dose rates to be provided by the Radiological Protection Director or Environmental Monitoring Teams.	Projected or measured in the environs dose rates greater than 1 rem/hr whole body or 5 rem/hr thyroid at the site boundary.	GENERAL EMERGENCY
SEE CHART A(2)	Effluent monitors detect levels corresponding to greater than 50 mr/hr for ½ hour OR greater than 500 mr/hr for two minutes (or five times these levels to the thyroid) OR for "adverse meteorology."	SITE EMERGENCY
Projected or measured dose rates to be provided by the Radiological Protection Director or Environmental Monitoring Teams.	At the site boundary, projected or measured dose rates greater than 50 mr/hr for ½ hours OR greater than 500 mr/hr for two minutes (or five times these levels to the thyroid) or EPA Protective Action Guidelines are projected to be exceeded outside the site boundary.	SITE EMERGENCY
SEE CHART A(2)	Radiological effluents greater than 10 times ODCM instantaneous limits.	ALERT
a. Containment R-2 OR R-7 ≥ 1,000 times baseline established in the Control Room, OR b. Charging Area R-4 ≥ 1,000 times baseline established in the Control Room, OR c. SFP Area R-5 ≥ 1,000 times baseline established in the Control Room, OR d. Plant area air sample indicates airborne contamination > 1,000 times the occupational DAC values.	Radiation levels or airborne contamination which indicate a severe degradation in the control of radioactive materials (e.g., radiation levels suddenly increase by a factor of 1,000).	ALERT
(1) Gaseous Releases: See Chart A(2) (2) Liquid Releases: Notification by the Rad-Chem Group of violating ODCM 3.3.1 limits.	Offsite Dose Calculation Manual limits exceeded.	UNUSUAL EVENT

CHART A(2)
GASEOUS EFFLUENT ACTION LEVELS

1. AUX BUILDING VENT RELEASES - *WITH SIGNIFICANT CORE DAMAGE*

Instrument readings assuming a post accident gas release AND Containment High Range Radiation Monitors 42599 (R-40) and 42600 (R-41) reads 1000 R/hr for > 2 minutes within one-half hour of the accident.

NOTE: Use adverse meteorology conditions (ADV MET) only when, 10m and 60m wind speed < 5mph AND Delta-T > +2.4 degrees F or Sigma Theta is < 3.01 degrees. All other cases are average meteorology (AVG MET).

NOTE: R-13 and R-14 are expected to be off scale high during all events on this page.

SV & SFP FANS	AUX BLDG SPING MONITORS				AUX BLDG STACK MONITORS				EMERG. CLASS.
	MID RANGE CPM (01-07) PPCS PT G9086G		HIGH RANGE CPM (01-09) PPCS PT G9088G		R-35 MR/HR		R-36 R/HR		
TOTAL NUMBER RUNNING	AVG MET	ADV MET	AVG MET	ADV MET	AVG MET	ADV MET	AVG MET	ADV MET	
1	**	1.1E+4	6.5E+1	*	**	7.9E+2	1.27E+2	7.9E-1	GENERAL EMERG.
2	8.8E+5	5.5E+3	3.25E+1	*	**	3.9E+2	6.35E+1	4.0E-1	
3	5.9E+5	3.7E+3	2.16E+1	*	**	2.6E+2	4.2E+1	2.6E-1	
4	4.4E+5	2.7E+3	1.62E+1	*	**	2.0E+2	3.175E+1	2.0E-1	
1	8.8E+4	5.5E+2	3.0E+0	*	6.3E+3	3.9E+1	6.3E+0	*	SITE EMERG.
2	4.4E+4	2.7E+2	1.5E+0	*	3.1E+3	1.9E+1	3.1E+0	*	
3	2.9E+4	1.8E+2	1.0E+0	*	2.1E+3	1.3E+1	2.1E+0	*	
4	2.2E+4	1.3E+2	*	*	1.5E+3	9.5E+0	1.5E+0	*	
1	1.0E+3	6.2E+0	*	*	7.0E+1	*	*	*	ALERT
2	5.0E+2	3.1E+0	*	*	3.5E+1	*	*	*	
3	3.3E+2	2.0E+0	*	*	2.3E+1	*	*	*	
4	2.5E+2	1.5E+0	*	*	1.75E+1	*	*	*	
1	1.0E+2	6.2E-1	*	*	7.0E+0	*	*	*	UNUSUAL EVENT
2	5.0E+1	3.1E-1	*	*	3.5E+0	*	*	*	
3	3.3E+1	2.0E-1	*	*	2.3E+0	*	*	*	
4	2.5E+1	1.5E-1	*	*	1.7E+0	*	*	*	

* Offscale Low
** Offscale High (Confirmation Only)

CHART A(2)
GASEOUS EFFLUENT ACTION LEVELS continued

2. AUX BUILDING VENT RELEASES *WITHOUT CORE DAMAGE*

NOTE: Use adverse meteorology conditions (ADV MET) only when, 10m and 60m wind speed < 5mph AND Delta-T > +2.4 degrees F or Sigma Theta is < 3.01 degrees. All other cases are average meteorology (AVG MET).

NOTE: R-13 and R-14 are expected to be off scale high during all events on this page.

SV & SFP FANS	AUX BLDG SPING MONITORS				EMERG. CLASS.
TOTAL NUMBER RUNNING	MID RANGE CPM (01-07) PPCS PT G9086G		HIGH RANGE CPM (01-09) PPCS PT G9088G		
	AVG MET	ADV MET	AVG MET	ADV MET	
1	**	9.4E+4	1.6E+4	1.0E+2	GENERAL EMERG.
2	**	4.7E+4	8.0E+3	5.0E+1	
3	**	3.1E+4	5.3E+3	3.3E+1	
4	**	2.3E+4	4.0E+3	2.5+1	
1	7.5E+5	4.6E+3	8.0E+2	5.0E+0	SITE EMERG.
2	3.7E+5	2.3E+3	4.0E+2	2.5E+0	
3	2.5E+5	1.5+3	2.6E+2	1.6E+0	
4	1.8E+5	1.1E+3	2.0E+2	1.2E+0	

SV & SFP FANS TOTAL NUMBER RUNNING	AUX BLDG SPING MONITORS		EMERG. CLASS.
	LOW RANGE Ci/cc (01-05) PPCS PT G9084G	MID RANGE CPM (01-07) PPCS PT 9086G	
1	**	8.6E+3	ALERT
2	**	4.3E+3	
3	**	2.8E+3	
4	**	2.1E+3	
1	6.3E-2	8.6E+2	UNUSUAL EVENT
2	3.1E-2	4.3E+2	
3	2.1E-2	2.8E+2	
4	1.5E-2	2.1E+2	

** Offscale High (Confirmation Only)

CHART A(2)
GASEOUS EFFLUENT ACTION LEVELS continued

3. STEAM LINE RELEASE ***WITH SIGNIFICANT CORE DAMAGE***

Instrument readings assuming radioactive steam is releasing at a total of 1.4E+5 pounds per hour to the atmosphere AND Containment High Range Radiation Monitor 42599 (R-40) or 42600 (R-41) reads 1000 R/hr for > 2 minutes within one-half hour of the accident.

	"A" Steam Line Monitors		"B" Steam Line Monitors		Emergency Classification
R-15 (cpm)	R-31 (mR/hr)	R-32 (R/hr)	R-33 (mR/hr)	R-34 (R/hr)	
**	1.3E+3	1.3E+0	1.3E+03	1.3E+0	General Emergency
**	6.0E+1	--	6.0E+1	--	Site Emergency
**	1.5E-1	--	1.5E-1	--	Alert
2.0E+05	--	--	--	--	Unusual Event

** Offscale High (Confirmation Only)

4. SHIELD BUILDING STACK RELEASE

Instrument readings assuming SBV System is operating in the recirculation mode.

Reactor Bldg. Discharge Vent SPING		Emergency Classification
PPCS PT G9077G (02-07) Mid Range (cpm)	PPCS PT G9079G (02-09) High Range (cpm)	
1.3E+05	1.5E+2	General Emergency
6.7E+03	7.0E+0	Site Emergency
1.5E+1	--	Alert
--	--	Unusual Event

CHART B
FUEL DAMAGE INDICATION

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) CET > 1,200 Degrees for greater than 15 minutes, OR (2) R40 or R41 > 1,000 R/hr, OR (3) SACRG-1, Severe Accident Control Room Guideline Initial Response has been implemented.	Plant conditions exist that make the release of large amounts of radioactivity in a short time period possible.	GENERAL EMERGENCY
(Major damage is more than one spent fuel element damaged.) (1) Fuel Handling accident in Containment a. Alarm on R-11 OR R-12, AND b. Dropped spent fuel assembly, OR c. Report of a large object dropped in Rx core, OR (2) Fuel Handling Accident in Auxiliary Bldg. a. Alarm on R-13 or R-14, AND b. A large object dropped in spent fuel pool, OR c. A dropped spent fuel assembly, OR d. A loss of water level below spent fuel.	Major damage to spent fuel in containment or auxiliary building.	SITE EMERGENCY
(1) R-9 indication is offscale high, AND (2) Laboratory analysis confirms RCS activity levels comparable to USAR Appendix D, Table D.4-1.	Severe loss of fuel cladding a. Very high coolant activity sample b. Failed fuel monitor indicates greater than 1% fuel failures within 30 minutes or 5% total fuel failures.	ALERT
(1) Fuel Handling Accident in Containment a. A confirming report, AND b. Alarm on R-11 OR R-12, OR (2) Fuel Handling Accident in Auxiliary Bldg. a. A confirming report, AND b. Alarm on R-13 OR R-14.	Fuel damage accident with release of radioactivity to containment or auxiliary building.	ALERT
(1) With RCS Temperature > 500°F, a. > 1.0 µCi/gram DOSE Equivalent I-131 for 48 hours, OR b. Exceeding 60 µCi/gram for Dose Equivalent I-131, OR c. $> 91/\bar{E}$ µCi/cc As determined by SP-37-065 (from T.S. 3.1.c)	High reactor coolant activity sample.	UNUSUAL EVENT
(1) R-9 is greater than 5.0 R/hr, AND (2) Verified by RCS chemistry sample analysis.	Failed fuel monitor indicates greater than 0.1% equivalent fuel failures within 30 minutes.	UNUSUAL EVENT

CHART C
PRIMARY LEAK TO LOCA

<u>NOTE</u>: This chart does not apply when leakage from the Reactor Coolant System is caused by a Steam Generator tube rupture.

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) <u>LOCA</u> is verified per IPEOP E-1, "Loss of Reactor or Secondary Coolant," <u>AND</u> (2) ECCS failure is indicated by: a. SI and RHR pumps not running, <u>OR</u> b. Verification of no flow to the reactor vessel, <u>OR</u> c. Core exit thermocouples indicate greater than 1,200°F, <u>AND</u> (3) Failure or potential failure of containment is indicated by: a. Physical evidence of containment structure damage, <u>OR</u> b. Containment Pressure is > 23 PSIG and loss of all containment fan coil units and both trains of ICS, <u>OR</u> c. Containment hydrogen monitor indicates ≥ 10% hydrogen concentration, <u>OR</u> d. Containment pressure exceeds 46 psig.	(1) Loss of coolant accident, <u>AND</u> (2) Initial or subsequent failure of ECCS, <u>AND</u> (3) Containment failure or potential failure exists (loss of 2 of 3 fission product barriers with a potential loss of 3rd barrier).	GENERAL EMERGENCY
(1) SI System is activated and RCS leakage exceeds charging system capacity as verified by Control Room indications or IPEOPs.	Reactor Coolant System leakage greater than make-up pump capacity.	SITE EMERGENCY
(1) Charging flow verses letdown flow indicates an unisolable RCS leak > 50 gpm.	Reactor Coolant System leak rate greater than 50 GPM.	ALERT
(1) Initiation of reactor shutdown <u>required</u> by Technical Specification, Section T.S. 3.1.d. Indicated leakage may be determined using Reactor Coolant System mass balance calculations performed by SP-36-082.	Exceeding Reactor Coolant System leak rate, Technical Specifications, requiring reactor shutdown.	UNUSUAL EVENT

CHART D
PRIMARY TO SECONDARY LEAK

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) Entry into IPEOP E-3, "Steam Generator Tube Rupture," is expected or has occurred, <u>AND</u> (2) Primary-to-secondary flow > 800 GPM OR RCS pressure decreasing uncontrollably, <u>AND</u> (3) All three transformers Main Aux., Reserve Aux., and Tertiary Aux., are de-energized.	Rapid failure of steam generator tubes with loss of off-site power.	SITE EMERGENCY
(1) Entry into IPEOP E-3, "Steam Generator Tube Rupture," is expected or has occurred, <u>AND</u> (2) All three transformers: Main Aux., Reserve Aux., and Tertiary Aux., are de-energized.	Rapid gross failure of one steam generator tube with loss of off-site power.	ALERT
(1) Entry into IPEOP E-3, "Steam Generator Tube Rupture," is expected or has occurred, <u>AND</u> (2) Primary-to-secondary leak rate greater than 800 GPM indicated by SI flow <u>OR</u> RWST level change.	Rapid failure of multiple steam generator tubes.	ALERT
(1) Primary-to-secondary leakage > 150 gallons per day for more than 4 hours (TS 3.1.d.2). (**Do not delay declaration** if leakage suddenly increases above 150 gallons per day <u>AND</u> plant shutdown actions are initiated.)	Exceeding Primary-to-Secondary leak rate Technical Specification.	UNUSUAL EVENT

CHART E
LOSS OF POWER

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) RCS is > 200°F, <u>AND</u> (2) Buses 1 through 6 are de-energized including the D/G supplies to buses 5 and 6, <u>AND</u> (3) Loss of the turbine driven AFW pump, <u>AND</u> (4) Conditions exist for greater than 2 hours.	Failure of off-site and on-site AC power, <u>AND</u> Total loss of auxiliary feedwater makeup capability for greater than 2 hours. (Loss of power plus loss of all AFW would lead to clad failure and potential containment failure.)	GENERAL EMERGENCY
(1) Buses 1 through 6 are de-energized including the D/G supplies to buses 5 and 6 for longer than 15 minutes.	Loss of off-site power, <u>AND</u> Loss of on-site AC power (for more than 15 minutes).	SITE EMERGENCY
(1) Low voltage lockout <u>OR</u> de-energized condition on all safeguards DC distribution cabinets for greater than 15 minutes. a. BRA 102 and BRB 102, <u>OR</u> b. BRA 104 and BRB 104, <u>OR</u> c. BRA 102 and BRB 104, <u>OR</u> d. BRB 102 and BRA 104	Loss of all vital on-site DC power (for more than 15 minutes).	SITE EMERGENCY
(1) Low voltage lockout <u>OR</u> de-energized condition on all safeguards DC distribution cabinets for less than 15 minutes. a. BRA 102 and BRB 102, <u>OR</u> b. BRA 104 and BRB 104, <u>OR</u> c. BRA 102 and BRB 104, <u>OR</u> d. BRB 102 and BRA 104	Loss of all vital on-site DC power (for less than 15 minutes).	ALERT
(1) Buses 1 through 6 are de-energized, <u>AND</u> (2) The D/G supplies to buses 5 and 6 do not respond as designed. AC power is restored to bus 5 or 6 within 15 minutes.	Loss of off-site power, <u>AND</u> Loss of on-site AC power (for less than 15 minutes.)	ALERT
(1) All three transformers: Main Aux., Reserve Aux., and Tertiary are de-energized, <u>OR</u> (2) Both D/Gs unavailable (unable to supply bus 5 or 6 by any means).	Loss of off-site power, <u>OR</u> Loss of on-site power capability.	UNUSUAL EVENT
(1) Core is unloaded or reactor cavity is flooded with internals removed, <u>AND</u> (2) Buses 1 through 6 are de-energized including the D/G supplies to buses 5 and 6 for longer than 15 minutes.	Loss of off-site power, <u>AND</u> Loss of on-site AC power (for more than 15 minutes).	UNUSUAL EVENT

CHART F
ENGINEERED SAFETY FEATURE ANOMALY

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) RCS > 200°F with a loss of cooling capability or inventory control: a. Sustained loss of negative reactivity control, OR b. Steam dump, S/G safeties, and power operating reliefs not operable (> 350°F), OR c. Inability to feed S/Gs (No AFW or Main Feedwater/Condensate Flow), OR d. Sustained loss of RCS inventory control, OR e. Sustained loss of both trains of RHR, AND the inability to sustain either natural OR forced circulation with the steam generators (≤ 350°F). (A Site Emergency should be declared upon the initiation of bleed and feed per FR H.1, "Response to Loss of Secondary Heat Sink.")	Complete loss of any function needed when RCS > 200°F.	SITE EMERGENCY
(Apply this criteria when the RCS is ≤ 200°F.) (1) Loss of both trains of RHR (Does not apply when core is unloaded OR cavity is flooded with internals removed.)	Complete loss of any function needed when RCS ≤ 200°F.	ALERT
(1) Failure of both Rx trip breakers to open upon receipt of a valid signal. Applies even if IPEOP FR S.1 is not entered.	Failure of the Reactor Protection System to initiate and complete a reactor trip which brings the reactor subcritical.	ALERT
(1) Loss of ESF function, required support function or required Tech Spec instruments OR Exceeding Tech Spec Safety Limits, AND (2) upon discovery, inability or failure to take required shutdown or mode change actions within the required time. (Total loss of AFW system when required (FR-H.1 implemented) should be declared a UE regardless of Tech Spec action compliance.)	Inability to reach required shutdown within Tech Spec limits	UNUSUAL EVENT

CHART G
LOSS OF INDICATION

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) Total loss of Annunciator System computer alarms, and sequence of events recorder, <u>AND</u> (2) Uncontrolled plant transient in progress or initiated during the loss.	Most or all alarms (annunciators) lost and a plant transient initiated or in progress.	SITE EMERGENCY
(1) Total loss of Annunciator System, computer alarms, and sequence of events recorder.	Most or all alarms (annunciators) lost.	ALERT
(1) Significant loss of ESF or Rx Protection instrumentation. An Unusual Event should <u>NOT</u> be declared for a non-emergency Tech Spec backdown, when the affected parameter remains monitorable.	Indications or alarms on process or effluent parameters not functional in control room to an extent requiring plant shutdown or other significant loss of assessment capability.	UNUSUAL EVENT

CHART H

(DELETED)

CHART I
SECONDARY SIDE ANOMALY

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) Main steam line break that results in a SI actuation, AND (2) a. R-15 or R-19 reads offscale high with confirmation by chemistry analysis, OR b. Primary-to-secondary leakage > 50 gpm, AND (3) a. R-9 or CNTMT high range rad monitors (42599, 42600) indicate > 10 R/hr, OR b. CNTMT hydrogen monitor indicates > 1% hydrogen concentration.	Steam line break, AND Primary-to-secondary leak > 50 GPM, AND Indication of Fuel Damage.	SITE EMERGENCY
(1) Main steam line break that results in a SI actuation, AND a. R-15 OR R-19 reads a factor of 1000 above normal, OR b. Primary-to-secondary leakage > 10 gpm.	Steam line break with significant (greater than 10 GPM) primary-to-secondary leakage. (Applies even if events occur in opposite steam generators.)	ALERT
(1) Turbine trip and observation of penetration of casing.	Turbine rotating component failure causing rapid plant shutdown.	UNUSUAL EVENT
(1) The uncontrolled depressurization of the secondary system that results in an SI actuation. (Unusual Event should be declared even if transient was mitigated by closure of the Main Steam Isolation Valves.	Rapid depressurization of the secondary side.	UNUSUAL EVENT

CHART J
MISCELLANEOUS ABNORMAL PLANT CONDITIONS

Note
Chart J is now a two page chart.

<table>
<tr><th>KNPP INDICATION</th><th>EMERGENCY CLASSIFICATION CRITERIA</th><th>CLASSIFICATION</th></tr>
<tr><td>(1) Containment boundary failure or potential failure:
a. Containment pressure > 46 psig, <u>OR</u>
b. Loss of all containment fan coil units and both trains of ICS, <u>OR</u>
c. Containment hydrogen monitor ≥ 10% hydrogen concentration,
<u>AND</u>
(2) Loss of core cooling capability:
a. Loss of SI and RHR flow,
<u>AND</u>
(3) Failure of shutdown system when required:
a. Entry into IPEOP FR-S.1, “Response to Nuclear Power Generation/ATWS,” <u>OR</u>
b. Loss of AFW for greater than 30 minutes with loss of main FW and condensate.</td><td>Other plant conditions that make a release of large amounts of radioactivity in a short time period possible; e.g., any core melt situation.

Examples:
- Failure of main FW and AFW systems for greater than 30 minutes without Safety Injection and Residual Heat Removal flow. Plus a containment failure is imminent.
- Transient requiring the operation of shutdown systems with a failure of these shutdown systems. In addition, failure of SI and RHR and containment failure is imminent.</td><td>GENERAL EMERGENCY</td></tr>
</table>

Note
Chart J is continued on the next page.

CHART J
MISCELLANEOUS ABNORMAL PLANT CONDITIONS

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
Two fission product barriers are lost with the potential or probability of losing the third barrier. Indications: (1) Containment boundary potential failure: a. Containment pressure > 46 psig, <u>OR</u> b. Loss of all containment fan coil units and both trains of ICS, <u>OR</u> c. Containment hydrogen monitor ≥ 10% hydrogen concentration, <u>AND</u> (2) RCS Boundary: a. Loss of RHR and SI flow, <u>OR</u> b. LOCA is verified per IPEOP E-1, <u>AND</u> (3) Fuel Cladding: a. R-9 > 10 R/hr, <u>OR</u> b. RCS Chemistry Analysis	Other plant conditions that make a release of large amounts of radioactivity in a short time period possible; e.g., any core melt situation. Examples: - Failure of main FW and AFW systems for greater than 30 minutes without Safety Injection and Residual Heat Removal flow. Plus a containment failure is imminent. - Transient requiring the operation of shutdown systems with a failure of these shutdown systems. In addition, failure of SI and RHR and containment failure is imminent.	GENERAL EMERGENCY
(1) Evacuation of Control Room (E-O-06 event).	Evacuation of control room and control of shutdown systems required from local stations.	SITE EMERGENCY
(1) Conditions that warrant increased awareness on part of the plant staff will be evaluated by the Plant Manager or his designate. This is to determine if conditions are applicable for activating the E.P. <u>Example</u>: Loss of AFW system when required, validated upon implementation of FR H.1 "Response to Loss of Secondary Heat Sink."	Other plant conditions that warrant increased awareness on the part of plant staff or state and/or local authorities.	UNUSUAL EVENT

CHART K
FIRE AND FIRE PROTECTION

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) A fire within the Auxiliary Building, Technical Support Center, safeguards alley, D/G rooms, Battery Rooms, or screenhouse that defeats redundant safety trains of ESF equipment causing the required ESF system to be inoperable.	A fire compromising the functions of safety systems.	SITE EMERGENCY
(1) A fire within the Auxiliary Building, Technical Support Center, safeguards alley, D/G rooms, Battery Rooms, or screenhouse that lasts more than 10 minutes OR causes a single train of required ESF equipment to be inoperable.	A fire potentially affecting safety systems.	ALERT
(1) Any fire within the protected area lasting more than 10 minutes.	A fire within the plant lasting more than 10 minutes.	UNUSUAL EVENT

CHART L

(DELETED)

CHART M
EARTHQUAKE

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) Activation of seismic recorder with TRIGGER, OBE, and DBE lights lit in relay room on RR159, AND (2) Verification of a seismic event by physical experience or from U. of W. - Milwaukee Seismic Center.	An earthquake greater than Design Basis Earthquake (DBE).	SITE EMERGENCY
(1) Activation of seismic recorder with TRIGGER, and OBE lights lit in relay room on RR159, AND (2) Verification of a seismic event by physical experience or from U. of W. - Milwaukee Seismic Center.	An earthquake greater than Operational Basis Earthquake (OBE).	ALERT
(1) Activation of seismic recorder with TRIGGER light lit in relay room on RR159, OR (2) An earthquake felt in the Plant*. (*Should be confirmed by evidence of physical damage or verification from University of Wisconsin Seismic Center.)	An earthquake felt in plant or detected on station seismic instrumentation.	UNUSUAL EVENT

NOTE: 1.) Telephone numbers for U of W - Milwaukee Seismic Center are in the KPB Emergency Telephone Directory, ETD 02.

2.) The Point Beach Seismic Monitor may be used if the KNPP Monitor is out of service.

CHART N
HIGH WINDS OR TORNADO

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) Winds in excess of 100 mph for greater than 1 hour, AND (2) Plant above cold shutdown condition.	Sustained winds in excess of design levels with plant not in cold shutdown.	SITE EMERGENCY
(1) A tornado which strikes the facility, AND (2) Causes damage that affects the continued safe operation of the plant.	Any tornado striking facility.	ALERT
(1) A tornado within sight of the plant which has caused the loss of at least one of the off-site transmission lines, OR (2) A tornado observed on-site.	Any tornado on-site.	UNUSUAL EVENT

CHART O
FLOOD, LOW WATER, OR SEICHE

<table>
<tr><th colspan="4">KNPP INDICATION</th><th>EMERGENCY CLASSIFICATION CRITERIA</th><th>CLASSIFICATION</th></tr>
<tr><td colspan="3">FOREBAY LEVEL
Indicated for > 15 minutes</td><td></td><td rowspan="5">Flood, low water, or seiche near design levels.</td><td rowspan="5">ALERT</td></tr>
<tr><td>0 PUMPS</td><td>1 PUMP</td><td>2 PUMPS</td><td>CORRESPOND TO LAKE LEVEL</td></tr>
<tr><td>NOTE 3</td><td>NOTE 1</td><td>≥ 94% *</td><td>≥ 588 ft.</td></tr>
<tr><td>< 50% *</td><td>NOTE 5</td><td>NOTE 5</td><td>< 568.5 ft.</td></tr>
<tr><td colspan="4"><u>OR</u> Deep water Wave ≥ 22.5 ft.</td></tr>
<tr><td colspan="3">FOREBAY LEVEL
Indicated for > 15 minutes</td><td></td><td rowspan="5">50-year flood, low water level or seiche</td><td rowspan="5">UNUSUAL EVENT</td></tr>
<tr><td>0 PUMPS</td><td>1 PUMP</td><td>2 PUMPS</td><td>CORRESPOND TO LAKE LEVEL</td></tr>
<tr><td>NOTE 2</td><td>≥ 98% *</td><td>≥ 88% *</td><td>≥ 586 ft.</td></tr>
<tr><td>< 53.1% *</td><td>< 46.9% *
NOTE 4</td><td>NOTE 5</td><td>< 569.5 ft.</td></tr>
<tr><td colspan="4"><u>OR</u> Deep water wave ≥ 18 ft. (as confirmed by the U.S. Coast Guard, Two Rivers)</td></tr>
</table>

NOTE 1: Above the bottom of bar No. 1 painted on the south wall of the forebay.

NOTE 2: Above the bottom of bar No. 2 painted on the south wall of the forebay.

NOTE 3: Above the bottom of bar No. 3 painted on the south wall of the forebay.

NOTE 4: Applies to an uncontrollable decrease (cannot be restored by operator action. If the water box inlet valves are throttled, use other means to determine lake level per E-CW-04, "Loss of Circulating Water.")

NOTE 5: The corresponding forebay level for the associated lake level is below the circulating water pump trip setpoint of 42%. Therefore, this criterion will not be reached.

* Computer point for forebay level is L09075A and should be used because of its greater accuracy. Plant elevations and lake elevations are referenced to International Great Lakes Datum (IGLD), 1955

(IGLD 1955 = IGLD 1985 - 0.7 FEET)

CHART P
EXTERNAL EVENTS AND CHEMICAL SPILLS

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) An aircraft crash into plant buildings which causes a complete loss of an ESF function.	Aircraft crash affecting vital structures by impact OR fire.	SITE EMERGENCY
(1) A missile strikes plant buildings, OR (2) An explosion occurs within a plant building, which causes a complete loss of an ESF function.	Severe damage to safe shutdown equipment from missiles or explosion.	SITE EMERGENCY
(1) Release of flammable or toxic gas from a ruptured container which enters a vital area. Portable H_2 monitor detects explosive concentration of H_2 in vital area.	Uncontrolled release of toxic or flammable gas is confirmed within vital area.	SITE EMERGENCY
(1) An aircraft crashes into plant buildings OR switchyard which affects plant operation.	Aircraft crash on facility.	ALERT
(1) A missile strikes the facility which affects plant operation.	Missile impact from whatever source on facility.	ALERT
(1) Release of toxic or flammable gas from a ruptured container such that the gases enter the plant protected area or buildings.	Uncontrolled release of toxic or flammable gas is confirmed within the protected area.	ALERT
(1) Self-explanatory.	Known explosion damage to facility affecting plant operation.	ALERT
(1) An aircraft crash within the site boundary, OR (2) Unusual aircraft activity such as erratic flying, dropped unidentified object, or other hostile acts, which threaten the plant or plant personnel. (Any other persistent aircraft activity for which identification attempts through the FAA or other agencies have been unsuccessful.)	Aircraft crash on-site or unusual aircraft activity over facility.	UNUSUAL EVENT
(1) Release of toxic or flammable gas on site, AND (2) Portable monitors indicate toxic or explosive concentrations at life threatening levels of the gas near the spill area.	Uncontrolled release of toxic or flammable gas is confirmed on site.	UNUSUAL EVENT

CHART Q
SECURITY CONTINGENCY

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) Physical attack on the plant that has resulted in unauthorized personnel occupying the control room or any other vital areas as described in the Security Plan.	Loss of physical control of the plant.	GENERAL EMERGENCY
(1) Physical attack on the plant involving imminent occupancy of the control room, auxiliary shutdown panels, or other vital areas as defined by the Security Plan.	Imminent loss of physical control of the plant.	SITE EMERGENCY
(1) Security safeguards contingency event that results in a hostile force entering the protected area of the plant, but not gaining control over shutdown capability or of any vital areas as defined in the Security Plan, <u>OR</u> (2) Security safeguards contingency event that results in a site specific HI level CREDIBLE threat as defined in the Security Plan.	Ongoing security compromise.	ALERT
(1) Security safeguards contingency event that results in a site specific LO level CREDIBLE threat as defined in the Security Plan, <u>OR</u> (2) Security safeguards contingency event that results in a Bomb threat accompanied by interception of bomb materials, <u>OR</u> (3) Security safeguards contingency event that results in an attempted entry into the protected area of the plant by a hostile force, <u>OR</u> (4) Security safeguards contingency event that results in undetonated bomb found within the protected area.	Security threat or attempted entry or attempted sabotage.	UNUSUAL EVENT

<u>NOTE</u>: Security staff will <u>NOT</u> act as notifier during **security** events. Utilize Control Room staff for notifications.

Nuclear Safety Related	☐ Yes ☑ No	PORC Review Required	☐ Yes ☑ No	SRO Approval Of Temporary Changes Required	☐ Yes ☑ No

1.0 Purpose

1.1 This procedure provides instruction for maintaining exposure to emergency workers As Low As Reasonably Achievable (ALARA).

2.0 General Notes

2.1 Definitions

2.1.1 FEMA - Federal Emergency Management Agency

2.1.2 HP - Health Physics

2.1.3 PA - Protected Area

2.1.4 RCA - Radiologically Controlled Area

3.0 Precautions and Limitations

3.1 None

4.0 Initial Conditions

4.1 This procedure shall be implemented upon declaration of an **Alert, Site Emergency, General Emergency**, or when directed by the Shift Manager or Emergency Director.

5.0 Procedure

5.1 **All emergency personnel** are responsible for adhering to the requirements of this procedure.

5.1.1 The requirements of the Health Physics Procedures shall be applicable during all radiological emergencies, except as authorized by the Radiological Protection Director (RPD) or Emergency Director (ED).

5.1.2 For RCA entries, if any 10CFR20 dose limit is likely to be exceeded, an "Emergency Radiation Work Permit" (ERWP), Form EPIPF-AD-11-01, shall be completed. Otherwise, use an existing Radiation Work Permit (RWP) or fill out a RWP in accordance with NAD-08.03, "Radiation Work Permit."

5.1.3 A PRIORITY ENTRY can be used for quick action to expedite the entry of emergency response personnel into the RCA in accordance with EPIP-RET-02D, "Emergency Radiation Entry Controls and Implementation" and should NOT exceed 10CFR20 dose limits.

Note

Dose extensions beyond 10CFR20 limits (see Table before Step 5.2) may be allowed within the dose guidelines specified by EPA-400 (see Table after Step 5.1.4).

5.1.4 For any entry where an exposure greater than 10CFR20 dose limits is likely, an Authorization For Increased Radiation Exposure (Form HPF-120) shall be completed in accordance with NAD-01.11, "Dosimetry and Personnel Monitoring." All exposures which could exceed 10CFR20 dose limits shall be approved by the ED.

10CFR20 RADIATION DOSE LIMITS

TEDE, ADULT	ANNUAL	5 REM
TODE, ADULT	ANNUAL	50 REM
LDE, ADULT	ANNUAL	15 REM
SDE, SKIN, ADULT	ANNUAL	50 REM
SDE, EXTREMITY, ADULT	ANNUAL	50 REM
DAC-HOUR, ADULT	ANNUAL	2,000 DAC-HOURS

EPA RADIATION DOSE GUIDELINES (EPA-400)

Projected Dose (rem) to Emergency Team Workers	Action/Condition	Comments
TEDE < 5 rem All Other Organs < 50 rem TODE	Control exposure of emergency workers to these levels except for those instances listed below. (Appropriate controls for emergency workers include time limitations, respirators, and stable iodine.)	"All Other Organs" include: Skin Extremities and Thyroid. Stable iodine may be made available for use where predicted doses exceed 25 rem to the thyroid. Although respirators and stable iodine should be used where effective to control dose to emergency team workers, thyroid dose may not be a limiting factor for lifesaving missions. For Environmental/Monitoring Teams, refer to RASCAL "Maximum Doses at Selected Distances" output screen. Check bone, lung, and thyroid doses.
TEDE < 10 rem All Other Organs < 100 rem TODE	Emergency workers' exposure should be controlled below these levels when their mission involves protecting valuable property.	
TEDE < 25 rem All Other Organs < 250 rem TODE	Emergency workers' exposure should be controlled below these levels when their mission involves life saving or protection of large populations.	
TEDE > 25 rem All Other Organs > 250 rem TODE	Exposures above these levels to emergency workers will be on a voluntary basis only to persons fully aware of the risks involved.	

5.1.5 Understand the risks and benefits of taking Potassium Iodide (KI) orally when radioactive iodine levels warrant taking KI.

5.2 The **Emergency Director** (ED) is responsible for approving all requests for exposure in excess of 10CFR20 dose limits and the distribution of Potassium Iodide.

5.3 The **Radiological Protection Director** (RPD) has the overall responsibility for in-plant personnel monitoring and shall:

5.3.1 Ensure that personnel within the protected area are issued the appropriate dosimetry.

5.3.2 Evaluate any potential exposure to radiation in excess of 10CFR20 dose limits for approval by the ED (Form HPF-110) in accordance with HP-05.001, "Survey and Sampling Techniques." In the absence of the RPD, the ED may authorize an overexposure after concurrence of the on-shift HP or an In-plant Radiation Emergency Team (IRET) member.

5.3.3 In accordance with EPIP-AD-18, "Potassium Iodide Distribution," advise the ED on the need to make thyroid blocking agent (Potassium Iodide, a stable iodine) available to Emergency Response Organization (ERO) members who may be subject to radioiodine intake.

5.3.4 Review all RWPs and ERWPs in use.

5.3.5 Establish control over radiation, high radiation or contamination areas discovered outside the normal RCA when levels are found to exceed the following:

- 2 mr/hr — Direct Radiation
- 2,000 DPM/100 cm^2 — Beta-Gamma
- 200 DPM/100 cm^2 — Alpha

5.3.5.1 This control may include:

a. Roping off and posting additional areas within the plant

b. Roping off and posting all doors to an entire building

c. Designating and posting the entire PA as RCA

d. Establishing roadblocks in conjunction with the ED and Site Protection Director (SPD), and designating the entire area within the roadblocks as RCA

Note

Emergency response personnel reporting to the plant shall NOT be required to initial their RWP prior to RCA entry when the RCA is expanded. The RWP shall be initialed prior to making entries at the Radiation Protection Office (RPO) or Radiological Access Facility (RAF). See Step 5.1, for Emergency RWP (ERWP) requirements.

5.4 The **In-Plant Radiation Emergency Team** (IRET) is responsible for performing those activities necessary to implement the purpose of this procedure.

5.4.1 Make radiological assessments of all in-plant areas requiring access during an emergency.

5.4.2 Report in-plant radiological conditions to the RPD.

5.4.3 Determine the projected amount of time in-plant emergency workers will be allowed to remain in any radiation/contaminated area through pre-entry review of:

a. Projected route exposures

b. Measured dose rates and airborne concentrations

c. Personnel exposure history

d. Projected duration of task

e. Information on current plant conditions and the plant area under consideration

5.4.4 Provide radiation monitoring coverage for all continuously occupied areas.

5.4.5 Perform air sample surveys and direct radiation surveys as directed.

5.4.6 Control exposure to airborne radionuclides in accordance with the following:

a. Use only Self-Contained Breathing Apparatus (SCBA) pressure demand respirators when entering areas of unknown airborne concentrations.

b. Limit airborne particulate exposures to < 200 DAC-HOURS to the maximum extent possible.

c. Remove any worker from further emergency duties upon exceeding 2,000 DAC-HOURS.

d. Assess internal dose by performing whole body counts in accordance with procedure HP-03.008, "Evaluation of Inhalations or Ingestions."

Note

For entry teams originating from the Operational Support Facility (OSF), document the items below (Steps 5.4.7.a. through Step 5.4.7.h.) on Form EPIPF-OSF-03-01.

5.4.7 Review all planned entries with the entry team members and discuss the following:

a. Potential stress conditions and problems

b. Work methods, work location, and description of task

c. Number of personnel required and access routes inside the RCA

d. Allowable exposure limits, expected doses, stay times

e. Tools, equipment, and parts

f. Lighting

g. Communications requirements

h. Abort instructions

5.4.8 As directed, accompany any worker entering an area where radiological conditions are unknown.

5.4.9 Remove any individual who has exceeded 10CFR20 dose limits from work involving additional radiation exposure. The worker's exposure record shall be reviewed by the RPD and ED prior to further radiation work. Any further radiation work must be authorized by the ED.

5.4.10 As directed, store samples collected post accident in designated storage locations. Liquid, air particulate, and halogen filter samples should be stored in the radioactive waste storage areas of the Auxiliary Building.

6.0 Final Conditions

6.1 Plant Emergency has been Terminated or Recovery actions have begun and the Emergency Response Manager has suspended the use of EPIPs.

7.0 References

7.1 Kewaunee Nuclear Power Plant Emergency Plan

7.2 NUREG-0654/FEMA-REP-1, REV. 1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants (Nov. 1980)

7.3 EPA-400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (Oct. 1991)

7.4 10CFR20, Standards for Protection Against Radiation

7.5 Kewaunee Nuclear Power Plant Health Physics Procedure Manual

7.6 KNPP Commitment Tracking System number 97-125, NRC Inspection Report 97-13, Repair Personnel

7.7 Implementing Procedures

7.7.1 EPIP-RET-02D, Emergency Radiation Entry Controls and Implementation

7.7.2 NAD-08.03, Radiation Work Permit

7.7.3 EPIP-AD-18, Potassium Iodide Distribution

7.7.4 EPIP Appendix B, Forms

7.7.5 HP-03.008, Evaluation of Inhalations or Ingestions

7.7.6 NAD-01.11, Dosimetry and Personnel Monitoring

7.7.7 HP-05.001, Survey and Sampling Techniques

8.0 Records

8.1 The following QA records and non-QA records are identified in this directive/procedure and are listed on the KNPP Records Retention Schedule. These records shall be maintained according to the KNPP Records Management Program.

8.1.1 QA Records

- Emergency Radiation Work Permit, Form EPIPF-AD-11-01

8.1.2 Non-QA Records

None

1.0 Purpose

1.1 This procedure provides instructions for determining Protective Action Recommendations (PARs).

2.0 General Notes

2.1 The Shift Manager is the initial ED in all situations. Any transfer of this responsibility should be documented in the Shift Manager's log and communicated to all other directors.

2.2 Upon declaration of a plant emergency, the Emergency Director (ED) is initially responsible to provide off-site authorities with Protective Action Recommendations (PARs). When the Emergency Operations Facility (EOF) has been activated, this responsibility will be assumed by the Emergency Response Manager (ERM). This responsibility shall NOT be delegated.

2.3 To be most effective protective actions must be taken before or shortly after the start of a major release to the atmosphere. PARs must be determined and communicated as quickly as possible.

2.4 Communication of PARs should be in progress to state and local emergency government authorities within 15 minutes of the emergency being declared or as soon as possible without further compromise to plant or public safety.

2.5 As more information becomes available, the most current PAR should be reviewed and revised, as necessary, in accordance with Section 5.0 of this procedure.

3.0 Precautions and Limitations

3.1 PARs are normally implemented for affected populations within the 10-mile plume exposure pathway EPZ. However, do NOT ignore populations outside the 10-mile plume exposure pathway if projected doses or field readings indicate doses > 1 rem TEDE or > 5 rem thyroid.

3.2 IF there is a PAR change due to a wind shift, THEN the new PAR should include the following as downwind sectors:

3.2.1 All downwind sectors from all previous PARs for this emergency, AND

3.2.2 All downwind sectors from the new PAR, AND

3.2.3 All downwind sectors through which the wind shift occurred.

3.3 PARs already implemented should never be withdrawn or reduced due to a wind shift.

3.4 Withdrawal or reduction of protective actions from areas where they have already been implemented is NOT advisable because of the potential for changing conditions and confusion.

3.5 Under normal conditions, evacuation of members of the general population should be initiated for most incidents at a projected dose of 1 rem. Sheltering may be preferable to evacuation as a protective action in some situations. Examples of situations or groups for which evacuation may NOT be appropriate at 1 rem include:

a. The presence of severe weather,

b. Competing disasters,

c. Institutionalized persons who are not readily mobile, AND

d. Local physical factors that impede evacuation.

4.0 Initial Conditions

4.1 The Emergency Director shall classify the emergency in accordance with EPIP-AD-02, "Emergency Class Determination," prior to the implementation of this procedure.

5.0 Procedure

Note

Adverse meteorology exists if:

1. The 10 AND 60 meter wind speed is less than 5 mph, AND

2. Delta T is greater than +2.4°F OR Sigma Theta is less than 3.01 degrees. (Refer to Graphic Display #52 from the PPCS terminal.)

5.1 Determine a default PAR for the declared emergency classification.

5.1.1 General Emergency

a. IF adverse meteorology exists, THEN recommend to off-site authorities using Form EPIPF-AD-07-01, "Event Notice," Box 10 to:

- [B] Evacuate ALL sectors (360°) out to 5 miles.

b. IF adverse meteorology does NOT exist, THEN recommend to off-site authorities using Form EPIPF-AD-07-01, "Event Notice," Box 10 to:

- [B] Evacuate ALL sectors (360°) out to 2 miles, AND

Note
To determine sectors in [B] include the downwind sector(s) from Form EPIPF-AD-07-01 Table 1.

- [B] Evacuate downwind sectors ________________ out to 5 miles.

c. Verify that security has implemented EPIP-SEC-02, "Security Force Response to Emergencies," to:

- Direct members of the general public (visitors, fisherman, tourists, etc.) to leave the site.
- Establish control measures for site access/egress.

5.1.2 Site Emergency

a. Immediate Planned Protective Action Recommendations from Form EPIPF-AD-07-01, "Event Notice," Box 10 for the general public are:

- [A] None

b. Verify that security has implemented EPIP-SEC-02, "Security Force Response to Emergencies," to:

- Direct members of the general public (visitors, fisherman, tourists, etc.) to leave the site.
- Establish control measures for site access/egress.

5.1.3 Alert

a. Immediate Planned Protective Action Recommendations from Form EPIPF-AD-07-01, "Event Notice," Box 10 for the general public are:

- [A] None

b. Verify that security has implemented EPIP-SEC-02, "Security Force Response to Emergencies," to:

- Direct members of the general public (visitors, fisherman, tourists, etc.) to leave the site.
- Establish control measures for site access/egress.

5.1.4 Unusual Event

a. Immediate Planned Protective Action Recommendations from Form EPIPF-AD-07-01, "Event Notice," Box 10 for the general public are:

- [A] None

5.2 Determine a PAR using dose projections results from EPIP-ENV-03C.

Note
A PAR from a dose projection may identify areas of concern beyond the default PAR. Any current default PAR should remain in place and be augmented with the information obtained from the dose projection, as necessary.

Note
Do NOT withdraw or reduce the default PAR based on dose projection results.

5.2.1 Obtain the most recent RASCAL dose projections for TEDE and thyroid doses from the Environmental Protection Director (EPD) or the Radiological Protection Director (RPD).

5.2.1.1 Compare the RASCAL dose projection results to EPIP-AD-19 Table 1.

5.2.1.1.1 IF dose projection results meet the criteria in Column 1, THEN determine recommended protective actions (Column 2) considering the following:

- Plant conditions (past, present, projected)
- Radiological conditions
- Impact time
- Weather (current and forecasted)
- Evacuation time estimates using EPIP-AD-19 Table 2, "Evacuation Time Estimates (KNPP/EPZ)."

5.2.2 Upon determining recommended protective actions, immediately notify the ERM or ED.

5.2.3 IF projected doses meet the criteria in EPIP-AD-02 Chart A(1), THEN relay the identified classification criteria to the ERM or ED immediately.

5.3 Determine a PAR from field radiation dose rate survey results.

5.3.1 Obtain the most recent field radiation dose rate survey results from the Environmental Monitoring Team Coordinator (ENVCd) or RPD.

5.3.2 Determine the receptor exposure dose.

5.3.2.1 IF exposure duration is unknown, THEN multiply the field radiation dose rate for a given point by a default six (6) hours.

Field radiation dose rate x 6 hours = ____________________ receptor exposure dose

5.3.2.2 IF exposure duration is known, THEN multiply the field radiation dose rate for a given point times the known exposure duration.

Field radiation dose rate × Known exposure duration = ___________ receptor exposure dose

5.3.3 Compare the receptor exposure dose, from Step 5.3.2, to EPIP-AD-19, Table 1.

5.3.3.1 IF the receptor exposure dose meets the criteria in Column 1, THEN determine recommended protective actions (Column 2) considering the following:

- Plant conditions (past, present, projected)
- Radiological conditions
- Impact time
- Weather (current and forecasted)
- Evacuation time estimates using EPIP-AD-19 Table 2, "Evacuation Time Estimates (KNPP/EPZ)."

5.3.4 Upon determining recommended protective actions, immediately notify the ERM or ED.

5.3.5 IF field radiation dose rate survey results meet the criteria in EPIP-AD-02 Chart A(1), THEN relay the identified classification criteria to the ERM or ED immediately.

5.4 Determine a PAR from air sample or ground deposition sample results.

5.4.1 Obtain the most recent air sample or ground deposition sample results from the Environmental Monitoring Team Coordinator (ENVCd) or RPD.

5.4.2 Perform a dose projection as per EPIP-ENV-03C using the air sample or ground deposition sample results as inputs to the dose projection.

5.4.3 Use dose projection results to perform Step 5.2 of this procedure.

5.5 Complete Form EPIPF-AD-07-01 "Event Notice," to inform off-site authorities of any newly developed or revised/upgraded PAR.

Note

The following step will normally be performed only if the State Emergency Operations Center (EOC) in Madison has been activated.

5.5.1 If appropriate, instruct the State Radiological Coordinator Liaison (SRCL) to discuss the changes or potential changes in PARs with the State Radiological Coordinator (SRC) for the State of Wisconsin.

5.6 Submit the Form EPIPF-AD-07-01 "Event Notice," to the appropriate Communicator for transmission to off-site authorities.

5.7 IF dose projections indicate a potential dose to the thyroid of > 25 rem, THEN verify "Potassium Iodide Distribution," EPID-AD-18, is being implemented.

5.8 IF dose projections or field readings indicate doses > 1 rem TEDE or > 5 rem thyroid to any population outside of the 10-mile plume exposure pathway EPZ, THEN report this immediately to the State and counties and if requested, provide assistance with *ad hoc* planning.

5.9 Repeat Steps 5.1, 5.2, 5.3, and 5.4 until the Final Conditions are met, see Section 6.0.

6.0 Final Conditions

6.1 Additional Protective Action Recommendations are no longer required when the plant emergency has been Terminated or Recovery Actions have begun and the responsible Director has suspended the use of EPIPs.

7.0 References

7.1 EPA-400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (May 1992)

7.2 NUREG/CR-2925, In-Plant Considerations for Optimal Off-site Response to Reactor Accidents (November 1982)

7.3 NUREG/CR-5247, Vol. 2, Rev. 2, RASCAL Version 2.1 Workbook (December 1994)

7.4 NUREG-0654, II.J.7 and II.J.8

7.5 U.S. Food and Drug Administration, 21CFR Part 1090

7.6 US-NRC RIS 2003-12, Clarification Of NRC Guidance For Modifying Protective Actions

7.7 EPIP-AD-02, Emergency Class Determination

7.8 EPIP-AD-18, Potassium Iodide Distribution

7.9 EPIP-SEC-02, Security Force Response to Emergencies

7.10 Form EPIPF-AD-07-01, Event Notice

8.0 Records

8.1 The following QA records and non-QA records are identified in this directive/procedure and are listed on the KNPP Records Retention Schedule. These records shall be maintained according to the KNPP Records Management Program.

8.1.1 QA Records

None

8.1.2 Non-QA Records

None

Column 1 PROJECTED DOSE (REM) TO THE POPULATION	Column 2 RECOMMENDED ACTIONS[a]	Column 3 COMMENTS
TEDE < 1 rem Thyroid < 5 Rem	No planned protective actions.[b] Monitor environmental radiation levels.	If the conditions of Section 6.0 are satisfied, previously recommended protective actions may be reconsidered.
TEDE > 1 rem Thyroid > 5 rem	Recommend evacuation in affected sectors. Monitor environmental radiation levels and adjust sectors recommended for evacuation based on these levels. Control access.	Seeking shelter would be an alternative if evacuation were not immediately possible. [c]

Column 1 PROJECTED DOSE (REM) TO EMERGENCY TEAM WORKERS	Column 2 RECOMMENDED ACTIONS[d]	Column 3 COMMENTS
TEDE < 5 rem All other Organs < 50 rem TODE	Control exposure of emergency workers to these levels except for those instances listed below. (Appropriate controls for emergency workers include time limitations, respirators, and stable iodine.)	"All other Organs" include: skin extremities and thyroid. Stable iodine may be made available for use where predicted doses exceed 25 rem to the thyroid. Although respirators and stable iodine should be used where effective to control dose to emergency team workers, thyroid dose may not be a limiting factor for lifesaving missions. For Environmental/Monitoring Teams, refer to RASCAL "Maximum Doses at Selected Distances" output screen. Check bone, lung, and thyroid doses.
TEDE < 10 rem All other Organs < 100 rem TODE	Emergency workers exposure should be controlled below these levels when their mission involves protecting valuable property.	
TEDE < 25 rem All other Organs < 250 rem TODE	Emergency workers exposure should be controlled below these levels when their mission involves lifesaving or protection of large populations.	

(a) These actions are recommended for planning purposes. Protective action decisions at the time of the incident must take existing conditions into consideration. These conditions include containment activity, probability of containment failure, plume transport time, release duration, and any other pertinent conditions.

(b) At the time of the incident, officials may implement low-impact protective actions in keeping with the principle of maintaining radiation exposures as low as reasonably achievable.

(c) Sheltering may be the preferred protective action when it will provide protection equal to or greater than evacuation, based on consideration of factors such as source term characteristics, and temporal or other site-specific conditions.

(d) These actions are recommended for planning purposes. Protective action decisions at the time of the incident must take existing conditions into consideration. These conditions include containment activity, probability of containment failure, plume transport time, release duration, and any other pertinent conditions.

EVACUATION TIME ESTIMATES (KNPP/EPZ)

1	2	3	4	5	6	7		8	9
Area (Pop.)	**Sectors**	**Total Vehicles**	**Route (Miles)**	**Avg. Speed (mph)**	**Travel Time (min.)**	**Notif. Time (min.)**	**Plan Time (min.)**	**Total Evac. Time (min.)**	**Severe Weather (min.)**
2 (203)	All	64	Hwy. 42 (4.75)	50	6	15	30	51	72
5N (700)	m,n,p,q, r,a,b	89	Co. Tk. B (3)	50	4	15	30	49	68
"	"	130	Hwy. 42 (7.75)	45	11	15	30	56	82
5S (585)	j,k,l,m	47	Co. Tk. AB (3)	50	4	15	30	49	68
"	"	136	Hwy. 42 (4)	50	5	15	30	50	70
10N (3559)	r,a,b	222	Co. Tk. B (4.5)	45	6	15	30	51	72
"	"	890	Hwy. 42 (7)	50	9	15	30	54	78
10NW (530)	p,q,r	89	Co. Tk. B (4.5)	45	6	15	30	51	72
"	"	77	Co. Tk. AB (2.5)	50	3	15	30	48	66
10W (803)	n,p,q	84	Co. Tk. Q (4)	45	6	15	30	51	72
"	"	167	Co. Tk. AB (8.5)	50	11	15	30	56	82
10SW (1456)	l,m,n	455	Co. Tk. Q (4)	45	6	15	30	51	72
10SSW (2029)	k,l	92	Co. Tk. Q (4)	45	6	15	30	51	72
"	"	542	Co. Tk. AB (5.25)	50	7	15	30	52	74
10S (663)	j,k	207	Hwy. 42 (5.25)	50	7	15	30	52	74

December 2, 1993

<table>
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Kewaunee Nuclear Power Plant

Emergency Plan Implementing Procedure</td><td colspan="3">No. EPIP-RET-02B
Title Gaseous Effluent Release Path, Radioactivity, and Release Rate Determination</td><td>Rev. T</td></tr>
<tr><td colspan="2"></td><td colspan="3">Date SEP 19 2003</td><td>Page 1 of 16</td></tr>
<tr><td colspan="2">Reviewed By Dan Bouche</td><td colspan="4">Approved By Jerrie Morlino</td></tr>
<tr><td>Nuclear Safety Related</td><td>☐ Yes
☑ No</td><td>PORC Review Required</td><td>☐ Yes
☑ No</td><td>SRO Approval Of Temporary Changes Required</td><td>☐ Yes
☑ No</td></tr>
</table>

1.0 Purpose

1.1 This procedure provides instruction for identifying the release path of radioactive gaseous effluents, to ensure that appropriate effluent samples are taken and appropriate monitor readouts are recorded, in order to quantify the release and provide input data for plume projection calculations.

2.0 General Notes

2.1 This procedure is used whenever radiological data is required to perform computerized dose projections per EPIP-ENV-03C, "Dose Projection Using Rascal Software."

2.2 The reference by HP effluent procedures to a four digit numeric value representing the SPING address and channel, directly relates to the address/channel numbers on the SPING CT unit. The first two digits refers to the address and the last two digits refer to the channel.

2.3 DNA (Distributed Network Architecture) is the name of the software used by RADSERV.

3.0 Precautions and Limitations

3.1 An Emergency Radiation Work Permit (ERWP) may be required as per EPIP-AD-11, "Emergency Radiation Controls" Step 4.1, for sampling within the Zone SV due to high airborne concentrations or other radiological concerns.

4.0 Initial Conditions

4.1 This procedure shall be implemented upon declaration of an **Alert, Site Emergency, General Emergency**, or when directed by the Radiological Protection Director.

5.0 Procedure

Note
Table 5 is a flow chart of the steps in this procedure.

5.1 IDENTIFY THE RELEASE PATH by monitoring the following indications:

5.1.1 Auxiliary Building Stack Release

a. R-13 cpm
b. R-14 cpm
c. R-22 cpm
d. R-35 mR/hr
e. R-36 R/hr
f. SPING 01
   1. 01-05 µCi/cc
   2. 01-07 cpm
   3. 01-09 cpm

5.1.2 Containment Stack Release

a. R-11 cpm
b. R-12 cpm
c. R-21 cpm
d. R-37 mR/hr
e. R-38 R/hr
f. SPING 02
   1. 02-05 µCi/cc
   2. 02-07 cpm
   3. 02-09 cpm

5.1.3 Steam Line Release

a. R-15 cpm
b. R-19 cpm
c. 1A Steam Line
   1. R-31 mR/hr
   2. R-32 R/hr
d. 1B Steam Line
   1. R-33 mR/hr
   2. R-34 R/hr

5.2 DETERMINE THE APPROPRIATE SAMPLE POINT for the release path identified.

Note

IF it is NOT possible to obtain in-plant grab samples for any reason, THEN go directly to Step 5.7.

5.2.1 Auxiliary Building Stack Release

a. R-13 or R-14 (657') - Step 5.4

b. Auxiliary SPING (642') - Step 5.5

5.2.2 Containment Stack Release

a. R-11/12 (657') - Step 5.4

b. R-21 (657') - Step 5.4

c. Cont. SPING (642') - Step 5.5

5.2.3 Steam Relief's Release - go to Step 5.6

5.3 PRESAMPLE PREPARATION

5.3.1 Follow the instructions in Procedure EPIP-RET-02D, "Emergency Radiation Entry Controls and Implementation" for entries into the controlled area.

5.3.2 Obtain the following equipment:

a. Marinelli Beaker

b. Particulate Filter Papers

c. Silver Zeolite Cartridges

d. Gas Syringes (10 cc, 1 cc, 0.1 cc)

!! Caution !!

To Prevent Injury or Death

IF radiation monitor R-35 reads greater than 1000mR/hr, THEN discontinue sampling both the containment stack at R-11/12 or R-21 and the Auxiliary Building stack at the R-13 or R-14 sample locations on the 657′ elevation of the Auxiliary Building.

Note

Use Silver Zeolite filters for sampling for iodine radioisotopes.

5.4 OBTAIN AUXILIARY BUILDING OR CONTAINMENT STACK RELEASE DATA BY GRAB SAMPLE

5.4.1 Containment Stack Release

a. If required, obtain grab sample from R-21 or R-11/12 for the determination of the effluent isotopic mixture.

b. Record the sample analysis conditions and results on Form EPIPF-RET-02B-01.

c. Calculate the total flow rate (cc/sec) by adding the operating fan flow rates at the top of Form EPIPF-RET-02B-01.

d. Calculate the release rate (Ci/sec) for each isotope by multiplying the release concentration (μCi/cc) by total fan flow rate (cc/sec) and by 10^{-6} (Ci/μCi). Record the results on Form EPIPF-RET-02B-01.

e. Transmit completed Form EPIPF-RET-02B-01 to the Radiological Protection Director (RPD) or Environmental Protection Director (EPD) for review and/or input to dose projection using EPIP-ENV-03C.

5.4.2 IF grab samples are required from the Auxiliary Building stack for determination of the effluent isotopic mixture, THEN collect the samples at either the R-13 or R-14 sampler located on the 657′ elevation of the Auxiliary Building. Use normal sampling procedures HP-05.004, "Radiation/Contamination Survey and Airborne Radioactivity Sampling Schedules," and HP-05.020, "Counting of High Activity Samples," for sampling the Auxiliary Building stack.

5.4.3 Calculate a Radiological Release Rate for an Auxiliary Building Stack Release.

a. Record the sampling analysis on Form EPIPF-RET-02B-02.

b. Calculate the total flow (cc/sec) by adding the operating fan flow rates at the top of Form EPIPF-RET-02B-02.

Note

Use back of form for alternate methods of calculating the total fan flow rate for the Auxiliary Building.

c. Calculate the release rate (Ci/sec) for each isotope by multiplying the release concentration (μCi/cc) by the total fan flow rate (cc/sec) and by 10^{-6} (Ci/μCi). Record the results on Form EPIPF-RET-02B-02.

5.4.4 Transmit completed Form EPIPF-RET-02B-02 to RPD or EPD for review and/or input to dose projection using EPIP-ENV-03C.

5.5 OBTAIN AUXILIARY BUILDING or CONTAINMENT STACK RELEASE DATA FROM SPING (if required).

Note

SPING 01 designates Auxiliary Building Stack (See Form EPIPF-RET-02B-03).
SPING 02 designates Containment Stack (See Form EPIPF-RET-02B-04).

5.5.1 Calculate the Total Stack Flow Rate using Form EPIPF-RET-02B-03 and/or Form EPIPF-RET-02B-04.

5.5.2 Noble Gas Release

a. View the current value of the Low, Medium, and High Range Gas channels (0105 or 0107 or 0109 for Auxiliary Building).
Gas channels (0205 or 0207 or 0209 for Containment).

b. Select which channel to use based on the channel status information.
Do NOT use a channel in the "FAIL HI" condition.

c. The Low Range Gas Channel (0105 or 0205) reads in μCi/cc. IF this channel is used to determine the release, THEN go to Step 5.5.2.e to calculate the noble gas release rate.

d. The mid (0107 or 0207) and high (0109 or 0209) range channels read out in cpm.

Convert the cpm reading obtained in "a" above to μCi/cc soup using the conversion tables attached to the SPING-4 consoles (See Table 3 or Table 4).

e. Multiply the concentration in µCi/cc by the stack flow rate in cc/sec and by 10^{-6} Ci/µCi to determine the NOBLE GAS RELEASE RATE (Ci/sec).

5.5.3 Iodine Release Data

a. Identify the start of the release from the DNA history plot from the RADSERV control terminal.

Calculate the sample volume (X) × (Y) = Z

X = Time from start of release to either present time or to knee of plateau (maximum printout value) in minutes.

Y = Flow of the sample pump (4.5E + 4 cc/min)

Z = Total volume in (cc)

b. Enter this on Form EPIPF-RET-02B-03 or EPIPF-RET-02B-04.

c. Determine iodine release concentration (µCi/cc) by dividing the maximum value from the history printout (µCi) by the sample volume in Step 5.5.3.a.

d. Enter this on Form EPIPF-RET-02B-03 or EPIPF-RET-02B-04.

e. Determine the iodine release rate (Ci/sec) by multiplying the iodine release concentration (µCi/cc) by the total flow rate (cc/sec) and by 10^{-6} Ci/µCi.

5.5.4 Calculate and record the total (gross) release rate (Ci/sec) by adding the Noble Gas Release Rate (5.5.2.e) and the Iodine Release Rate (5.5.3.e) on Form EPIPF-RET-02B-03 for Auxiliary Building and/or Form EPIPF-RET-02B-04 for containment.

5.5.5 Calculate and record the % Noble Gas and/or % Iodine by following equation:

$$\frac{\textit{Noble Gas or Iodine Release Rate}}{\textit{Total Release Rate}} \times 100 = \%\text{ NG or }\%\text{ I}$$

5.5.6 Transmit completed Form EPIPF-RET-02B-02 to the RPD or EPD for review and/or input to dose projection using EPIP-ENV-03C.

5.6 OBTAIN STEAM LINE RELIEF VALVE RELEASE DATA

5.6.1 Contact the Technical Support Center for the release path (valves open). Determine the Steam Line Monitor readings. Record reading on Form EPIPF-RET-02B-05 (X value).

5.6.2 Select the proper calibration factor from Table 2, based on T=0 being the time of the reactor trip, to calculate the gross release concentration. Record calibration factor on Form EPIPF-RET-02B-05 (Y value).

5.6.3 Contact the Technical Support Center for steam release flow rate calculation results in cc/sec, per Form EPIPF-TSC-08A-04. Record steam flow rate on Form EPIPF-RET-02B-05 (Z value).

5.6.4 Refer to EPIP-RET-05, "Site Boundary Dose Rates During Controlled Plant Shutdown" for additional information when the power operated relief valves are used for plant cooldown.

5.6.5 Calculate release concentration (μCi/cc) using equation on Form EPIPF-RET-02B-05.

5.6.6 Calculate release rate (Ci/sec) using equation on Form EPIPF-RET-02B-05.

5.6.7 Transmit completed Form EPIPF-RET-02B-05 to RPD or EPD for review and/or input to dose projection using EPIP-ENV-03C.

5.7 OBTAIN FIELD SAMPLES (if required)

5.7.1 IF it is impossible or impractical to obtain in-plant grab samples as previously described, THEN actual grab samples in the field taken in the downwind plume will be required.

5.7.2 Inform the EPD of the need for field grab samples and to coordinate this task.

5.7.3 Field samples collected by the Environmental Monitoring (EM) Teams are ultimately delivered to the plant for counting and isotopic analysis.

5.7.4 Record field sample analysis results on Form EPIPF-RET-02B-06, and attach all sample results.

5.7.5 Transmit completed Form EPIPF-RET-02B-06 and attachments to RPD or EPD for review and/or input to dose projection using EPIP-ENV-03C.

6.0 Final Conditions

6.1 Plant Emergency has been Terminated or Recovery actions have begun and the Emergency Response Manager has suspended the use of EPIPs.

7.0 References

7.1 COMTRAKS 89-028, 89-207, 89-208, and PLS 84-005

7.2 EPIP-AD-11, Emergency Radiation Controls

7.3 EPIP-AD-18, Potassium Iodide Distribution

7.4 EPIP-ENV-03C, Dose Projection Using RASCAL Software

7.5 EPIP-RET-02D, Emergency Radiation Entry Controls and Implementation

7.6 EPIP-RET-05, Site Boundary Dose Rates During Controlled Plant Cooldown

7.7 Form EPIPF-TSC-08A-04, Steam Release Data/Calculation Sheet (STMRLS Program)

7.8 HP-05.004, Radiation/Contamination Survey and Airborne Radioactivity Sampling Schedules

7.9 HP-05.018, SPING-4 Filter Change-Post-Accident

7.10 HP-05.020, Counting of High Activity Samples

7.11 HP-05.017, RADSERV Control Terminal

8.0 Records

8.1 The following QA records and non-QA records are identified in this directive/procedure and are listed on the KNPP Records Retention Schedule. These records shall be maintained according to the KNPP Records Management Program.

8.1.1 QA Records

- Containment Stack Release (Grab Sample), Form EPIPF-RET-02B-01
- Auxiliary Building Stack Release (Grab Sample), Form EPIPF-RET-02B-02
- Auxiliary Building Stack Release (Sping Reading), Form EPIPF-RET-02B-03
- Containment Stack Release (Sping Reading), Form EPIPF-RET-02B-04
- Steam Release, Form EPIPF-RET-02B-05
- Field Reading (Grab Sample), Form EPIPF-RET-02B-06

8.1.2 Non-QA Records

None

ISOTOPE NORMALIZATION FACTOR

f_i

ISOTOPES	TIME SINCE REACTOR TRIP							
I	0.0 HOUR	0.5 HOUR	1.0 HOUR	2.0 HOURS	8.0 HOURS	1 DAY	1 WEEK	1 MONTH
1. KR-83M	0.0171	0.02340	0.02470	0.02130	0.00643	0.00010	-0-	-0-
2. KR-85	0.0010	0.00148	0.00167	0.00148	0.00231	0.00280	0.00690	0.1230
3. KR-85M	0.0383	0.05460	0.05730	0.05330	0.02670	0.02650	-0-	-0-
4. KR-87	0.0705	0.08580	0.07140	0.04550	0.00218	-0-	-0-	-0-
5. KR-88	0.1010	0.13300	0.13200	0.11600	0.03280	0.00077	-0-	-0-
6. KR-89	0.1210	0.00027	-0-	-0-	-0-	-0-	-0-	-0-
7. XE-131M	0.0010	0.00156	0.00167	0.00184	0.00231	0.00280	0.00545	0.0350
8. XE-133	0.2670	0.41300	0.46700	0.51400	0.64300	0.75200	0.94400	0.8410
9. XE-133M	0.0408	0.06320	0.07140	0.07750	0.09350	0.09810	0.04360	0.0065
10. XE-135	0.0605	0.09360	0.11500	0.13600	0.17000	0.11200	-0-	-0-
11. XE-135M	0.0554	0.04450	0.03610	0.03200	0.02060	0.00471	-0-	-0-
12. XE-138	0.2270	0.08580	0.02200	0.00126	-0-	-0-	-0-	-0-
$\sum_{I=1}^{12} fi\,(DF)\,I$	7.2400	4.26800	3.54200	2.93000	1.26800	0.57730	0.33040	0.2887

GROSS RELEASE RATE DETERMINATION FOR STEAM RELEASES

GROSS RELEASE RATE

$$X \times Y \times Z \times 10^{-6} \frac{Ci}{\mu Ci} = \underline{\hspace{6em}} \text{ Ci/sec}$$

WHERE:

X = Steam Header Monitor Reading $\left(\frac{R}{hr}\right)$

Y = Calibration Factor $\left(\frac{\mu Ci/cc}{R/hr}\right)$

Z = Flow Rate $\left(\frac{cc}{sec}\right)$

CALIBRATION FACTOR TABLE

TIME (SINCE REACTOR TRIP)	$\frac{\mu Ci/cc}{R/hr}$
-0-	14.5
1 HOUR	16.7
2 HOURS	20.3
4 HOURS	30.4
8 HOURS	67.9
1 DAY	887
1 WEEK	3.08×10^4
1 MONTH	1.93×10^4

MID RANGE

SPING MID-RANGE CPM TO µCi/cc SOUP

TIME AFTER RX TRIP	0.0 HOUR	0.5 HOUR	1.0 HOUR	2.0 HOURS	8.0 HOURS	24 HOURS	1 WEEK	1 MONTH
MID-RANGE CPM	µCi/cc	µCi/cc	µCi/cc	µCi/cc	µCi/cc	µCi/cc	µCi/cc	µCi/cc
1.00E+01	7.40E-04	1.30E-03	1.50E-03	1.80E-03	4.20E-03	9.30E-03	1.60E-02	1.90E-02
2.00E+01	1.48E-03	2.60E-03	3.00E-03	3.60E-03	8.40E-03	1.86E-02	3.20E-02	3.80E-02
3.00E+01	2.22E-03	3.90E-03	4.50E-03	5.40E-03	1.26E-02	2.79E-02	4.80E-02	5.70E-02
4.00E+01	2.96E-03	5.20E-03	6.00E-03	7.20E-03	1.68E-02	3.72E-02	6.40E-02	7.60E-02
5.00E+01	3.70E-03	6.50E-03	7.50E-03	9.00E-03	2.10E-02	4.65E-02	8.00E-02	9.50E-02
6.00E+01	4.44E-03	7.80E-03	9.00E-03	1.08E-02	2.25E-02	5.58E-02	9.60E-02	1.14E-01
7.00E+01	5.18E-03	9.10E-03	1.05E-02	1.26E-02	2.94E-02	6.51E-02	1.12E-01	1.33E-01
8.00E+01	5.92E-03	1.04E-02	1.20E-02	1.44E-02	3.36E-02	7.44E-02	1.28E-01	1.52E-01
9.00E+01	6.66E-03	1.17E-02	1.35E-02	1.62E-02	3.78E-02	8.37E-02	1.44E-01	1.71E-01
1.00E+02	7.40E-03	1.30E-02	1.50E-02	1.80E-02	4.20E-02	9.30E-02	1.60E-01	1.90E-01
2.00E+02	1.48E-02	2.60E-02	3.00E-02	3.60E-02	8.40E-02	1.86E-01	3.20E-01	3.80E-01
3.00E+02	2.22E-02	3.90E-02	4.50E-02	5.40E-02	1.26E-01	2.79E-01	4.80E-01	5.70E-01
4.00E+02	2.96E-02	5.20E-02	6.00E-02	7.20E-02	1.68E-01	3.72E-01	6.40E-01	7.60E-01
5.00E+02	3.70E-02	6.50E-02	7.50E-02	9.00E-02	2.10E-01	4.65E-01	8.00E-01	9.50E-01
6.00E+02	4.44E-02	7.80E-02	9.00E-02	1.08E-01	2.25E-01	5.58E-01	9.60E-01	1.14E+00
7.00E+02	5.18E-02	9.10E-02	1.05E-01	1.26E-01	2.94E-01	6.51E-01	1.12E+00	1.33E+00
8.00E+02	5.92E-02	1.04E-01	1.20E-01	1.44E-01	3.36E-01	7.44E-01	1.28E+00	1.52E+00
9.00E+02	6.66E-02	1.17E-01	1.35E-01	1.62E-01	3.78E-01	8.37E-01	1.44E+00	1.71E+00
1.00E+03	7.40E-02	1.30E-01	1.50E-01	1.80E-01	4.20E-01	9.30E-01	1.60E+00	1.90E+00
2.00E+03	1.48E-01	2.60E-01	3.00E-01	3.60E-01	8.40E-01	1.86E+00	3.20E+00	3.80E+00
3.00E+03	2.22E-01	3.90E-01	4.50E-01	5.40E-01	1.26E+00	2.79E+00	4.80E+00	5.70E+00
4.00E+03	2.96E-01	5.20E-01	6.00E-01	7.20E-01	1.68E+00	3.72E+00	6.40E+00	7.60E+00
5.00E+03	3.70E-01	6.50E-01	7.50E-01	9.00E-01	2.10E+00	4.65E+00	8.00E+00	9.50E+00
6.00E+03	4.44E-01	7.80E-01	9.00E-01	1.08E+00	2.25E+00	5.58E+00	9.60E+00	1.14E+01
7.00E+03	5.18E-01	9.10E-01	1.05E+00	1.26E+00	2.94E+00	6.51E+00	1.12E+01	1.33E+01
8.00E+03	5.92E-01	1.04E+00	1.20E+00	1.44E+00	3.36E+00	7.44E+00	1.28E+01	1.52E+01
9.00E+03	6.66E-01	1.17E+00	1.35E+00	1.62E+00	3.78E+00	8.37E+00	1.44E+01	1.71E+01
1.00E+04	7.40E-01	1.30E+00	1.50E+00	1.80E+00	4.20E+00	9.30E+00	1.60E+01	1.90E+01
2.00E+04	1.48E+00	2.60E+00	3.00E+00	3.60E+00	8.40E+00	1.86E+01	3.20E+01	3.80E+01
3.00E+04	2.22E+00	3.90E+00	4.50E+00	5.40E+00	1.26E+01	2.79E+01	4.80E+01	5.70E+01
4.00E+04	2.96E+00	5.20E+00	6.00E+00	7.20E+00	1.68E+01	3.72E+01	6.40E+01	7.60E+01
5.00E+04	3.70E+00	6.50E+00	7.50E+00	9.00E+00	2.10E+01	4.65E+01	8.00E+01	9.50E+01
6.00E+04	4.44E+00	7.80E+00	9.00E+00	1.08E+01	2.25E+01	5.58E+01	9.60E+01	1.14E+02
7.00E+04	5.18E+00	9.10E+00	1.05E+01	1.26E+01	2.94E+01	6.51E+01	1.12E+02	1.33E+02
8.00E+04	5.92E+00	1.04E+01	1.20E+01	1.44E+01	3.36E+01	7.44E+01	1.28E+02	1.52E+02
9.00E+04	6.66E+00	1.17E+01	1.35E+01	1.62E+01	3.78E+01	8.37E+01	1.44E+02	1.71E+02
1.00E+05	7.40E+00	1.30E+01	1.50E+01	1.80E+01	4.20E+01	9.30E+01	1.60E+02	1.90E+02
2.00E+05	1.48E+01	2.60E+01	3.00E+01	3.60E+01	8.40E+01	1.86E+02	3.20E+02	3.80E+02
3.00E+05	2.22E+01	3.90E+01	4.50E+01	5.40E+01	1.26E+02	2.79E+02	4.80E+02	5.70E+02
4.00E+05	2.96E+01	5.20E+01	6.00E+01	7.20E+01	1.68E+02	3.72E+02	6.40E+02	7.60E+02
5.00E+05	3.70E+01	6.50E+01	7.50E+01	9.00E+01	2.10E+02	4.65E+02	8.00E+02	9.50E+02
6.00E+05	4.44E+01	7.80E+01	9.00E+01	1.08E+02	2.25E+02	5.58E+02	9.60E+02	1.14E+03
7.00E+05	5.19E+01	9.10E+01	1.05E+02	1.26E+02	2.94E+02	6.51E+02	1.12E+03	1.33E+03
8.00E+05	5.92E+01	1.04E+02	1.20E+02	1.44E+02	3.36E+02	7.44E+02	1.28E+03	1.52E+03
9.00E+05	6.66E+01	1.17E+02	1.35E+02	1.62E+02	3.78E+02	8.37E+02	1.44E+03	1.71E+03
1.00E+06	7.40E+01	1.30E+02	1.50E+02	1.80E+02	4.20E+02	9.30E+02	1.60E+03	1.90E+03

HIGH RANGE

SPING HIGH RANGE CPM TO µCi/cc SOUP

TIME AFTER RX TRIP	0.0 HOUR	0.5 HOUR	1.0 HOUR	2.0 HOURS	8.0 HOURS	24 HOURS	1 WEEK	1 MONTH
HIGH-RANGE CPM	µCi/cc	µCi/cc	µCi/cc	µCi/cc	µCi/cc	µCi/cc	µCi/cc	µCi/cc
1.00E + 01	7.00E - 01	1.18E + 00	1.43E + 00	1.72E + 00	3.98E + 00	8.70E + 00	1.54E + 01	1.75E + 01
2.00E + 01	1.40E + 00	2.36E + 00	2.86E + 00	3.44E + 00	7.96E + 00	1.74E + 01	3.08E + 01	3.50E + 01
3.00E + 01	2.10E + 00	3.54E + 00	4.29E + 00	5.16E + 00	1.19E + 01	2.61E + 01	4.62E + 01	5.25E + 01
4.00E + 01	2.80E + 00	4.72E + 00	5.72E + 00	6.88E + 00	1.59E + 01	3.48E + 01	6.16E + 01	7.00E + 01
5.00E + 01	3.50E + 00	5.90E + 00	7.15E + 00	8.60E + 00	1.99E + 01	4.35E + 01	7.70E + 01	8.75E + 01
6.00E + 01	4.20E + 00	7.08E + 00	8.58E + 00	1.03E + 01	2.39E + 01	5.22E + 01	9.24E + 01	1.05E + 02
7.00E + 01	4.90E + 00	8.26E + 00	1.00E + 01	1.20E + 01	2.79E + 01	6.09E + 01	1.08E + 02	1.23E + 02
8.00E + 01	5.60E + 00	9.44E + 00	1.14E + 01	1.38E + 01	3.18E + 01	6.96E + 01	1.23E + 02	1.40E + 02
9.00E + 01	6.30E + 00	1.06E + 01	1.29E + 01	1.55E + 01	3.58E + 01	7.83E + 01	1.39E + 02	1.58E + 02
1.00E + 02	7.00E + 00	1.18E + 01	1.43E + 01	1.72E + 01	3.98E + 01	8.70E + 01	1.54E + 02	1.75E + 02
2.00E + 02	1.40E + 01	2.36E + 01	2.86E + 01	3.44E + 01	7.96E + 01	1.74E + 02	3.08E + 02	3.50E + 02
3.00E + 02	2.10E + 01	3.54E + 01	4.29E + 01	5.16E + 01	1.19E + 02	2.61E + 02	4.62E + 02	5.25E + 02
4.00E + 02	2.80E + 01	4.72E + 01	5.72E + 01	6.88E + 01	1.59E + 02	3.48E + 02	6.16E + 02	7.00E + 02
5.00E + 02	3.50E + 01	5.90E + 01	7.15E + 01	8.60E + 01	1.99E + 02	4.35E + 02	7.70E + 02	8.75E + 02
6.00E + 02	4.20E + 01	7.08E + 01	8.58E + 01	1.03E + 02	2.39E + 02	5.22E + 02	9.24E + 02	1.05E + 03
7.00E + 02	4.90E + 01	8.26E + 01	1.00E + 02	1.20E + 02	2.79E + 02	6.09E + 02	1.08E + 03	1.23E + 03
8.00E + 02	5.60E + 01	9.44E + 01	1.14E + 02	1.38E + 02	3.18E + 02	6.96E + 02	1.23E + 03	1.40E + 03
9.00E + 02	6.30E + 01	1.06E + 02	1.29E + 02	1.55E + 02	3.58E + 02	7.83E + 02	1.39E + 03	1.58E + 03
1.00E + 03	7.00E + 01	1.18E + 02	1.43E + 02	1.72E + 02	3.98E + 02	8.70E + 02	1.54E + 03	1.75E + 03
2.00E + 03	1.40E + 02	2.36E + 02	2.86E + 02	3.44E + 02	7.96E + 02	1.74E + 03	3.08E + 03	3.50E + 03
3.00E + 03	2.10E + 02	3.54E + 02	4.29E + 02	5.16E + 02	1.19E + 03	2.61E + 03	4.62E + 03	5.25E + 03
4.00E + 03	2.80E + 02	4.72E + 02	5.72E + 02	6.88E + 02	1.59E + 03	3.48E + 03	6.16E + 03	7.00E + 03
5.00E + 03	3.50E + 02	5.90E + 02	7.15E + 02	8.60E + 02	1.99E + 03	4.35E + 03	7.70E + 03	8.75E + 03
6.00E + 03	4.20E + 02	7.08E + 02	8.58E + 02	1.03E + 03	2.39E + 03	5.22E + 03	9.24E + 03	1.05E + 04
7.00E + 03	4.90E + 02	8.26E + 02	1.00E + 03	1.20E + 03	2.79E + 03	6.09E + 03	1.08E + 04	1.23E + 04
8.00E + 03	5.60E + 02	9.44E + 02	1.14E + 03	1.38E + 03	3.18E + 03	6.96E + 03	1.23E + 04	1.40E + 04
9.00E + 03	6.30E + 02	1.06E + 03	1.29E + 03	1.55E + 03	3.58E + 03	7.83E + 03	1.39E + 04	1.58E + 04
1.00E + 04	7.00E + 02	1.18E + 03	1.43E + 03	1.72E + 03	3.98E + 03	8.70E + 03	1.54E + 04	1.75E + 04
2.00E + 04	1.40E + 03	2.36E + 03	2.86E + 03	3.44E + 03	7.96E + 03	1.74E + 04	3.08E + 04	3.50E + 04
3.00E + 04	2.10E + 03	3.54E + 03	4.29E + 03	5.16E + 03	1.19E + 04	2.61E + 04	4.62E + 04	5.25E + 04
4.00E + 04	2.80E + 03	4.72E + 03	5.72E + 03	6.88E + 03	1.59E + 04	3.48E + 04	6.16E + 04	7.00E + 04
5.00E + 04	3.50E + 03	5.90E + 03	7.15E + 03	8.60E + 03	1.99E + 04	4.35E + 04	7.70E + 04	8.75E + 04
6.00E + 04	4.20E + 03	7.08E + 03	8.58E + 03	1.03E + 04	2.39E + 04	5.22E + 04	9.24E + 04	1.05E + 05
7.00E + 04	4.90E + 03	8.26E + 03	1.00E + 04	1.20E + 04	2.79E + 04	6.09E + 04	1.08E + 05	1.23E + 05
8.00E + 04	5.60E + 03	9.44E + 03	1.14E + 04	1.38E + 04	3.18E + 04	6.96E + 04	1.23E + 05	1.40E + 05
9.00E + 04	6.30E + 03	1.06E + 04	1.29E + 04	1.55E + 04	3.58E + 04	7.83E + 04	1.39E + 05	1.58E + 05
1.00E + 05	7.00E + 03	1.18E + 04	1.43E + 04	1.72E + 04	3.98E + 04	8.70E + 04	1.54E + 05	1.75E + 05
2.00E + 05	1.40E + 04	2.36E + 04	2.86E + 04	3.44E + 04	7.96E + 04	1.74E + 05	3.08E + 05	3.50E + 05
3.00E + 05	2.10E + 03	3.54E + 04	4.29E + 04	5.16E + 04	1.19E + 05	2.61E + 05	4.62E + 05	5.25E + 05
4.00E + 05	2.80E + 04	4.72E + 04	5.72E + 04	6.88E + 04	1.59E + 05	3.48E + 05	6.16E + 05	7.00E + 05
5.00E + 05	3.50E + 04	5.90E + 04	7.15E + 04	8.60E + 04	1.99E + 05	4.35E + 05	7.70E + 05	8.75E + 05
6.00E + 05	4.20E + 04	7.08E + 04	8.58E + 04	1.03E + 05	2.39E + 05	5.22E + 05	9.24E + 05	1.05E + 06
7.00E + 05	4.90E + 04	8.26E + 04	1.00E + 05	1.20E + 05	2.79E + 05	6.09E + 05	1.08E + 06	1.23E + 06
8.00E + 05	5.60E + 04	9.44E + 04	1.14E + 05	1.38E + 05	3.18E + 05	6.96E + 05	1.23E + 06	1.40E + 06
9.00E + 05	6.30E + 04	1.06E + 05	1.29E + 05	1.55E + 05	3.58E + 05	7.83E + 05	1.39E + 06	1.58E + 06
1.00E + 06	7.00E + 04	1.18E + 05	1.43E + 05	1.72E + 05	3.98E + 05	8.70E + 05	1.54E + 06	1.75E + 06

PROCEDURAL FLOW CHART

Identify Release Path:	Auxiliary Building Stack	Containment Stack	Steam Reliefs
Verify By Monitoring These Indications	R-13 cpm R-14 cpm R-22 cpm R-35 mR/hr R-36 R/hr SPING 01-05 μCi/cc 01-07 cpm 01-09 cpm	R-11 cpm R-12 cpm R-21 cpm R-37 mR/hr R-38 R/hr SPING 02-05 μCi/cc 02-07 cpm 02-09 cpm	R-15 cpm R-19 cpm 1A Steam Line R-31 mR/hr R-32 R/hr 1B Steam Line R-33 mR/hr R-34 R/hr
Get Grab Samples At	Auxiliary SPING (642') Field Samples R-13 (657') R-14 (657')	R-11/12 (657') R-21 (657') Cont. SPING (642') Coordinate Field Sampling with EPD	Coordinate Field Sampling with EPD
Determine Effluent Release Rate (cc/sec)	1A Auxiliary Exh. 1B Auxiliary Exh. 1A Zone SV 1B Zone SV 1A SFP 1B SFP	RBV 1A SBV 1B SBV	Get From TSC, per Form TSC-08-02
Transmit Obtained Data	EPD or EOF/RAF Liaison	EPD or EOF/RAF Liaison	EPD or EOF/RAF Liaison
OR Perform Dose Projection	EPIP-ENV-03C	EPIP-ENV-03C	EPIP-ENV-03C

FOR EASE OF DATA RETRIEVAL, SEE:

a) Honeywell Graphic Output 12 - For All SPING Data
b) Honeywell Graphic Output 17 and 18 - For Selected Non-SPING Data

HONEYWELL POINT NO.	MONITOR NO.	UNITS	DESCRIPTION / LOCATION
G0001G	R-1	mR/hr	Control Room
G0002G	R-2	mR/hr	Containment, 649'
G0004G	R-4	mR/hr	Charging Pump
G0005G	R-5	mR/hr	Spent Fuel Pools, 649'
G0006G	R-6	mR/hr	Sampling Room
G0007G	R-7	mR/hr	Seal Table
G0009G	R-9	R/hr	Letdown Line
G0010G	R-10	mR/hr	New Fuel Pit
G0011G	R-11	cpm	Containment Air Particulate
G0012G	R-12	cpm	Containment Gas
G0013G	R-13	cpm	Auxiliary Building Vent
G0014G	R-14	cpm	Auxiliary Building Vent Stack
G0015G	R-15	cpm	Air Ejector
G0016G	R-16	cpm	Containment Fan Coil Southwest Return
G0017G	R-17	cpm	Component Cooling Southwest Return
G0018G	R-18	cpm	Liquid Waste Disposal Line
G0019G	R-19	cpm	S/G Blowdown
G0020G	R-20	cpm	Service Water Standpipe
G0031G	R-31	mR/hr	1A Steam Line - LO
G0102G	R-32	R/hr	1A Steam Line - HI
G0105G	R-33	mR/hr	1B Steam Line - LO
G0104G	R-34	R/hr	1B Steam Line - HI
G0101G	R-35	mR/hr	Auxiliary Building Vent Duct - LO
G0100G	R-36	R/hr	Auxiliary Building Vent Duct - HI
G0113G	R-37	mR/hr	Containment Vent Duct - LO
G0112G	R-38	R/hr	Containment Vent Duct - HI
G0114G	R-40	R/hr	1A Containment HI-Range Monitor
G0115G	R-41	R/hr	1B Containment HI-Range Monitor
EBERLINE SPING AREA MONITORS			
G9041G	08-01	mR/hr	586' Sulfuric Acid Tank
G9042G	08-02	R/hr	586' Containment Spray Pumps
G9043G	03-01	R/hr	586' Waste Disposal Panel
G9044G	03-02	mR/hr	586' Hi-Rad Sample Room
G9045G	05-01	R/hr	586' RHR Pump Pit Area
G9046G	05-02	mR/hr	586' Waste Compactor
G9048G	08-03	mR/hr	606' Heating Boiler
G9049G	04-01	mR/hr	606' Machine Shop
G9050G	04-02	mR/hr	606' RPO/Laundry Room
EBERLINE SPING AREA MONITORS (Continued on Page 2 of 2)			

HONEYWELL POINT NO.	MONITOR NO.	UNITS	DESCRIPTION / LOCATION
EBERLINE SPING AREA MONITORS (Continued)			
G9051G	03-03	R/hr	606′ Elevator/CC Heat Exchanger
G9052G	05-03	mR/hr	606′ Loading Dock
G9053G	04-03	mR/hr	626′ Makeup Demineralizer Area
G9055G	04-04	mR/hr	626′ I&C Shop - South
G9056G	06-01	mR/hr	626′ I&C Shop - North
G9057G	06-02	R/hr	642′ Shield Building Filter Area
G9058G	06-03	mR/hr	642′ Control Room A/C Room
G9059G	06-04	R/hr	657′ Containment Vent Filter Area
G9060G	06-05	R/hr	657′ Zone SV Filter Area
G9061G	07-03	mR/hr	606′ TSC Building Stairwell
G9062G	07-01	mR/hr	586′ RAF Count Room
G9063G	07-02	mR/hr	586′ Technical Support Center
G9066G	07-04	cpm	586′ Technical Support Center
G9067G	03-04	cpm	586′ Hi-Rad Sample Room
G9069G	03-06	cpm	586′ Charging Pump Area
G9070G	03-08	cpm	606′ VCT Area
CONTAINMENT STACK SPING			
G9071G	02-01	μCi	Beta Particle
G9072G	02-02	cpm	Alpha Particulate
G9073G	02-03	μCi	Iodine
G9074G	02-04	cpm	Iodine Background
G9075G	02-05	μCi/cc	Low-Range Noble Gas
G9076G	02-06	mR/hr	Area Monitor - Gamma
G9077G	02-07	cpm	Mid-Range Noble Gas
G9078G	02-08	cpm	Background - Gas Channels
G9079G	02-09	cpm	Hi-Range Noble Gas
AUXILIARY BUILDING STACK FLOW			
F7001G	N/A	cfm	Auxiliary Building Stack Flow
N/A	06-10	milli-amps	Auxiliary Building Stack Flow (Form EPIPF-RET-02B-02)
AUXILIARY BUILDING STACK SPING			
G9080G	01-01	μCi	Beta Particulate
G9081G	01-02	cpm	Alpha Particulate
G9082G	01-03	μCi	Iodine
G9083G	01-04	cpm	Iodine Background
G9084G	01-05	μCi/cc	Low-Range Noble Gas
G9085G	01-06	mR/hr	Area Monitor - Gamma
G9086G	01-07	cpm	Mid-Range Noble Gas
G9087G	01-08	cpm	Background - Gas Channels
G9088G	01-09	cpm	Hi-Range Noble Gas

1.0 Purpose

1.1 This procedure provides instruction for determining the maximum allowable venting period of the reactor vessel head when noncondensable gases are present in the RCS.

2.0 General Notes

2.1 None

3.0 Precautions and Limitations

3.1 This procedure should be performed concurrently with FR-I.3, "Response to Voids in Reactor Vessel," Section 4.17. Determine maximum allowable venting time.

3.2 The procedure may only be performed when containment hydrogen concentration is below 3%.

4.0 Initial Conditions

4.1 The range selector switch on each monitor must remain in the 0-10% position. The high range is NOT calibrated and will result in false Control Room and computer indication.

4.2 The containment hydrogen analyzer has been placed in service per EPIP-RET-03C, "Post Accident Operation of the High Radiation Sample Room." Allow a minimum sample purge time of 10 minutes.

4.3 All available containment air circulating equipment should be operating to prevent the formation of hydrogen gas pockets and ensure a representative sample is obtained. IF only one containment dome fan is operating, THEN the sample should be taken from the operating fan discharge.

5.0 Procedure

5.1 Obtain the RCS Pressure from the PPCS/SPDS Computer Point ID P0420A or Control Room meters and record it on Form EPIPF-TSC-07-01.

5.2 Obtain the Containment Pressure from the PPCS/SPDS Computer Point IDs P8004A, P8005A, or Control Room meters and record it on Form EPIPF-TSC-07-01.

5.3 Obtain the Containment Hydrogen Concentration from PPCS/SPDS Computer Point IDs X8001A, X8002A, or Control Room meters (41615 and 41616). IF only one containment dome fan is running, THEN use the appropriate hydrogen analyzer channel to get a representative sample. IF both fans are running, THEN use the channel with the higher concentration indication. Record the hydrogen concentration, the dome fans in service, and which analyzer was used on Form EPIPF-TSC-07-01.

5.4 Obtain the Containment Temperature from PPCS/SPDS Computer Point ID T1000A (instrument 15293) and record it on Form EPIPF-TSC-07-01.

5.5 Complete the calculations on Form EPIPF-TSC-07-01 and report the results (maximum head venting time) to the TSCD.

6.0 Final Conditions

6.1 Emergency declaration is terminated, OR

6.2 The Reactor Coolant System has been stabilized and recovery operations have been entered per EPIP-AD-15, "Recovery Planning and Termination."

7.0 References

7.1 "Background Information for Westinghouse Emergency Response Guidelines, FR-I.3 Void in Reactor Vessel," Rev. LP-BASIC, September 15, 1981

7.2 "FR-I.3, Response to Voids in Reactor Vessel," LP-Rev. 1, September 1, 1983

7.3 EPIP-AD-15, Recovery Planning and Termination

7.4 EPIP-RET-03C, Post Accident Operation of the High Radiation Sample Room

7.5 Flow Diagram, "Reactor Building Vent System Post-LOCA Hydrogen Control," M-403

8.0 Records

8.1 The following QA records and non-QA records are identified in this directive/procedure and are listed on the KNPP Records Retention Schedule. These records shall be maintained according to the KNPP Records Management Program.

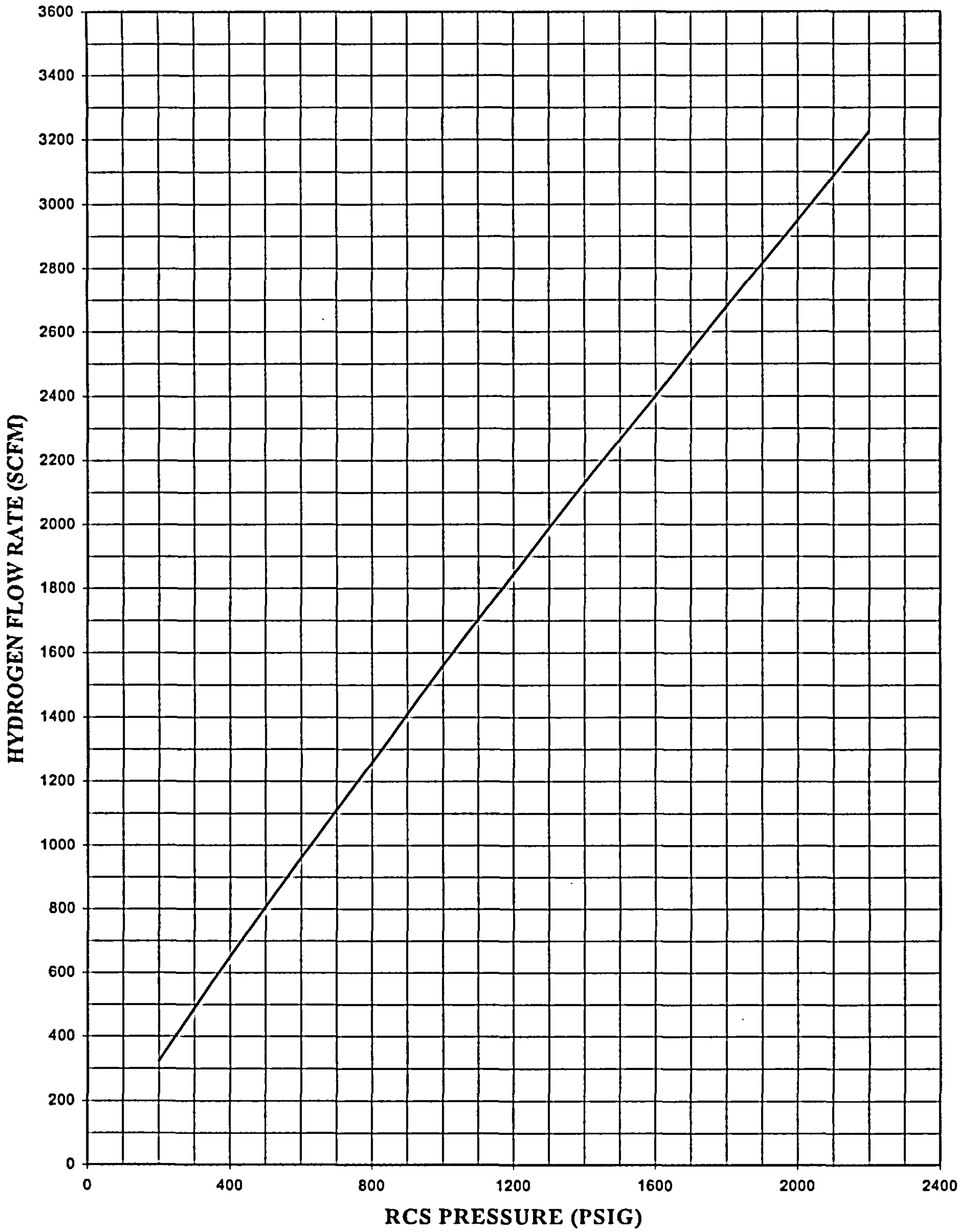
8.1.1 QA Records

- Head Venting Calculation, Form EPIP-TSC-07-01

8.1.2 Non-QA Records

None

HYDROGEN FLOW RATE VERSUS RCS PRESSURE



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Kewaunee Nuclear Power Plant
Emergency Plan Implementing Procedure</td><td colspan="3">No. EPIP-TSC-09A
Title Core Damage Assessment
Date MAY 5 2003</td><td>Rev. K

Page 1 of 11</td></tr>
<tr><td colspan="2">Reviewed By John Helfenberger</td><td colspan="4">Approved By Jerrie Morlino</td></tr>
<tr><td>Nuclear Safety Related</td><td>☐ Yes ☑ No</td><td>PORC Review Required</td><td>☑ Yes ☐ No</td><td>SRO Approval Of Temporary Changes Required</td><td>☑ Yes ☐ No</td></tr>
</table>

1.0 Purpose

1.1 This procedure provides instruction for assessing the degree of core damage during an accident. In addition, the guideline provides information for the assessment of the appropriate Emergency Action Level for off-site radiological protective actions based on the degree of core damage. Specifically, the information contained in this guideline relates to:

- Determination of the degree of damage to the fuel rod cladding that results in the release of the fission product inventory in the fuel rod gap space
- Determination of the degree of core overheating that results in the release of the fission product inventory in the fuel pellets
- Determination of the appropriate Emergency Action Level for off-site radiological protective actions based on the degree of damage to the reactor core

2.0 General Notes

2.1 None

3.0 Precautions and Limitations

3.1 None

4.0 Initial Conditions

4.1 This guideline is to be used when Technical Support Center is activated and a Core Damage Assessment is requested.

5.0 Procedure

5.1 Identify Current Plant Status

5.1.1 Complete Form EPIPF-TSC-09A-05 except for the last column.

5.1.2 Using the table below, determine the possible status of the reactor core.

5.1.3 Record status on Form EPIPF-TSC-09A-05.

5.1.4 Go to the appropriate section of this guideline as indicated from the table.

High Level Core Damage Assessment	
Plant Status	**Fuel Rod Fission Product Status**
Core Exit Thermocouple Temperature LESS THAN 700°F, AND Containment Radiation Below Curve in Attachment A	No Core Damage Continue to Monitor Plant Parameters
Core Exit Thermocouple Temperature LESS THAN 1800°F, AND Containment Radiation Below Curve in Attachment B	Possible Fuel Rod Clad Damage Go to step 5.2
Core Exit Thermocouple Temperature GREATER THAN 1800°F, OR Containment Radiation Above Curve in Attachment B	Possible Fuel Overtemperature Damage Go to step 5.3

5.2 Estimate Fuel Rod Clad Damage

5.2.1 Estimate fuel rod clad damage based on containment radiation levels.

5.2.1.1 Complete lines 1-4 of Form EPIPF-TSC-09A-02.

5.2.2 Estimate fuel rod clad damage based on core exit thermocouple readings.

5.2.2.1 Complete lines 5-10 of Form EPIPF-TSC-09A-02.

5.2.3 Verify reasonableness of clad damage estimates.

5.2.3.1 Complete lines 11-17 of Form EPIPF-TSC-09A-02.

5.2.3.2 IF expected response is not obtained, THEN determine if the deviation can be explained

from the accident progression

- injection of water to the RCS
- bleed paths from the RCS
- direct radiation to the containment radiation monitors, OR

from conservatisms in the predictive model

- fuel burnup
- fission product retention in the RCS
- fission product removal form containment

5.2.3.3 Record explanation of deviations on Form EPIPF-TSC-09A-02.

5.2.4 Report Findings.

5.2.4.1 Fill out Form EPIPF-TSC-09A-07 if level of damage has changed and give to Emergency Director.

5.2.5 Go to Step 5.1.

5.3 Estimate Fuel Overtemperature Damage

5.3.1 Estimate fuel overtemperature damage based on containment radiation levels.

5.3.1.1 Complete lines 1-4 of Form EPIPF-TSC-09A-03.

5.3.2 Estimate fuel overtemperature damage based on core exit thermocouple readings.

5.3.2.1 Complete lines 5-7 of Form EPIPF-TSC-09A-03.

5.3.3 Estimate fuel overtemperature damage based on hydrogen concentration.

5.3.3.1 Complete lines 8-11 of Form EPIPF-TSC-09A-03.

5.3.4 Verify reasonableness of fuel overtemperature damage estimates.

5.3.4.1 Complete lines 12-18 of Form EPIPF-TSC-09A-03.

5.3.4.2 IF expected response is not obtained, THEN determine if the deviation can be explained

from the accident progression

- injection of water to the RCS
- bleed paths from the RCS
- direct radiation to the containment radiation monitors
- hydrogen burn in containment or operation of hydrogen igniters, OR

from conservatisms in the predictive model

- fuel burnup
- fission product retention in the RCS
- fission product removal form containment

5.3.4.3 Record explanations of deviations on Form EPIPF-TSC-09A-03.

5.3.5 Report findings.

5.3.5.1 Fill out Form EPIPF-TSC-09A-07 if level of damage has changed and give to Emergency Director.

5.3.6 Go to Step 5.1.

6.0 Final Conditions

6.1 None

7.0 References

7.1 WCAP-14696-A Revision 1, "Westinghouse Owners Group, Core Damage Assessment Guidance," November 1999

7.2 Letter from John G. Lamb (NRC) to Mark Reddemann (NMC) transmitting the NRC SER for Amendment 160 to the Operating License allowing removal of Post Accident Sampling System from Technical Specifications, Letter No. K-02-006, Dated January 16, 2002

7.3 COMTRAK's 89-026 and 89-027

7.4 Calculation C11403, Determination of Setpoints for EPIP-TSC-09A

7.5 EPIP-EOF-04, EOF Staff Action for Alert or Higher

8.0 Records

8.1 The following QA records and non-QA records are identified in this directive/procedure and are listed on the KNPP Records Retention Schedule. These records shall be maintained according to the KNPP Records Management Program.

8.1.1 QA Records

- Core Exit Thermocouple Data, Form EPIPF-TSC-09A-01
- Fuel Rod Clad Damage Estimate, Form EPIPF-TSC-09A-02
- Fuel Rod Overtemperature Damage Estimate, Form EPIPF-TSC-09A-03
- Core Damage Assessment (Monitoring Data), Form EPIPF-TSC-09A-05
- Core Damage Assessment Results, Form EPIPF-TSC-09A-07

8.1.2 Non-QA Records

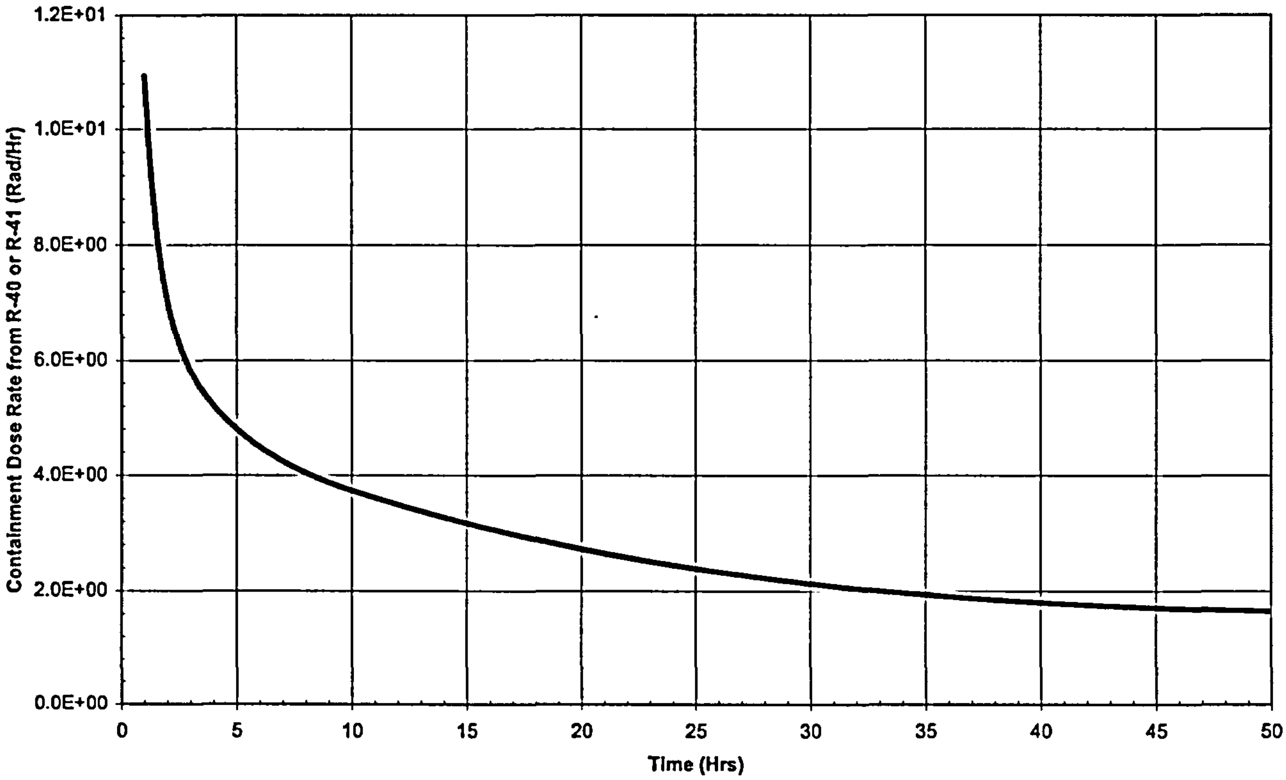
None

LIST OF ATTACHMENTS/FORMS

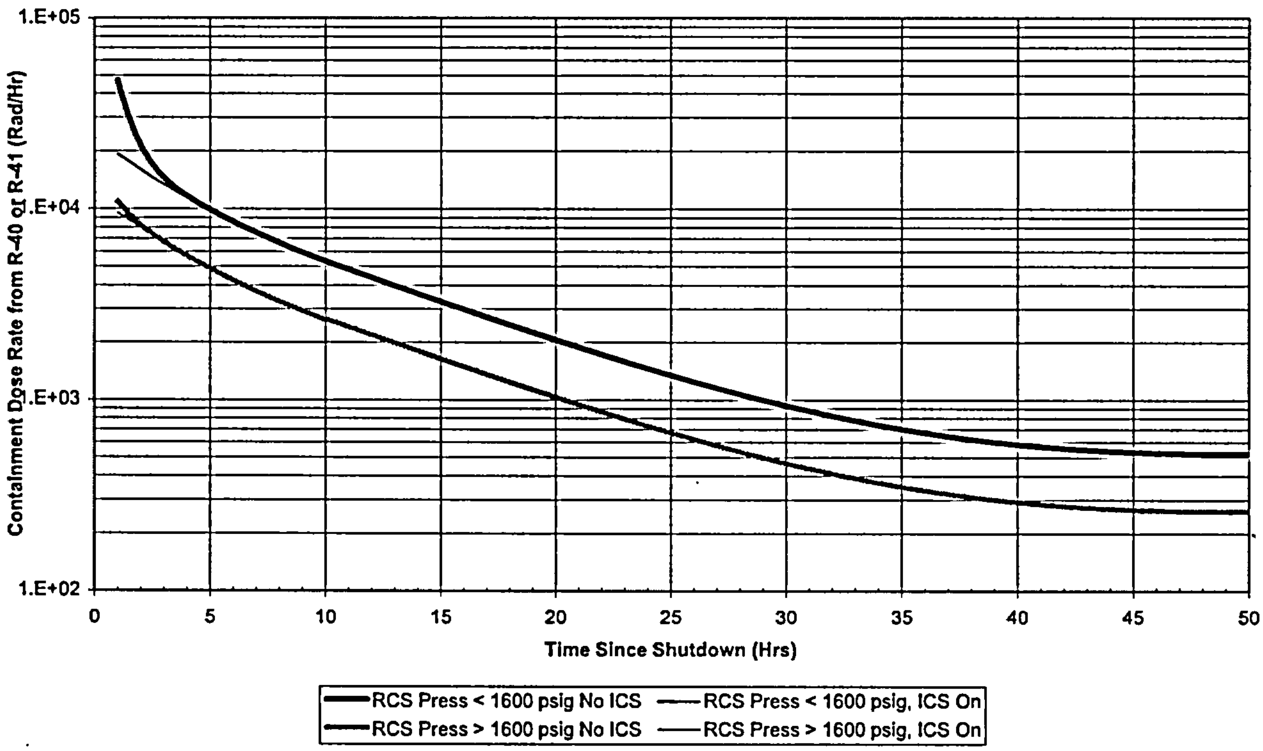
ATTACHMENT	TITLE
Attachment A	Containment Radiation Level vs. Time for RCS Release
Attachment B	Containment Radiation Level vs. Time for 1% Fuel Overtemperature Release
Attachment C	Containment Radiation Level vs. Time for 100% Clad Damage Release
Attachment D	Containment Radiation Level vs. Time for 100% Fuel Overtemperature Release
Attachment E	Hydrogen Concentration for 100% Core Overtemperature Damage

FORM	TITLE
Form EPIPF-TSC-09A-01	Core Exit Thermocouple Data
Form EPIPF-TSC-09A-02	Fuel Rod Clad Damage Estimate
Form EPIPF-TSC-09A-03	Fuel Rod Overtemperature Damage Estimate
Form EPIPF-TSC-09A-05	Core Damage Assessment (Monitoring Data)
Form EPIPF-TSC-09A-07	Core Damage Assessment Results

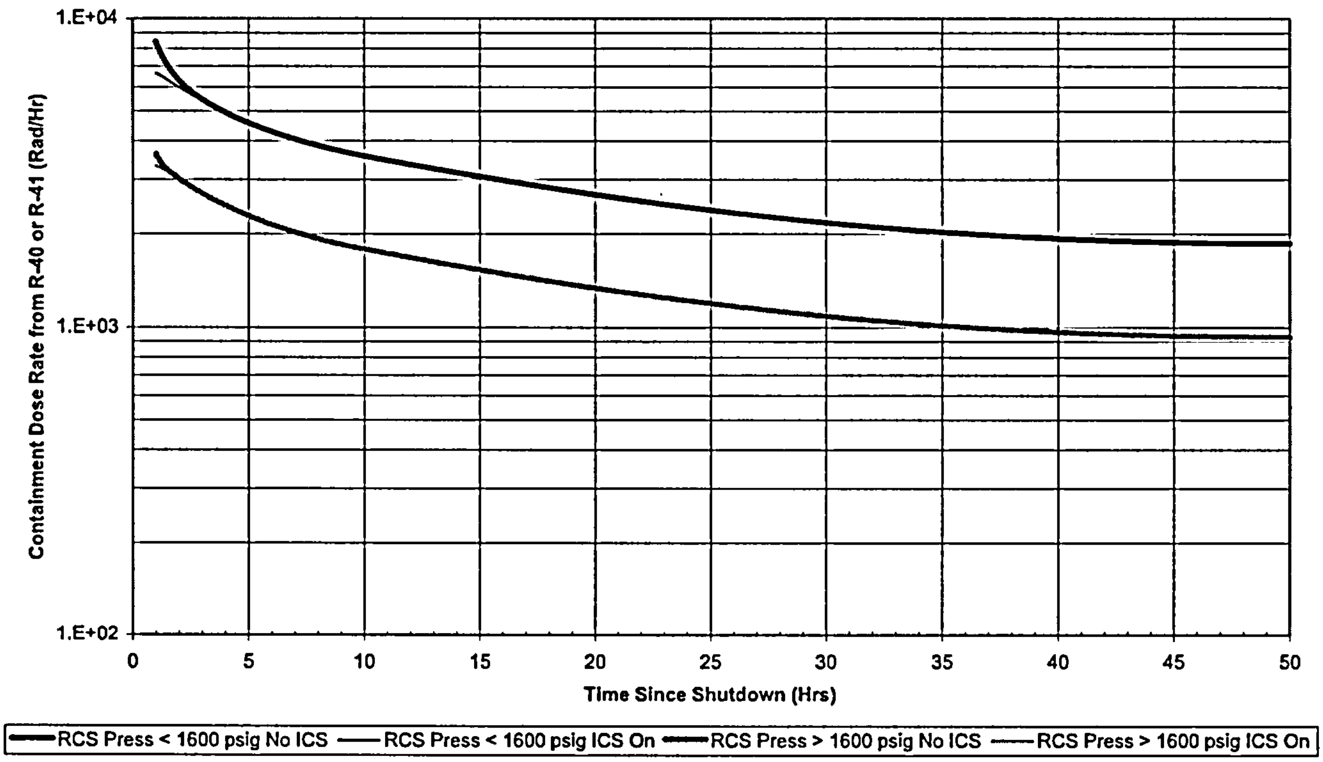
CONTAINMENT RADIATION LEVEL vs. TIME FOR RCS RELEASE



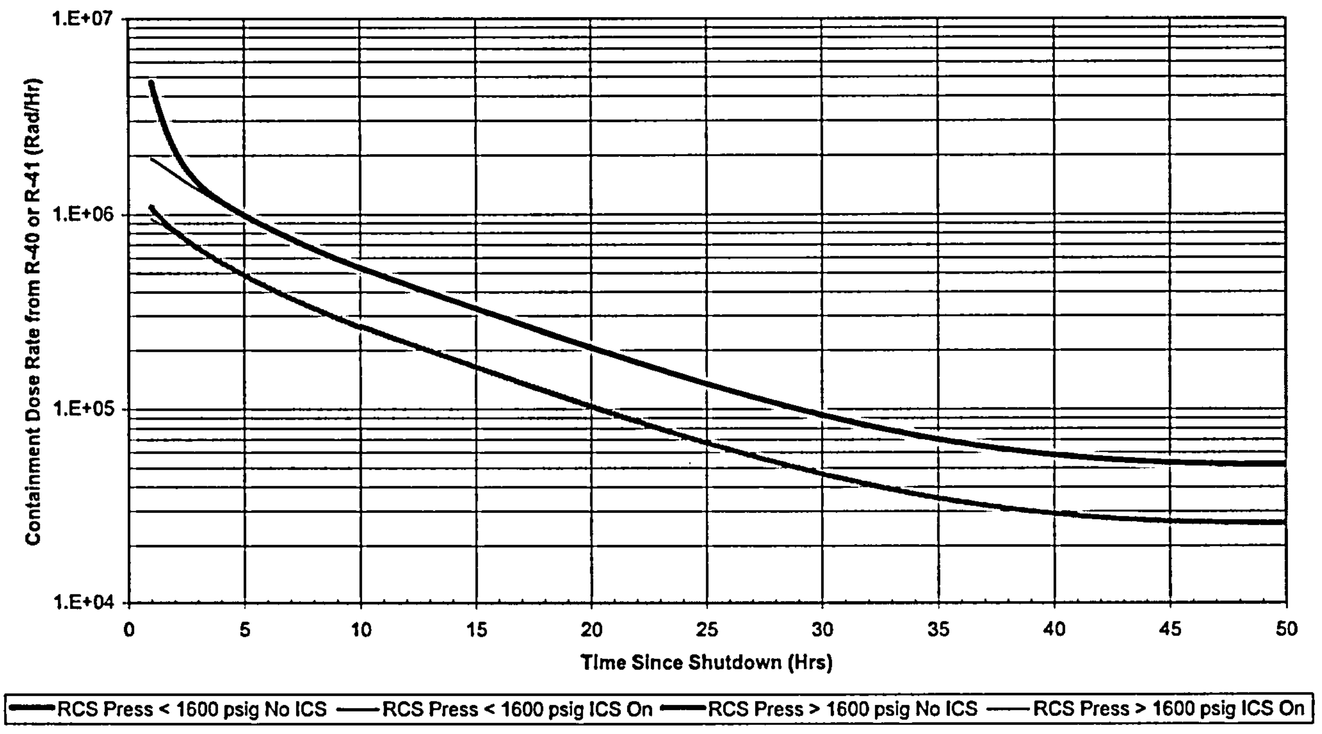
CONTAINMENT RADIATION LEVEL vs. TIME FOR 1% FUEL OVERTEMPERATURE RELEASE



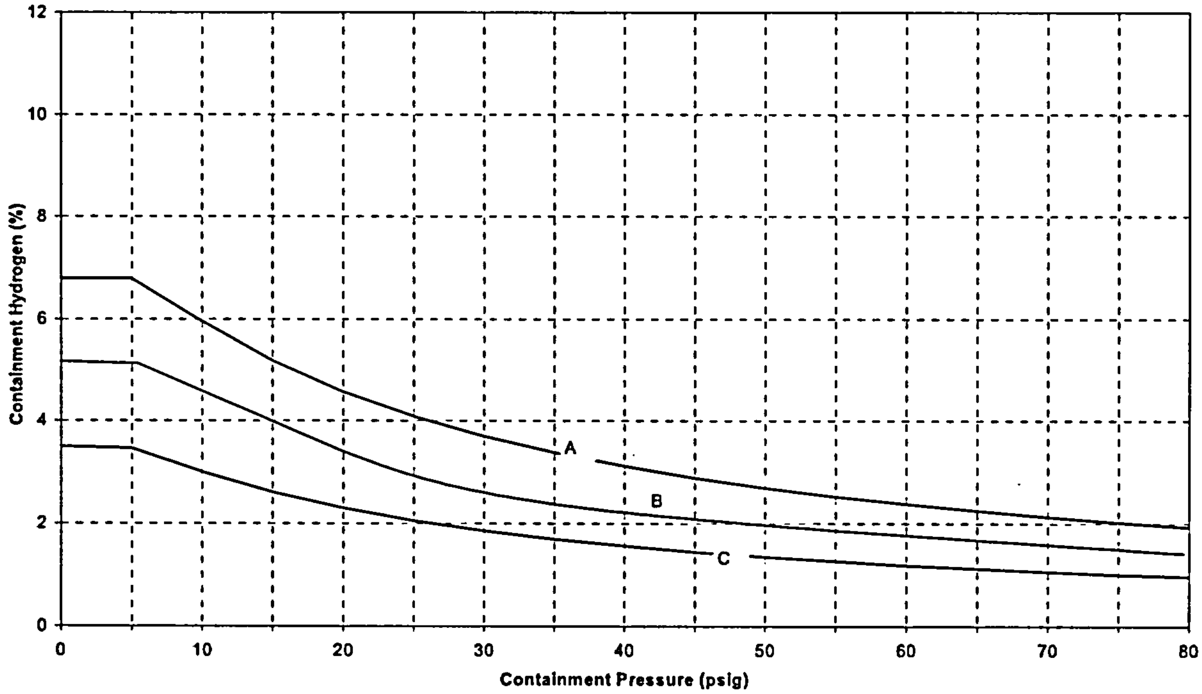
CONTAINMENT RADIATION LEVEL vs. TIME FOR 100% CLAD DAMAGE RELEASE



CONTAINMENT RADIATION LEVEL vs. TIME FOR 100% FUEL OVERTEMPERATURE RELEASE



HYDROGEN CONCENTRATION FOR 100% CORE OVERTEMPERATURE DAMAGE



A RCS Pressure < 1050 psig, core recovered
B RCS Pressure > 1050 psig, core recovered
C Core not recovered

NMC KNPP System Description	System No. 18	Rev. 1
Title Reactor Building Ventilation (RBV) & Post-LOCA Hydrogen Control	Date 10/09/02	Page 1 of 39

Author ______ Bruce R. Heida	Ops Review ______ Jeffrey L. Stoeger
Tech Review ______ Jeffrey L. Stoeger	Approved ______ Manager of Engineering

INDEX

1.0 Summary

1.1 Overview (See Figures KNP-RBV1, RBV2, RBV3, RBV4)

The Reactor Building Ventilation (RBV) System consists of several sub-systems that operate together to cool the air inside Containment during all modes of plant operation.

The RBV System contains the following subsystems:

- Four Containment Fan Coil Units (CFCU), cooled by Service Water (SW),
- Two Control Rod Drive Mechanism (CRDM) fans,
- Two CRDM Shroud cooling coils, -cooled by Service Water (SW),
- Two Containment Dome Area Fans,
- Two Reactor Support Cooling Fans,
- Two Reactor Gap & Neutron Detector Cooling Fans,
- Two Containment Purge and Vent System redundant trains,
- Two Containment vacuum breakers (VB),
- Two Post LOCA Hydrogen Control Subsystem trains.

1.2 System Operation versus Plant Modes

Plant Mode	System Support
Startup	The Reactor Building Ventilation (RBV) System is required to be available and operating. If an SI actuation occurs, the four CFCUs and both CDV fans automatically start.
Normal Power / Shutdown	The RBV System is required to maintain Containment air temperature within the normal operating limits.
Refueling	The RBV purge and vent are required for normal fresh air supply. Containment ventilation radiation monitoring and Containment isolation is required to limit any potential discharges.
Casualty Events	The CFCUs and CDV fans are required to keep Containment atmosphere well mixed and to limit the post-LOCA pressure transient to <46 psig and to reduce Containment pressure to <1 psig within 24 hours after the accident. The post-LOCA H_2 Control System is required to be operable to limit post-accident H_2 concentration below 3.5%, which is the lowest H_2 combustion concentration.
Infrequent Operations	During natural convection cooldown, operating both CRDM fans cools the Reactor vessel head and avoids a 9 hour soak to prevent steam bubble formation in the head.
Maintenance	Maintenance can be performed on the RBV System while operating. The appropriate KNPP Technical Specifications must be reviewed to ensure meeting the limiting conditions for operation.

1.3 **Startup** (OPs # N-RBV-18A, N-RBV-18B)

During normal plant startup, the 36" Containment Vent Isolation Valves RBV-1, RBV-2, RBV-3, and RBV-4 are closed. CFCUs 1A, 1B, 1C and 1D are started. Containment temperature is checked by means of the remote indication in the Control Room on the Omniguard Panel. One CDV fan, one Reactor Gap & Neutron Detector Cooling Fan, one CRDM fan, and one Reactor Support Cooling Fan are manually started from the Control Room. The Operator selects which fan in each pair that is started. The switches for the above fans should be placed in the **AUTO** position which would allow the idle fan to start if the running fan trips off. An annunciator alarms if neither fan in a pair is running.

To start the Containment Purge & Vent Subsystem during maintenance or refueling operations, all isolation valves and dampers in the supply and exhaust ducts are opened. The Containment purge fan and the Containment vent exhaust fan are started in the purge mode. If the Containment sampling and vent stack radiation monitors indicate an acceptable low level of activity, the vent mode of exhaust may be initiated by a selector switch which closes the purge path and stops the purge fan. During the purge mode the vent and supply fan is shutdown. When proceeding to the vent mode, the Containment supply fan unit is started up.

The Post LOCA H_2 Control Subsystem is normally in a non-operating status or isolation-mode status. Surveillance testing is routinely performed to verify system operability.

1.4 **Normal** (OP's # N-RBV-18A, N-RBV-18B, N-RBV-18C)

During normal operation, indicating lights in the Control Room display which active components are running and the position of valves and dampers. Status lights are normally off and all temperatures and pressure points are constantly monitored. When the Containment Purge & Vent Subsystem is operating, the DP across each filter should be checked and not allowed to exceed 3" water gauge.

The Post LOCA H_2 Control Subsystem is not used during normal plant operation. The subsystem is designed for use after a loss of coolant accident to prevent H_2 concentration inside Containment from reaching 3.5%. Air pressurization of Containment is initiated manually by opening the manual Valve IA-1002A/B from the IA System. A portable air compressor can be connected to the emergency connection provided. Operator control of the Post LOCA H_2 Control Subsystem isolation valves is at the Post LOCA H_2 Control Panel (locally). Administrative controls are exercised to assure proper position of the isolation valves.

When venting the Containment atmosphere to the Shield Bldg annulus, the Operator adjusts the air flow rate. A flow rate of 25 cfm should maintain H_2 concentration below 3.5% and the Kates control valve is adjusted based on H_2 concentrations inside Containment. Reduced air flow rates may be desired because of the quantity of radioactive gases that are being released to the Shield Bldg annulus. In this case pressurization can be used in conjunction with the venting operation. An air flow rate limited to 100 cfm should ensure that the annulus is maintained at a negative differential pressure.

1.5 Shutdown

The Containment Purge & Vent Subsystem can be shutdown and all isolation valves closed from the Control Room. The Reactor Gap & Neutron Detector Cooling Fans, Reactor Support Cooling Fans, and CRDM fans may be shutdown from the Control Room by turning both switches in each subsystem to **OFF**. Shutdown of the Post LOCA H_2 Control Subsystem consists of closing all isolation valves at the local control panels.

1.6 Abnormal (OP #A-RBV-18)

If an annunciator associated with the RBV System is actuated, the Control Room Operator observes the indications and symptoms and carries out immediate and subsequent actions in accordance with Procedure A-RBV-18. The abnormal procedure provides action steps to take in the event of any annunciator being actuated (see Section 3.13).

2.0 Functions

The functional requirements of the RBV System include several distinct functions under both normal and emergency operation. These functions are as follows:

- Provides Containment cooling and pressure reduction during post-accident conditions
- Provides adequate heat removal capacity during full power operation
- Relieves the Containment vessel of excessive negative pressure if a rapid temperature drop occurs
- Disperses possible high concentrations of H_2 in the Containment dome following a LOCA
- Removes particulate and gaseous (non-noble gases) radioactive contamination from the Containment air before discharge to atmosphere
- Has Containment isolation valves to assure leak-tight Containment isolation whenever Containment integrity is required
- Provides a supply of fresh tempered air to Containment during refueling operations
- Mitigates or prevents a H_2 buildup above 3.5% in the Containment vessel after a LOCA

The specific purposes of the subsystems of the RBV System include the following:

- The CRDM Subsystem cools the CRDMs. Each fan is capable of providing 100% capacity based on heat generation rate as determined by Westinghouse. During normal operation, one fan is running and one fan is in standby. SW supplied cooling coils are added to increase normal operations Containment cooling.
- The Reactor Support Cooling Subsystem has two fans to maintain proper temperature profiles in the Reactor vessel steel at its supports, shoes, shims, and supporting steel. Each fan has 100% capacity. Heating coils aid in maintaining temperature profiles and ensure temperatures do not drop below 100°F. During normal operation one fan is operating and the second fan is in standby.
- The Reactor Vessel Gap & Neutron Detector Cooling Subsystem limits the maximum temperature of the concrete surrounding the Reactor vessel and cools the neutron detectors. Each fan has 100% capacity. During normal operation, one fan is running and one fan is in standby.
- The Containment Dome Ventilation Subsystem consists of two fans and provides general mixing of Containment air. This mixing aids in the prevention of explosive gas concentration pockets.

- The Purge and Ventilation Subsystem reduces the radioactivity inside the Containment vessel, following full power operation, to the level defined by 10 CFR 20 for a 40 hour occupational work week, within 2 to 6 hours after Reactor shutdown. The subsystem has two modes of operation: exhaust or purge.
- The Vacuum Relief Subsystem consists of two isolation valves and two vacuum breaker valves. This subsystem prevents a Containment vessel internal pressure of <-0.8 psig.
- The Containment Air Cooling Subsystem consists of four CFCUs cooled by SW, operating during normal operation. Following an SI signal, all four CFCUs start automatically and have maximum SW flow to the cooling coils.
- The Post LOCA Hydrogen Control Subsystem prevents H_2 buildup in the Containment vessel after a LOCA. The subsystem-consists of two independent trains with the capability of allowing the installation of two H_2 recombiners if necessary.

3.0 Design Description

3.1 System Arrangement and Flowpath (See Figure KNP-RBV1)

3.1.1 Control Rod Drive Mechanism (CRDM) Cooling Subsystem

The CRDM Cooling Subsystem is located above the CRDM shroud on top of the Reactor vessel. Air is drawn from the refueling pool area, up through the CRDM enclosing shroud and out through three ducts. Air flows through two CRDM shroud cooling coils and the two fans mounted on the missile shield above the CRDMs. Discharge air is directed up at high velocity to mix with Containment air. SW cooling coils were added to reduce the Containment ambient air temperature.

3.1.2 Reactor Support Cooling Subsystem

Air is drawn from the refueling floor and forced through ductwork to each of three pairs of Reactor support plenums. The Reactor support plenums contain heat transfer fins. From the plenums air returns to the 1B RXCP vault and back to mix with the Containment atmosphere. Normally one fan is operating and the second fan is in standby.

3.1.3 Reactor Vessel Gap & Neutron Detector Cooling Subsystem

Cool air is drawn from the ring duct above the refueling floor and forced through ductwork to two branch ducts. One branch duct supplies air to the eight neutron detector wells. The second branch duct supplies air to the bottom of the Reactor vessel and the liner. From these areas air passes to the general Containment space. Normally one fan is operating and the second fan is in standby.

3.1.4 Containment Dome Ventilation (CDV) Subsystem

Two CDV fans, safeguards Train A and Train B, are installed to recirculate and mix hot air and any post-accident H_2 from the top of Containment vessel with the Containment atmosphere. Each CDV fan has separate ductwork which allows the fan to pull air from the Containment dome and discharge the air at the intakes of the CFCUs. The ductwork and fans are designed to withstand seismic forces and the rapid pressure transient caused by a LOCA.

3.1.5 Purge and Ventilation Subsystem

Fresh outside air is drawn in through Intake Damper TAV-10/CD-34040, a filter, steam preheat coil, steam reheat coil, and to the Containment Purge & Vent Supply Fan. The 33,000 cfm air flow enters the Containment through Shield Bldg Damper TAV-12/CD-34036, Outside Containment Isolation Valve RBV-1/CV-31125, and Inside Containment Isolation Valve RBV-2/CV-31126. From the Containment Vent & Purge Supply Unit, air is supplied to the 626' elevation in Containment. Exhaust air from Containment is discharged to the Reactor Bldg Discharge Vent using one of two following modes of operation:

- **High Flow Vent Mode** – This mode provides a high rate of air circulation, 33,000 cfm. Air is exhausted from Containment through Containment Isolation Valves RBV-3/CV-31127 and RBV-4/CV-31128. Air operated Dampers RBV-5/CD-34006 and RBV-6/CD-34009 are open in this mode of operation. The Containment exhaust air then passes through a prefilter, a HEPA filter, and through the Containment Vent Exhaust Fan. Air flow passes air operated Damper RBV-7/CD-34043, past high range Radiation Detector R-37/RE-29075 and low range Radiation Detector R-38/RE-29076, then out the vent stack to atmosphere.
- **Low Flow Purge Mode** – This mode recirculates a small amount (4000 cfm) of Containment air through the Containment Vent Exhaust Filter prefilter, HEPA filter, a charcoal filter, and a purge exhaust fan. The air then passes through the prefilter and HEPA filter and is mixed with a large volume of fresh air drawn through Outside Air Dilution Damper RBV-10/CD-34005. Any contamination that may still be present in the small volume of Containment air is significantly diluted prior to discharging to atmosphere via the Reactor Bldg Discharge Vent. In the Purge Mode, - Damper RBV-6 is closed and the Containment vent & purge supply fan is not operating.

In either mode the same quantity of air is exhausted to atmosphere. In The Purge Mode only a small fraction of the air is from Containment. Mode selection is performed in the Control Room. Vent and purge lines automatically close if a high Containment activity occurs. The exhaust mode is used when radiation level inside Containment is below that defined in 10 CFR 20. The Containment air is continuously monitored by Radiation Monitors R-11, R-12, and R-21.

The Containment purge & vent air supply fan can be used to supply air to the Turbine Bldg by opening Damper TAV-11. Before opening TAV-11, Damper TAV-12 is first closed. TAV-12 is the supply to the Containment purge & vent.

3.1.6 Vacuum Relief Subsystem (See Figure KNP-RBV2)

This subsystem consists of

- Isolation Valves VB-10A/CV-31337 and VB-10B/CV-31338, and Vacuum
- Breaker Valves VB-11A/CV-31339 and VB-11B/CV-31340, in series in each of two large vent lines.

Each vacuum relief line permits air to flow from the Shield Bldg annulus into the Containment vessel. The vent lines enter the Containment through independent and widely separated Containment penetrations.

The subsystem is designed on the basis of inadvertent or simultaneous operation of all Containment air cooling systems during normal operation or following a plant shutdown. After plant shutdown, heat inputs to the Containment are minimal. The cooling water temperatures produce the largest heat removal rates for the respective cooling systems. A vacuum could result in Containment due to the rapid air cooldown.

3.1.7 Containment Air Cooling System

The CFCUs pull air from the immediate vicinity through the SW supplied cooling coils and into the fan intake plenum. The vane axial fan and motor are mounted above the plenum and discharge into the RBV ductwork distribution system.

Immediately downstream of each CFCU, up to the associated emergency discharge damper, the ductwork is designed to withstand seismic forces. Each CFCU has a passive pressure equalizing damper immediately downstream. This damper opens and allows pressure to equalize if the duct external pressure ever exceeds the duct internal pressure. These pressure equalizing dampers provide protection from the rapid LOCA pressure transient causing the downstream ductwork to collapse.

Located within a short distance of the pressure equalizing damper are Emergency Discharge Dampers RBV-150A/CD-34130, RBV-150B/CD-34131, RBV-150C/CD-34132, and RBV-150D/CD-34133. These dampers are normally closed and receive an open signal if Containment pressure exceeds 3.85 psig. These dampers allow for the design basis flow rate through the CFCUs, which provides the required post-accident heat removal. These dampers are air operated, fail open on a loss of air, fail closed on loss of power. These emergency discharge dampers have instrument air accumulators to assure an air supply to keep the dampers closed. When an emergency discharge damper opens, normal flow through the downstream ductwork is severely curtailed. If both dampers of a safeguards pair go open, cooling air to the RXCP motor is lost. Downstream of the emergency discharge dampers ductwork, the remaining ductwork is seismically supported but may collapse during a LOCA pressure transient.

- Downstream of RBV-150A/D: The cool air from CFCU 1A and 1D is supplied to the respective RXCP Pump Vault supply header. These CFCUs can supply only the associated Pump Vault.
- Downstream of RBV-150B/C: The cool air from CFCU 1B and 1C is supplied to either the associated Pump Vault supply header or a header up to the ring duct located above the refueling floor, elevation 649' 6".

These CFCUs are connected at opposite ends of the ring duct to maintain safeguards separation.

The flow path of cool air through CFCU 1B or 1D is controlled by motor operated Dampers RBV-100AB/MD-32348 and RBV-100CD/MD-32349, respectively.

- If CFCU 1A is off or RBV-150A is open, RBV-100AB is closed to direct the flow from CFCU 1B to the associated Pump Vault.
- If CFCU 1D is off or RBV-150D is open, RBV-100CD is closed to direct the air flow from the CFCU 1C to the associated Pump Vault.

Backdraft dampers are installed in the CFCU supply headers upstream of the junction to the Pump Vault header. The backdraft dampers prevent reverse air flow through an idle fan or open emergency discharge damper.

The Pump Vault supply headers direct cooling air to the intakes of the RXCP motors and distribute the remainder to lower elevations in Containment. The ring duct has large registers to blow cool air over the refueling floor. The ring duct also supplies cool air to the intakes of the Reactor Vessel Gap & Neutron Detector Cooling fans.

There are separate SW supply and return headers for each CFCU. Outside Containment one 10" safeguards train supply header divides into two 8" supply headers, - one for each CFCU.

- Manual Isolation Valves SW-900A/B/C/D, located outside the Containment vessel penetrations, allow for isolating each individual CFCU.
- Swing Check Valves SW-901A/B/C/D, located immediately inside the Containment vessel penetrations, provide the second redundant SW supply Containment isolation valve.

Downstream of the CFCUs, the SW return flows in separate headers through the CRDM Shroud Cooling coil control valve stations and out through the Containment vessel penetrations. Outside the Containment vessel penetrations are manual Isolation Valves SW-902A/B/C/D and motor operated CFCU SW Return Isolation Valves SW-903A/B/C/D. These motor operated valves are high capacity two position valves. These valves receive an open signal at SI sequence step 5 and when in the full open position, provide the required post-accident cooling water flow through the respective CFCUs.

CFCU B and D have an air operated thermostatically controlled modulating valve SW-904B or D installed in parallel with the SW Return Isolation Valves SW-903B or D respectively. Valves SW-904B or D are used to control Containment temperature when full CFCU SW flow could overcool Containment. These valves throttle the SW return flow to maintain temperature at the value set on the temperature controller. The temperature controller is located in the Control Room on Mechanical Vertical Panel "A".

Downstream of Valves SW-903A/B/C/D and SW-904B/D, the separate CFCU return headers join into two 10" safeguards headers. These headers discharge into the 24" Aux. Bldg. standpipe. Radiation monitor R-16 monitors the two 10" safeguards return headers for radioactivity. A high radiation signal indicates a leaking CFCU cooling coil or SW header. SW-903A/B/C/ or D is closed in order to clear the alarm and isolate the leaking CFCU.

3.1.8 Post LOCA Hydrogen Control Subsystem (See Figure KNP-RBV4)

The Post LOCA Hydrogen Control Subsystem consists of:

- Two redundant safe-guards Train,
- Sets of instrument air (IA) supply headers,
- Flow regulators,
- Control valves,

- Containment Air Hydrogen Analyzers,
- Piping and tie-in locations for H_2 recombiners.

Each Train can provide a Containment air sample from the CDV fan subsystem exhaust, through two Containment isolation valves to the Containment Air Hydrogen Analyzers. The tubing from the penetration to the analyzers is heat traced to prevent condensation of the sample.

If H_2 is detected in the sample, three distinct methods can be used to control the H_2 concentration below 3.5%. The methods are:
- Dilute the H_2 concentration by pressurizing Containment using compressed air.
- Feed and Bleed using compressed air and a controlled release to the Shield Bldg Annulus at a flow rate of up to 100 cfm.
- Install H_2 recombiners which process the Containment atmosphere. The recombiners remove H_2 and return the subsequent steam mixture back to Containment.

The specific system lineups are discussed below, using Train A valves and equipment (Train B equipment numbers are in parenthesis).

To sample the Containment atmosphere:
- LOCA-2A/MV-32145 (LOCA-2B/MV-32146) and LOCA-10A/CV-31387 (LOCA-10B/CV-31389) are opened at the Local Control Panel to provide the sample to Containment Air Hydrogen Analyzer A or B.
- Valves SA-7003A/MV-32147 (SA-7003B/MV-32148) and LOCA-201A/CV-31726 (LOCA-201B/CV-31727) are opened at the Local Control Panel to return the sample to Containment.

To dilute the Containment H_2 concentration:
- IA is supplied through Valve IA-1001A/CV-31391 or IA-1001B/CV-31392 and manual Isolation Valve IA-1002A or IA-1002B.
- Or, a portable air compressor can be connected via SA-7007A (SA-7007B) and SA-7001A (SA-7001B).
- Containment Isolation Spring Loaded ball Check Valve SA-7003A (SA-7003B) is opened from the Local Control Panel and air flows through the Containment Isolation Spring Loaded Check Valve SA-7004A (SA-7004B).

To "Feed and Bleed" or remove H_2 via a controlled release:
- A H_2 free air source is provided via the dilution pathway.
- The Containment atmosphere is released to the Shield Bldg Annulus via LOCA-2A (LOCA-2B) and LOCA-3A/CV-31386 (LOCA-3B/CV-31388) controlled from the Local Control Panel.

The release flow rate is locally read at FI-18239 (FI-18240) and controlled using LOCA-3A (LOCA-3B). The released gases and particulates are processed by the Shield Bldg Vent System through HEPA and charcoal filters prior to discharge to the environment.

All piping, valves, control stations, and power sources necessary to connect the H_2 recombiners are installed. When needed, one H_2 recombiner can be installed in the Machine Shop and the second in the Aux Bldg Loading Dock. Once connected, flow to the H_2 recombiner package is through LOCA-2A (2B), LOCA-100A/CV-31724 (LOCA-100B/CV-31725) controlled from the Local Control Station and local manual Valves LOCA-101A (101B).

The H_2 recombiners will burn the H_2 using a source of combustion air. The burning takes place in a chamber and forms steam. The subsequent steam/air mixture is returned to Containment via manual Valves LOCA-109A (109B) and Isolation Valves SA-7003A (7003B) and LOCA-201A (201B) controlled from the Local Control Panel.

Due to NRC (Nuclear Regulatory Commission) concerns, KNPP has committed to keeping RBV-1, RBV-2, RBV-3, and RBV-4 closed above hot shutdown. Evidence suggests these 36" diameter butterfly valves may <u>not</u> be able to close when exposed to LOCA forces. Therefore, Train B of the Post-LOCA H_2 Control System has been modified to allow a small continuous purge/vent flow through the system piping.

Fresh air is supplied by a 2" Containment supply blower capable of supplying up to 50 cfm. The fan discharges past a safety/relief valve, LOCA-200, set at 1.9 psig to prevent pressurizing the Containment above the administrative limit of 2.0 psig. Flow is through LOCA-109B, SA-7001B, Check Valve SA-7002B and Containment Isolation Valves SA-7003B/MV-32147 and LOCA-201B/CV-31727.

The air flow from Containment is through Containment Isolation Valves LOCA-2B/MV-32146 and LOCA-100B/CV-31725, and local manual Control Valve LOCA-202 to the Aux. Bldg. exhaust ductwork.

When used during normal operation, Containment Isolation Valves LOCA-2B, LOCA-100B, SA-7003B, and LOCA-201B must be controlled from the Control Room. In addition, all four valves must be open and controlled from the Control Room in order for the supply fan to operate.

3.2 Design Bases

The system design bases include the following:

- The RBV System is capable of cooling the Containment atmosphere following a LOCA. The system ensures that Containment pressure does not exceed the design value of 46 psig at 268°F (100% relative humidity).
- The RBV System is designed to reduce Containment pressure to within 3 psig of atmospheric pressure within the first 24 hours following the LOCA. This design assures all core residual heat is released to the Containment as steam and water.
- During normal full power operation, the RBV System maintains Containment air temperature less than 120°F.
- The Containment ventilation supply unit is capable of providing 1 to 1-1/2 air changes per hour to Containment.

- HEPA filters in both the exhaust and purge subsystem are designed to remove 99.9% of all particulate matter of 0.3 microns and larger.
- The VB Subsystem is designed to ensure the Containment vessel is not subjected to a negative pressure equal to or less than -0.8 psig.
- Charcoal filter in the purge subsystem is designed to remove 99.9% of elemental iodine and 95% of methyl iodide at 70% relative humidity.
- The Post LOCA Hydrogen Control Subsystem is designed to pressurize the Containment vessel over a normal range of 0 to 2 psig (up to a maximum of 21 psig). The higher pressure can be used for dilution by pressurization without venting. The system is sized to provide 100 cfm of air at a supply pressure of 85 to 110 psig. Design pressure and temperature are 150 psig and 300°F.

3.3 Containment Fan Coil Units (CFCUs 155-011, 012, 013, and 014)

Four units are provided: A, B, C, and D.

- Units A and B are located in the Reactor Containment Building on the 626' elevation.
- Units C and D are located in the Reactor Containment Building on the 606' elevation.

Each unit is an enclosed, vane axial fan and cooling coil unit. The units are capable of continuous operation 24 hours a day, 365 days a year, at 120°F. The units are capable of 48 hours operation at post-accident conditions of 270°F, with a pressure rise to 46 psig in 15 seconds. Each fan is driven by 480 VAC, 3 phase, 60 Hz, 125 hp direct-drive motor.

The four Containment Fan Coil Units (CFCU) are installed in pairs on either side of the Reactor vessel.

- CFCU A and B, Safeguards Train A, are located northeast of the Reactor vessel outside of Reactor Coolant Pump (RXCP) A Pump Vault on elevation 626' inside Containment.
- CFCU C and D, Safeguards Train B, are located southwest of the Reactor vessel outside of RXCP 1B Pump Vault on elevation 606' inside Containment.

These four CFCUs provide the majority of air cooling during normal operation. Each CFCU pair is connected to ductwork which distributes the cool air to the RXCP Pump Vaults, the ring duct above the refueling floor, the intake of the Reactor Gap & Neutron Detector Cooling System, and various other floor levels in Containment.

Each CFCU has 12 SW cooling coils located upstream of the fan. Each CFCU has a separate SW supply and return header inside Containment. Therefore, any SW leakage can be isolated using Containment isolation valves without removing more than one CFCU from service.

During RXCP operation, a minimum of two CFCUs, one in each safeguards pair, must be operating. These CFCUs provide cooling air to the RXCP motors via the RXCP Pump Vault ductwork. If cooling air to the RXCP Pump Vault is lost, the RXCP motor overheats. Westinghouse estimates the two RXCP motors overheat within 5-10 minutes after cooling air is lost.

Typically three of the four (3/4) CFCUs are operating to provide proper air distribution. Optimum Containment temperature is maintained with thermostatically controlled modulating valves downstream of CFCU 1B and 1D outside Containment. Valves SW-904B/CV-31120 and SW-904D/CV-31119, throttle the discharge flow from their respective CFCUs to maintain Containment temperature as selected at the Control Room controller located on Mechanical Vertical Panel "A". During the summer or when lake temperature is abnormally high, the fourth CFCU is started.

If a Safety Injection (SI) signal is generated, all four CFCUs start automatically at the respective SI Train sequence step 5. At the same time, motor operated CFCU SW Return Isolation Valves SW-903A/CV-32060, SW-903B/CV-32061, SW-903C/CV-32058, and SW-903D/CV-32059 go full open to provide the maximum SW flow through the respective CFCUs.

If Containment pressure exceeds 3.85 psig, the CFCU emergency discharge dampers go full open. Dampers RBV-150A/CD-34130, RBV-150B/CD-34131, RBV-150C/CD-34132, and RBV-150D/CD-34133 are located immediately downstream of the associated CFCU. These emergency discharge dampers provide a post-LOCA (Loss Of Coolant Accident) or HELB (High Energy Line Break) flow path for cooling air. The dampers are positioned to maximize the mixing of cooled air in the post-accident Containment atmosphere.

In post-accident operation, the four fan coil units have sufficient capacity to maintain post-accident Containment pressure below the design value (46 psig), even in the event of the failure of the Internal Containment Spray (ICS) System.

Containment Fan Coil Units	Post-Accident	Normal
Technical Manual	XK-141422-2 Rev C	
Heat Removal Capability	5.588E +7 Btu/hr	2.21E+7 Btu/hr
Air Flow Rate	44,000 acfm	41,000 scfm
SW Flow Rate	900 gpm	>500 gpm
Air Density (lb/ft^3)	0.17	0.0685
Air Pressure	46 psig	0 psig
Air Temp Entering Coils	270° F	120° F
SW Temp Entering Coils	85° F	70° F
Air Temp Leaving Coils	266° F	---
Design Fouling Factor (Water Side)	0.001	0.001
Radiation Resistance, (RADS)	---	5.0 E+ 7

	Fan Coil 1A	Fan Coil 1B	Fan Coil 1C	Fan Coil 1D
Power Supply	Bus 1-51	Bus 1-61	Bus 1-51	Bus 1-61
Control Room Switch	ES-46539	ES-46540	ES-46541	ES-46542
DSP LOCAL/REMOTE	ES-87168	ES-87170	N/A	N/A
DSP START/STOP	ES-87167	ES-87169	N/A	N/A
Ring Duct Damper	RBV-100AB	RBV-100AB	RBV-100CD	RBV-100CD
Motor Operator	MD-32348	MD-32348	MD-32349	MD-32349
Fan Emerg Damper	RBV-150A	RBV-150B	RBV-150C	RBV-150D
Control Oper.	CD-34130	CD-34131	CD-34132	CD-34133
Emerg Damp Control	ES-40013	ES-40014	ES-40015	ES-40016

Refer to Figure KNP-RBV5 during the following discussion of the CFCUs controls. Each CFCU is controlled by a four position switch, **PULLOUT/STOP/AUTO/START**, which spring returns from **STOP** or **START** to **AUTO**. The **PULLOUT** position is maintained. The CFCU switches are located on Mechanical Vertical Panel "A" in the Control Room.

- Placing a CFCU switch in the **START** position starts that particular fan coil unit. Placing the switch in the **STOP** position stops the CFCU.
- Placing the switch in **AUTO** allows the CFCU to be automatically started by the respective safeguards Train SI sequence signal step 5.

CFCUs A and B respond to sequence signals from Train A and Fans C and D respond to signals from Train B.

Separate temperature detectors are provided on the discharge of each fan. The temperature indications are provided on the Omniguard Panel in the Control Room.

CFCUs A and B can also be operated from the Dedicated Shutdown Panel (DSP). If the Control Room has been evacuated and the Operator is operating from the DSP, local control of CFCUs A and B is taken with the associated **LOCAL/REMOTE** Switches ES-87168 and ES-87170. The CFCUs are then operated using the three position **START/BLANK/STOP** Control Switches ES-87167 and ES-87169. The control switches are maintained in each of the three positions.

Red **(RUNNING)** and green **(STOPPED)** indicating lights are located with the CFCU control switches. If CFCUs A and B are switched to local, the DSP red and green indicating lights are activated and the Control Room lights are deactivated. A white light with the switch indicates a CFCU has tripped when the unit was running. White status lights are also supplied on the SI Ready and SI Active Status Panels, discussed later in Section 3.13.

Cooling water flow through the CFCUs (See Figure KNP-RBV3) is thermostatically controlled by modulating Valves SW-904B/CV-31120 and SW-904D/CV-31119. The modulating valves are designed to maintain optimum cooling during normal operations. Each modulating valve controls cooling water to one CFCU (SW-904B for CFCU B, SW-904D for CFCU D). If more cooling is needed, SW to a third CFCU can be cut in by opening the SW Outlet Valve SW-903A or SW-903C from the Control Room. The fourth CFCU coil cooling water can be placed in service in the same manner.

Following an accident (SI):

- SW flow bypasses Modulating Valves SW-904B and SW-904D through high capacity, two position **(OPEN/CLOSE)**.
- Valves SW-903A, 903B, 903C, 903D. SW Valves SW-903A and SW-903B can also be controlled from the DSP.

3.3.1 CFCU Emergency Discharge Dampers (See Figures KNP-RB01, RBV01, RBV-150A, RBV-150B, RBV-150C, and RBV-150D)

Four dampers are provided, one for each CFCU. The dampers are located a short distance downstream of the respective CFCU in the ductwork which is designed to withstand seismic and LOCA pressure transient forces.

CFCU Emergency Ductwork Dampers	
Manufacturer	Quality Air Design
Technical Manual	XK-73317-1
Design Flowrate	44,000 acfm
Design Temperature/Pressure	293° F/46 psig
Actuator	Bettis Model NCB-520-SR80, Fails open loss of air
Solenoid Valve	ASCO Model NP-8321-A8E Fails closed loss power

Each damper has a three position spring return to center, **CLOSE/AUTO/OPEN**, control switch mounted on Mechanical Vertical Panel "A".

- When in **OPEN**, spring return to **AUTO**, the damper opens and is held open by a damper limit switch contact.
- When in **CLOSE**, spring return to **AUTO**, removes power from the solenoid valve and the damper closes.
- When in **AUTO**, 2 out of 3 Train A (Train B) Containment pressure transmitters exceeding 3.85 psig causes the dampers to open. The dampers stay open until the initiating signal is removed by manually resetting SI at the pushbuttons on Mechanical Control Console "C".

Once the SI is reset, the dampers can be individually closed using the associated control switches. Red (**OPEN**) and green (**CLOSE**) indicating lights are located above the associated control switches. A white status light on the SI Active Status Panel is lit when all four dampers are open.

Two annunciators are used to alert Control Room Operators to abnormal conditions:

- Annunciator 47011-35 actuates if either Train receives an automatic actuation signal. This annunciator warns the Operators that the dampers should be open.
- Annunciator 47007-44 is lit if a damper is open without an automatic signal being received. This annunciator warns the Operator that the damper is open and should be closed. In this condition the CFCU is not capable of supplying cooling air to the RXCP motor.

CFCU Emerg Disch Damper	Solenoid Valve	Control Switch	Solenoid Pwr
RBV-150A/CD-34130	SV-33809	ES-40013	DC5-44
RBV-150B/CD-34131	SV-33810	ES-40014	DC5-45
RBV-150C/CD-34132	SV-33811	ES-40015	DC6-44
RBV-150D/CD-34133	SV-33812	ES-40016	DC6-45

Each damper has a 15 gallon capacity instrument air accumulator upstream of the actuator. This accumulator keeps the damper closed for a minimum of 4 hours if IA is lost to Containment. Keeping the dampers closed allows continued operation of the RXCPs. The control power is DC from the vital buses. This feature permits the damper to be energized to open per KNPP design basis instead of fail open on loss of control power.

3.4 Reactor Support Cooling Subsystem

The purpose of the Reactor Support Cooling Subsystem is to maintain the proper temperature profile in the Reactor vessel steel at the supports, shoes, shims, and supporting steel. This system draws air from the refueling floor and forces the air through ductwork to each of three pairs of Reactor support plenums. The six Reactor support plenums contain heat transfer fins. From the plenums, air returns to the RXCP Vault 1B and back to the refueling floor. One Reactor Support Cooling Subsystem fan is sufficient to provide air movement through the ductwork. Fans 1A and 1B are located at the 649' elevation on the west side of Containment.

Electric heating coils are installed directly in front of the Reactor Support Cooling fans and heat the air to maintain the proper temperature profile. The heating coils are thermostatically controlled so air temperature does not drop below 100°F. Two 100% capacity Reactor Support Cooling fans and coils are installed to provide redundancy.

Fans A and B are provided. The fans are located on the west side of the 649' elevation. Each fan is a horizontal, vane axial, direct drive unit.

Reactor Support Cooling Fans (132-231, 132-232)	
Manufacturer	Joy Mfg. Co.
Technical Manual	XK-141422-4
Air Flow Capacity	9,000 scfm (Each)
Fan Pressure (Total)	10.15" water
Motor Horsepower	30 hp
Power Supply	480 VAC, 3 phase, 60 Hz
Power Source 1A / 1B	MCC 1-32E / 1-42E
Control Switches 1A / 1B	ES-46535 / ES-46536

Each Reactor Support Cooling fan has a motor operated damper (RBV-120A/MD-32344 and RBV-120B/MD-32345) on the discharge side of the fan. (See Figure KNP-RBV6.) A three position maintained, **OFF/AUTO/ON**, control switch is provided for each fan motor. Control Switches ES-46535 and ES-46536 are located in the Control Room on Mechanical Vertical Panel "A".

Electrical interlocks perform the following functions:

- The motor operated dampers on the discharge side of a fan open automatically when the fan motor starter is energized (Fan ON).
- An annunciator is actuated if both fans are OFF, unless electric power to the fans is off.
- Only one fan can operate at a time, with Fan 1A having priority over fan 1B.
- The electric air heaters on the suction side of the fan are turned on and off with the respective fan operation.

The electric air heaters are coils with elements embedded in refractory steel sheaths. The outer surfaces of the sheaths have welded fins which are coated with fired ceramic. The electric heater is enabled by the associated fan. The heater cycles automatically to maintain air temperature above 100°F. The heaters are powered from the same source as the fans. The heater controls are located on the mezzanine level of the Aux. Bldg., just inside the entrance on the wall adjacent to the filter room.

The system draws air from the general Containment space and forces it through ductwork through three parallel balancing dampers. Approximately 3000 cfm is directed to three pairs of Reactor support plenums. The plenums are located below the Reactor vessel and have heat transfer fins to improve heat transfer capability. The 9000 cfm is returned to the RXCP Vault 1B.

3.5 Reactor Gap & Neutron Detector Cooling Subsystem

The purpose of the Reactor Gap & Neutron Detector Cooling Subsystem is to limit the maximum temperature of the concrete surrounding the Reactor vessel and to keep the neutron detectors cool. The Reactor Gap & Neutron Detector Cooling Subsystem draws cool air from the ring duct above the refueling floor to two branch ducts. One branch supplies air to the eight neutron detector wells. The other branch supplies air to the bottom of the Reactor vessel and up through the gap between the vessel and liner. From these areas, air passes to the general Containment space. Two 100% capacity fans are installed. Normally one fan is operating and one is in standby. The two Reactor Gap & Neutron Detector Cooling fans are located at elevation 649' on the west side of Containment.

Fans A and B are provided. The fans are located on the west side of the Containment 649' elevation. Each fan is a horizontal, vane axial, direct drive unit.

Reactor Gap & Neutron Detector Cooling Fans (132-221, 132-222)	
Manufacturer	Joy Mfg. Co.
Technical Manual	XK-141422-4 Rev.0
Air Flow Rate	12,000 scfm
Fan Pressure (Total)	6.18" water
Motor Horsepower	25 hp
Power Supply	480 VAC, 3 phase, 60 Hz,
Power Source 1A/1B	MCC 1-32E/1-42E
Control Switches 1A/1B	ES-46537/ES-46538

Each Reactor Gap & Neutron Detector Cooling fan has a motor operated damper (RBV-110A/MD-32342, RBV-110B/MD-32343) on the discharge side of the fan. The motor operated damper prevents short circuitry of air through the non-operating fan. (See Figure KNP-RBV6.) A three position maintained, **OFF/AUTO/ON**, control switch is provided for each fan motor. Control Switches ES-46537 and ES-46538 are located in the Control Room on Mechanical Vertical Panel "A".

Electrical interlocks perform the following functions:

- The motor operated dampers on the discharge side of a fan open automatically when the fan motor starter is energized (**Fan ON**).
- An annunciator is actuated if both fans are **OFF**, unless electric power to the fans is off.
- Only one fan can operate at a time, with Fan A having dominance over Fan B.

The system draws cool air from the fan coil unit discharge header and forces it through ductwork to two branch ducts. One branch duct supplies air to the bottom of the Reactor vessel and up through the gap between the vessel and the liner. The second duct supplies the eight neutron detector wells. From these areas, air passes to the general Containment space.

3.6 Containment Dome Vent (CDV) Fans (132-061, 132-062)

Containment Dome Vent (CDV) Fans A and B prevent stratification of hot air. The CDV fans pull air and any post-LOCA hydrogen from the Containment vessel dome. Each CDV fan pulls air from the Containment vessel dome through separate supply ductwork and discharges the air at the intakes of the associated CFCU pair. The CDV ductwork and fans are designed to withstand seismic and post-LOCA pressure transients. These CDV fans are safeguards Train A and Train B respectively and receive a start signal at the SI sequence step 5.

Containment Dome Fans A and B are provided. Each fan is horizontal, vane axial, direct drive unit.

Containment Dome Vent Fans (132-061, 132-062)	
Manufacturer	Joy Mfg. Co.
Technical Manual	XK-141422-3 Rev 0
Air Flow Rate	8,000 scfm
Static Pressure	2.0" water
Motor Horsepower	20 hp
Power Supply	480 VAC, 3 phase, 60 Hz,
Power Source A/B	MCC 1-52B/1-62B
Control Switches 1A/1B	ES-46589/ES-46590

(See Figure KNP-RBV6.) A three position **OFF/AUTO/ON** control switch is provided for each fan motor. The control switch spring returns to **AUTO** from **OFF** or **ON** position. Control Switches ES-46589 and ES-46590 are located in the Control Room on Mechanical Vertical Panel "A".

- Placing the switch in the **ON** position starts the CDV fan.
- Placing the switch in the **OFF** position stops the fan.

When the Control Room switch is in **AUTO**, the CDV fan is automatically started by SI signals. Automatic starting takes place on step 5 of the loading sequence. After sequence starting, the fan can be turned **OFF** if the SI signal has been reset.

Red (**ON**) and green (**OFF**) indicating lights are located with the CDV fan control switches. White status lights are provided on the SI Ready Status Panel. Refer to System Description 55, entitled, "Engineered Safety Features (ESF)" for further details.

3.7 **CRDM Cooling Subsystem** (See Figure KNP-RBV3)

The CRDMs are cooled by drawing air from the refueling pool area up through the CRDM enclosing shroud and up through three ducts connected to the CRDM shroud plenum mounted on top of the missile shield above the CRDMs. The air leaving the ducts is pulled through two service water (SW) CRDM Shroud cooling coils to the fan inlets. The normal heat generation rate requires one fan to be operating and the second fan to be in standby. Discharge air from the CRDM fans is directed up at a high velocity so as to mix with the Containment air. If required to expedite a natural convection cooldown, both CRDM fans can be run at the same time.

Two CRDM Shroud Cooling coils, supplied with SW, were installed to increase normal operations air cooling capacity. The air temperature at the coil intake surface is nominally 237°F.

The CRDM Shroud Cooling coils are supplied with SW from the discharge of the CFCU cooling coils. Two coils are installed to maintain safeguards Train separation. Each cooling coil can be supplied from either CFCU of the safeguards pair.

- SW Diversion Valves SW-901A-1/CV-31704, SW-901B-1/ CV-31705, SW-901C-1/CV-31706, and SW-901D-1/CV-31707, go partially closed to divert approximately 75 gpm of CFCU flow through the CRDM shroud cooling coils.
- CRDM Shroud Cooling Coil Isolation Valves SW-910/A/B/C/D, SW-913A/B/C/D, and SW-914A/B/C/D are open to permit flow through the CRDM Shroud Cooling coils.

Upon receipt of an SI signal sequence step 5, the CRDM Shroud Cooling coils are isolated.

- The SW diversion valves go full open to provide unrestricted flow through the CFCUs,
- The CRDM Shroud Cooling coils supply and return isolation valves go full closed.

CRDM Cooling Fans 1A and 1B are provided. The fan intakes pull air through the CRDM cooling shroud. Each fan is a vertical, heavy duty centrifugal fan.

CRDM Cooling Fans (132-091, 132-092)	
Manufacturer	Westinghouse Electric
Technical Manual	XK-478-1
Purchase Order	XK-54106
Air Flow Rate	18,000 scfm
Fan Pressure (Total)	3.0" water
Motor Horsepower	75 hp
Power Supply	480 VAC 3 Phase, 60 Hz
Power Source 1A / 1B	MCC 1-32E / 1-42E

Control Switches 1A / 1B	ES-46823 / ES-46824
Shroud Cooling Coils	Two
SW Flow to each coil	75 gpm
Cooling Coil Heat Removal	2.13E + 6 Btu/hr

The CRDM shroud cooling coils (one in train A and one in train B) are supplied with SW from the discharge side of the CFCUs. (See Figure KNP-RBV3.) Each CRDM shroud cooling coil:

- Is supplied from either CFCU of its respective train.
- Gets approximately 75 gpm SW flow.

The SW supply and isolation valves for the CRDM shroud cooling coils operate as follows:

- To supply CRDM Shroud Cooling Coil 1A/1B (1C/1D) from CFCU 1A (1C), the diversion valves SW-901A-1 (SW-901C-1) and SW-901B-1 (SW-901D-1) go partially closed.
- This diverts 75 gpm SW flow through the CRDM shroud cooling coil 1A/1B (1C/1D).
- The CRDM Supply Isolation Valves SW-910A (SW-910C) and SW-911AB (SW-911CD) and the CRDM Return Isolation Valves SW-914A (SW-914C) go fully open to establish the flow path. The SW Supply Isolation Valves SW-911B (SW-911D) and the Return Isolation Valve 914B (SW-914D) remain closed, since the associated CFCU is not supplying CRDM shroud cooling coil flow. Diversion Valves SW-901B-1 and SW-901D-1 are partially closed, even though the valves are not diverting flow through the CRDM shroud cooling coils, to balance flow through the CFCUs since they have a common header outside of Containment. To supply CRDM Shroud Cooling Coil 1A/1B (1C/1D) from CFCU 1B (1D), the following valves change position.
  - SW-910B (SW-910D) and SW-914B (SW-914D) go fully open,
  - SW-91OA (SW-910C) and SW-914A (SW-914C) go fully closed.

All other valves remain in the same position.

The CRDM Shroud Cooling Coils are <u>not</u> intended for use as post-accident cooling. Upon receipt of the safeguards CFCU start signal, SI sequence step 5:

- SW Diversion Valves SW-901A-1 through D-1 go fully open to provide unrestricted SW flow through the CFCUs.
- The CRDM Shroud Cooling Coils Supply Isolation Valves SW-910A through SW-910D, and SW-911AB and SW-911CD, and the CRDM Shroud Cooling Coils Return Isolation Valves SW-914A through D go closed to isolate CRDM shroud cooling coils SW.

All CRDM shroud cooling coil SW valves are air operated and fail to the safe post-accident position on loss of electrical power or loss of instrument air.

Each CRDM Cooling fan has a motor operated damper (RBV-111A/MD-32346, RBV-111B/MD-32347) on the discharge side of the fan. (See Figure KNP-RBV6.) A three position maintained **OFF/AUTO/ON** control switch is provided for each fan motor. Control Switches ES-46823 and ES-46824 are located in the Control Room on Mechanical Vertical Panel "A". Red (**ON**) and green (**OFF**) indicating lights are located with the fan control switches.

- Normally, one fan is **ON** and the second fan is in standby.
- If in **AUTO**, the standby fan starts when the running fan trips.

Electrical interlocks perform the following functions:

- The motor operated dampers on the discharge side of a fan open automatically when the fan motor starter is energized/fan **ON**.
- An annunciator is actuated if both fans are **OFF**, unless electric power to the fans is off.
- Normally only one fan operates at a time with the idle fan in stand-by.

Both fans can be run at the same time to maximize the heat removal rate from the Reactor vessel head during a natural convection cooldown. Running both fans avoids a soak time of 9 hours to prevent steam bubble formation in the Reactor vessel head area.

3.8 Containment Vacuum Relief Subsystem (See Figure KNP-RBV9)

The Containment is protected from damage due to negative pressure by the Vacuum Breaker (VB) Subsystem. The VB Subsystem consists of an isolation valve and a vacuum breaker in series in each of two large vent lines. These large vent lines permit air flow for equalization between the Shield Bldg. and Containment. The vent lines enter Containment through independent and widely separated Containment penetrations. The design basis for sizing the VB Subsystem is based on the inadvertent or simultaneous operation of all Containment air cooling systems during normal operation or following a plant shutdown, when heat inputs to the Containment are minimal and the cooling water temperatures produce the largest heat removal rates for the respective cooling systems.

Automatic vacuum relief is provided for the Reactor Containment Bldg. This subsystem contains two vacuum relief lines between the Containment and the Shield Bldg annulus. Each vacuum relief line contains a power operated butterfly valve (VB-10A/CV-31337), and a self actuated swing check valve (VB-11A/CV-31339). Each power operated relief valve is individually controlled by a differential pressure (DP) switch. Each valve is also controlled by a Containment isolation signal and a Control Room switch. Self actuated Swing Check Valves VB-11A or VB-11B, have a test feature which allows testing while the Reactor is at power. The two power operated isolation valves (VB-10A and VB-10B) and swing check relief valves VB-11A and VB-11B are located in the Shield Bldg. Annulus near the air locks.

Containment Vacuum Breaker Isolation Valves VB-10A/CV-31337 and VB-10B/CV-31338 are 18" wafer butterfly style with air to close, spring to open operator. An accumulator is provided with each valve. The valve closes in three seconds or less. The valve disk seats on EPT base material which maintains the valve leak-proof at maximum Containment pressure and temperature. The valve is capable of performing its design function after a total gamma radiation dose of 5.0E+7 rads.

Vacuum Breaker Valves VB-11A/CV-31339 and VB-11B/CV-31340 are swing disk type check valves with 21" diameter seats. A unique magnetic latch assures both tight closure up to the design opening DP and full open travel of the valve upon slightly higher DPs. The valve disk seats on EPT base material. An air cylinder exerciser is provided for periodic cycling of the valve. A bracket and weights are provided for confirming cracking pressure and full open DP. Limit switches indicate valve position indication.

Power operated Containment Vacuum Breaker Isolation Valves VB-10A/CV-31337 and VB-10B/CV-31338 have air diaphragm operators which are controlled by independent three way Solenoid Pilot Valves SV-33291 and SV-33292. The VB-10A and VB-10B valves fail closed on a loss of air or solenoid power.

Each solenoid pilot valve air supply is equipped with a reservoir, capable of at least one operation of the VB-10A (or VB-10B) if the normal air supply fails. Each reservoir has a pressure switch to provide a signal to the sequence of events recorder (SER) and to actuate an alarm in the Control Room when air pressure drops below 90 psig. If either valve VB-10A or VB-10B is <u>not</u> fully closed, an alarm is actuated in the Control Room.

Vacuum Relief Components		
	VB-10A	**VB-10B**
Manufacturer	Henry Pratt Company	
Technical Manual	XK-158-8, -9	
Air Operator	CV-31337	CV-31338
Control switch	ES-46828	ES-46829
Solenoid Pilot	SV-33291	SV-33292
Containment Isolation	Train A	Train B
Reservoir press switch	DPS-16427	DPS-16428
	VB-11A	**VB-11B**
Manufacturer	GPE Division of Singer	
Technical Manual	XK-414-11	
Swing check air Operator	CV-31339	CV-31340
Swing Check Solenoid	SV-33355	SV-33356
Swing Check Valve Test PB	PB-19585	PB-19586

The control logic input signals for the Containment relief isolation valves are identical and receive inputs from the following.

- Three position **CLOSE/AUTO/OPEN** Control Switches ES-46828, ES-46829, with a spring return from **CLOSE** or **OPEN** to **AUTO**. The control switches are located on Mechanical Vertical Panel "A".
- Containment Isolation Signal Train A, Train B.
- The Reactor Containment Bldg. to Shield Bldg. DP switch measures the difference between the two buildings. The DP switch has a range of 0 to 1.0 psid. A reading on scale indicates that the atmosphere within Containment is at a vacuum compared to the air pressure in the Shield Bldg. Annulus.

A reading of 0.3 psid or greater (Annulus pressure > Containment) actuates a trip signal from the associated DP switch to de-energize the solenoid pilot valve for Valves VB-10A or VB-10B. This action vents the air from the diaphragm operator causing it to open. The 0.3 psid signal is dominant and prevents a Containment Isolation signal or the control switch from closing the VB-10A or VB-10B. When the valve opens, air flows from the

Annulus into Containment causing the DP to decrease.

With a reading of 0.3 to 0.2 psid, Valve VB-10A or VB-10B can be closed by turning the control switch to the **CLOSE** position. Once closed and when the control switch spring returns to **AUTO**, the valve stays closed by a seal-in signal from the valve limit switch. A Containment isolation signal automatically closes the valve. When the control switch is in the **AUTO** position, the valve is open, and the DP switch reading is below 0.2 psid.

Valve VB-10A or VB-10B may be opened any time by holding the control switch in the **OPEN** position. If Containment pressure is greater than that in the Shield Bldg annulus, air flow from Containment is prevented by vacuum breaker Swing Check Valve VB-11A or VB-11B.

Self Operated Vacuum Relief Valves VB-11A and VB-11B are provided, each with an air cylinder exerciser, a solenoid valve, and a test pushbutton.

- When the local test pushbutton is pressed, the self operated relief valve opens.
- When the test pushbutton is released, the relief valve closes.

Red (**OPEN**) and green (**CLOSED**) indicating lights are provided for the self operated vacuum relief valves VB-11A and VB-11B. Red (**OPEN**) and green (**CLOSED**) indicating lights for VB-10A and VB-10B are located directly above the control switches. White indicating lights for VB-10A and for VB-10B are on the Containment Isolation Active Status Panel and indicate the Containment isolation status.

3.9 Containment Purge & Vent Subsystem

The Purge and Vent Subsystem consists of a fresh air supply and exhaust and filtration. Isolation valves are installed both inside and outside the Containment where ducts from this system penetrate Containment. Air can be exhausted from the Containment by either of two modes.

- **Exhaust Mode** - The first mode provides a high rate of air circulation through HEPA (high efficiency particulate activity) filters and exhausts out the Reactor Bldg Discharge Vent to the atmosphere. This mode is used when the radiation level in Containment is below the limits defined in 10CFR20. The Containment air radioactive level is constantly monitored by means of Radiation Monitors R-11, R-12, and R-21. High radiation, as detected by one of these monitors, generates a Containment Vent Isolation signal.
- **Purge Mode** - The second mode of operating the Purge and Vent Subsystem is through a purge fan and additional HEPA filter and charcoal filter. A small amount of Containment air is exhausted via the prefilter, HEPA filter, charcoal filter, small capacity fan, through the second filter set, and then to the suction of the ventilation exhaust fan where it is mixed with fresh outside air. Any contamination that gets through the filters is significantly diluted before finally being discharged to atmosphere via the Reactor Bldg Discharge Vent. In either mode the same quantity of air is exhausted to the atmosphere. In the purge mode, only a fraction of the air is taken from Containment.

Both the exhaust and purge mode are controlled remotely from the Control Room on Mechanical Vertical Panel "A". If Containment air activity increases above the setpoint of the Containment Atmosphere Radiation Monitors R-11, R-12, and/or the Containment Vent Radiation Monitor R-21, a Containment Vent Isolation signal closes Containment Isolation Valves RBV-1/CV31125, RBV-2/CV31126, RBV-3/CV31127, and RBV-4/CV31127.

NOTE: When the isolation valves close, the exhaust fans stop automatically.

3.9.1 Purge & Vent Supply Unit

The Purge & Vent Supply Unit is located in the Aux. Bldg. on the 657' elevation next to the Turb Bldg Supply Fan Room. The supply unit has a conventional sheetmetal housing with an external motor and V-belt drive to the fan. The enclosed construction contains:

- Control Dampers TAV-10/CD-34030 and TAV-11/CD-34034,
- A fresh air filter,
- A steam preheat coil, and
- A steam reheat coil.

During the high flow Vent Mode, the inlet fan supplies 33,000 cfm of fresh air to Containment. The Containment vent exhaust fan removes an equal quantity of air and exhausts the air through filters to the Reactor Bldg Discharge Vent. After the Containment vent exhaust fan is started, the purge & vent supply fan is started. An interlock requires the purge & vent supply fan or the purge exhaust fan to be started within 45 seconds or the Containment vent exhaust fan stops.

The steam preheat coil and the steam reheat coil are supplied from Heating Steam to maintain the supply air at 70°F. The purge & vent supply fan has a V-belt drive.

Purge & Vent Supply Unit	
Manufacturer	Westinghouse Electric
Technical Manual	XK-397-11
Fan capacity	33,000 scfm
Static pressure (inches of water)	3.5"
Motor Horsepower	25 / MCC 1-45E
Power Supply	480 VAC, 3 phase, 60 Hz
Air Temp entering (minimum)	-20° F
Air Temp leaving	70° F
Fan Control Switch	ES-46533
Preheat Coil Air Operator	CV-31057
Reheat Coil Air Operator	CV-31058
Supply Unit Inlet Damper	TAV-10/CD-34030
Turb. Bldg. Makeup Damper	TAV-11/CD-34034
Supply Unit Exhaust Damper	TAV-12/CD-34033

The Purge & Vent Supply Fan can be started to supply Containment for high flow vent or to supply the Turbine Bldg. with preheated air during the winter. Interlocks to start the supply fan are either the Turbine Bldg. Supply Damper TAV-11 is open, or the Containment isolation valves are all open and the Containment vent exhaust fan is running. The fan has a three position **OFF/AUTO/ON** Control Switch ES-46533 with a spring return to **AUTO** feature. The purge & vent supply fan trips on any of the following conditions:

- Low inlet (freezestat) temperature, below 40°F
- Condensate temperature from either of the two preheat coils is less than 100°F
- DP across the inlet filter ≥1.5 inches of water on DPS-16403
- Containment isolation signal closes all Containment valves
- Loss of electrical power or loss of Instrument Air (IA)
- Containment vent exhaust fan stops with all Containment dampers open
- Containment purge exhaust fan starts
- A high radiation level causes a Containment isolation

3.9.2 Containment Vent Exhaust Fan

The Containment vent exhaust fan is a vane-axial type, direct-connected to a motor in the air stream. The fan is of the all welded steel construction. The fan delivers 33,000 scfm and is driven by a 100 hp motor. The motor is powered from MCC 1-35E.

Containment Vent Exhaust Fan (132-191)	
Manufacturer	Joy Manufacturing Co.
Technical Manual	XK-141422-4 Rev 0
Fan capacity	33,000 scfm
Fan Pressure (Total)	10.7" Wg
Motor HP/ Power MCC	100 / MCC 1-35E
Power Supply	480 VAC, 3 phase, 60 Hz
Fan Control Switch	ES-46831
CNTMT Vent Exh Fan Damper	RBV-7/CD-34043
CNTMT Purge Exh Filter Bypass Damper	RBV-6/CD-34009
CNTMT Exhaust Damper	RBV-5/CD-34006

The Containment Vent Exhaust Fan is provided with Control Switch ES-46831 on Mechanical Vertical Panel "A". This switch is a three position **OFF/AUTO/ON** switch with a spring return from **ON** to **AUTO**. The switch is maintained in the **OFF** and **AUTO** positions.

When the control switch is turned to **ON**, Vent Exhaust Fan Damper RBV-7 is opened by deenergizing Solenoid Valves SV-33301-01 and SV-33301-02. Simultaneously, - RBV-6 is opened by de-energizing the associated Solenoid Valve SV-33286-01 and SV-33286-02.

- When both dampers reach the full open position, the Containment vent exhaust fan starts.
- When the control switch is released and spring returns to **AUTO**, and all Containment ventilation isolation valves are in the fully open position, the fan continues to run for 45 seconds.

The vent exhaust fan continues to run if the Containment purge & vent supply fan is started, or if the purge exhaust fan is started. Red **(RUNNING)** and green **(STOPPED)** indicating lights are associated with the vent exhaust fan control switch.

3.9.3 Containment Purge Exhaust Fan

The Containment Purge Exhaust Fan is a vane-axial type, direct-connected to a motor in the air stream. The fan is of the all welded steel construction. The fan delivers 4,000 scfm.

Containment Purge Exhaust Fan (132-161)	
Manufacturer	Joy Manufacturing Co.
Technical Manual	XK-141422-4 Rev 0
Air Flow Rate	4,000 scfm
Fan Pressure (Total)	6.8" Wg
Motor Horsepower	25HP
Power Supply	480 VAC, 3 phase, 60 Hz
MCC	MCC 1-35E
Fan Control Switch	ES-46534
CNTMT Vent Outside Air Damper	RBV-10/CD-34005
CNTMT Purge Exh Filter Assy Supply Damper	RBV-20/CD-34008
CNTMT Purge Exhaust Fan Damper	RBV-21/CD-34042

The Containment purge exhaust fan is provided with Control Switch ES-46534 on Mechanical Vertical Panel "A". This control switch is a three position **OFF/AUTO/ON** switch with a spring return from **ON** to **AUTO**. The switch is maintained in the **OFF** and **AUTO** positions.

When the control switch is turned to **ON**, Purge Exhaust Fan Damper RBV-21 is opened by deenergizing associated Solenoid Valve SV-33300.

- Simultaneously RBV-20 is opened by deenergizing associated Solenoid Valve SV-33285.
- Containment Vent Exhaust Outside Air Dilution Damper RBV-10 is also opened.
- Containment Purge Exhaust Filter Assembly Bypass Damper RBV-6 is closed.

When all dampers reach the fully open position and RBV-6 is closed, the Containment purge exhaust fan starts.

When the control switch is released and spring returns to **AUTO**, and all Containment ventilation isolation valves are in the fully open position, the Containment purge exhaust fan continues to run if the following conditions exist:

- Containment vent exhaust fan is running
- Inlet air freezstat indicates that the inlet air temperature is above 40°F
- DP across the inlet damper TAV-10 is low
- Condensate leaving the preheat coil is above 100°F

An auxiliary contact on the Purge Exhaust Fan provides a signal bypass to the 45 second timer on the Containment vent exhaust fan.

Starting the Containment Purge Exhaust Fan also provides a signal to open Containment Purge & Vent Supply Unit Inlet Damper TAV-10 and activates the control system for the flow of steam to the preheat and reheat coils.

The Containment purge exhaust filter assembly is provided with a water deluge system. When high temperature is detected in the filter assembly by heat detectors, Deluge Valve SV-33361 sprays water on the filter assembly and the Containment purge fan is tripped. Momentarily placing Containment Purge Exhaust Fan Control Switch ES-46534 to the **OFF** position and removing the heat detector signal (heat detectors have a fusible link that must be replaced to remove signal) resets the retentive memory. Resetting the memory allows normal operation of the Containment Purge Exhaust Fan. The removing of the heat detector signal also stops the deluge flow of water.

The Containment Purge Exhaust Fan is stopped by any one of the following conditions:

- High DP, 0.3 psid, across the Containment purge & vent supply inlet damper
- Freezstat temperature of 40°F
- Turning the control switch to the **OFF** position
- Operation of the exhaust filter deluge heat detectors
- Closure of any Containment isolation valve or damper
- Stopping the Containment ventilation exhaust fan
- Loss of power or loss of instrument air
- Condensate temperature from either of the two preheat or reheat coils below 100°F.
- A Containment isolation or high radiation signal

3.9.4 Containment Vent Exhaust Fan Filter Assembly

This filter assembly is located on the Aux. Bldg. fan floor. The assembly consists of a prefilter with an 25% (ASHRAE Std. 52.1-1992) efficiency rating and a HEPA filter with a 99.97% efficiency rating for removing particles of 0.3 microns or larger. Local instrumentation includes a prefilter DP indicator and a HEPA filter DP indicator.

3.9.5 Containment Purge Exhaust Filter Assembly

This filter assembly is located on the Aux. Bldg. fan floor. The assembly consists of a prefilter with 25% (ASHRAE Std. 52.1-1992) efficiency rating and a HEPA filter with a 99.97% efficiency rating for removing particles of 0.3 microns or larger. A charcoal bank removes 99.9% elemental iodine and 95% methyl iodide at a 70% relative humidity. The fire protection deluge valve opens at 200° F air temperature. Local instrumentation includes a prefilter DP indicator, a HEPA filter DP indicator, and a charcoal bank DP indicator.

3.9.6 Containment Vent Isolation Valves (RBV-1, 2, 3 and 4)

Four valves are installed and are of QA Type 1 construction for reliable operation. Two valves are in the Containment air supply and two valves are in the exhaust air ducts. A valve is installed in the duct on either side of Containment. These valves are designed to withstand the projected seismic accelerations due to a design basis earthquake.

The valves are of the butterfly type, 36" diameter, and are designed and installed to prevent leakage from bypassing the valve seat. A special offset seat design places the valve shaft completely upstream or downstream of the valve seat, depending on valve orientation.

Each valve operator is an air cylinder and piston with a spring return. Air pressure is required to keep the valve open. On loss of air pressure the spring closes the valve is 2 seconds or less. For manual closing of these valves during routine operation, air is vented from the cylinder through a restrictor to cause slow closing of the valve.

The valve disk seats on a seal ring made of EPT base material which maintains the valve leak-proof at maximum Containment pressure and temperature. The valve is designed to function following a total gamma radiation exposure of 5.0E+7 rads.

Containment Isolation Valves				
Manufacturer	Henry Pratt Co.			
Technical Manual	XK-158-1			
Model				
Size	36" dia.			
	RBV-1	RBV-2	RBV-3	RBV-4
Control Valve Oper.	CV-31125	CV-31126	CV-31124	CV-31123
Solenoid Valve Inlet	SV-33127	SV-33130	SV-33124	SV-33121
(restricted 1A supply or vent to atmosphere)				
Solenoid Valve Vent	SV-33128	SV-33131	SV-33125	SV-33122
Solenoid Valve Vent	SV-33129	SV-33132	SV-33126	SV-33123
Control Switch (CLOSE/AUTO/OPEN)	ES-46598	ES-46596	ES-46595	ES-46597

- A Containment ventilation isolation signal from Train A automatically closes Outside Containment Isolation Valves RBV-1 and RBV-4.
- A signal from Train B automatically closes the Inside Containment Isolation Valves RBV-2 and RBV-3.

As a prerequisite for isolation valve operation, these signals must not be present.

When no Containment ventilation isolation signals are present and the associated Control Room switch is placed in the **OPEN** position, the inlet solenoid valve is positioned to allow IA to flow to the cylinder operator of the control valve. At the same time, the high capacity vent solenoid valves are energized to prevent venting the operator to atmosphere. The restricting solenoid valve allows air to be slowly applied to the Control Operator which slowly opens the associated Containment isolation valve.

When the control valve reaches the fully open position and the Operator releases the control switch, the control switch spring returns to **AUTO**. The valve remains in the fully open position. The Containment isolation valve closes due to any of the following conditions:
- Control Room switch placed in the **CLOSE** position.
- Containment ventilation isolation signal Train A or Train B
- Loss of electrical power or loss of instrument air (IA)

If Containment Isolation Valves RBV-1, 2, 3, or 4 is closed using the Control Room switch, venting occurs only via the restricted solenoid valve and valve closure is slow. The valve closes rapidly 2 seconds if the closing is a result of deenergizing the high capacity solenoid vent valves.

Red (**OPEN**) and green (**CLOSE**) indicating lights are provided with the Control Room switches. White status lights are provided on the Containment Isolation Active Status Panel.

3.10 Post LOCA Hydrogen Control Subsystem

The Post LOCA Hydrogen Control Subsystem controls the hydrogen (H_2) concentration in the post-accident Containment atmosphere. Venting and replacement of the Containment atmosphere, dilution by pressurization, or a combination of both methods can be used. H_2 free air can be introduced into and slightly pressurizes the Containment. The mixed air is discharged into the slightly negative Shield Bldg annulus at a rate up to 100 cfm. The necessary piping, control valves, and power supplies for the H_2 recombiners has been installed. If required, the H_2 recombiners are brought on site.

The Post LOCA Hydrogen Control Subsystem is connected to Instrument Air (IA) to assure an oil-free air supply. In an emergency, the supply line can be connected to a portable oil-free air compressor. The system is normally isolated. Following a LOCA, the system is started manually by Operations personnel, as required.

3.10.1 Air supply (See Figure KNP-RBV4)

The 2" supply lines are redundant. Figure KNP-RBV4 displays the Train B side. The IA supply connects to both trains. A portable air compressor can be connected to either train at the normally closed Valve SA-7000A/B.

- A Kates constant air flow regulator (IA-1001A/CV-31391 or IA-1001B/CV-31392) controls the air flow up to 100 cfm when IA is used. The air flow rate is manually adjustable at the valve (nominally 25 cfm). The IA is normally isolated via Valves IA-1002A and IA-1002B.
- Isolation Check Valve SA-7004A/B (inside Containment) is a ball check valve with a 5 psig spring.
- Isolation Valves SA-7003A/ MV-32147 or SA-7003B/MV-32148 (located outside Containment in the Shield Bldg. Annulus) are Limitorque motor operated valve (MOV).
- Solenoid Valve SA-7010A/B vents to the annulus and prevents any through-line leakage to the Aux. Bldg.

Both the MOVs and solenoid operated valves are controlled with local Switches ES-19578 and ES-19582.

3.10.2 Air Venting

The 2" vent lines draw air from the dome fan header discharge and vent into the Annulus.

- Isolation Valve LOCA-2A/MV-32145 or LOCA-2B/MV-32146 (inside Containment) are Limitorque motor operated 2" ball valves.
- Inside Containment Isolation Valves LOCA-2A or LOCA-2B are operated by Local Control Switch ES-19571 or ES-19581.
- Inside the Shield Bldg. Annulus a 1" air modulated ball valve, LOCA-3A/CV-31386 or LOCA-3B/CV-31388, is used to regulate discharge air flow to the annulus.
- A 1" air controlled valve, LOCA-10A/CV-31387 or LOCA-10B/CV-31389, is also located in the Annulus.
- LOCA-10A and LOCA-10B) are isolation valves in the gas sampling line to the Containment Air H_2 Analyzer A or B. LOCA-10A and LOCA-10B are operated from a local panel with Control Switches CS-19580 and CS-19584 respectively.
- Normally open ball Valves LOCA-1A and LOCA-1B is installed inside Containment in the line to isolate the MVs during leak testing. A 3/8" loop seal and test connection is located between the manual valve and the MV. During normal plant operation, the loop seal prevents any collection of water in the vent line.

3.10.3 Containment Supply Blower

A Containment supply blower is provided in the Train B Post LOCA H_2 Control Subsystem. The function of this blower is to continuously purge normally 30 cfm from the Containment atmosphere during normal power operations to the Aux. Bldg. Vent Exhaust. The Containment supply blower is powered from 120 VAC RPB 8 circuit 16. The interlocks for this fan are that four Train B valves must be open and all four local panel **LOCAL/REMOTE** switches are in **REMOTE** in order to start this fan. The following valves must be open with the associated local control switch in **REMOTE:**

- LOCA-2B/MV-32146
- SA-7003B/MV-32148
- LOCA-100B/CV-31725
- LOCA-201B/CV-31727

Post LOCA H_2 Control Valves & Operators-TRAIN A			
Train A Valves	**Air/Motor Oper.**	**Local Switch**	**Pwr Supply**
LOCA-2A	MV-32145	ES-19577	MCC 1-52B
LOCA-3A	CV-31386	CS-19579	IA
LOCA-10A	CV-31387	CS-19580	IA
LOCA-100A	CV-31724	ES-19686	IA/SV-33788
LOCA-201A	CV-31726	ES-19698	IA/SV-33814
SA-7003A	MV-32147	ES-19578	MCC 1-52B
SA-7010A	----	SA-7003A	IA/SV-33394

Post LOCA H_2 Control Valves & Operators-TRAIN B				
Train B Valves	**Air/Motor Oper.**	**Control Sw.**	**LOCAL Sw.**	**Pwr Supply**
LOCA-2B	MV-32146	ES-40017	ES-19581	MCC 1-62B
LOCA-3B	CV-31388	CS-19583		IA
LOCA-10B	CV-31389	CS-19584		IA
LOCA-100B	CV-31725	ES-40019	ES-19687	IA/SV-33789
LOCA-201A	CV-31727	ES-40021	ES-19715	IA/SV-33815
SA-7003B	MV-32148	ES-40018	ES-19582	MCC 1-62B
SA-7010B	----	SA-7003B		IA/SV-33395
LOCA Valves PO's	K-565			

Valve position indication and white status light indication is provided in the Control Room for all Containment isolation valves except check valves. Control of the valves is exercised from local Post LOCA H_2 Control Station Panels 1A and 1B in the CC Heat Exchanger Area. Red **(OPEN)** and green **(CLOSE)** indicating lights are provided with the control switches on the local panel.

The Post LOCA H_2 Control Train B valves can be operated from the Control Room Mechanical Vertical Panel "A" provided the **LOCAL/REMOTE** switches are in the **REMOTE** position.

- If any **LOCAL/REMOTE** switch is placed in **LOCAL**, Alarm 47016-3 is sounded in the Control Room.
- Placing a Train B Post LOCA valve in **LOCAL** defeats the automatic isolation signals from Containment Vent Isolation (CVI) and from Zone SV Isolation.

The local switches and indications are enabled and the Control Room indications and switches are disabled.

3.11 Mechanical Vertical Panel "A" Controls

The Mechanical Vertical Panel "A" has the following control switches associated with the RBV System and the Post LOCA H_2 Control Subsystem:

Switch		Switch No.	Function
Containment Purge & Vent Supply Fan		ES-46533	**OFF/AUTO/ON** Spring return to **AUTO** from **ON**
Containment Vent Exhaust Fan		ES-46831	
Containment Purge Exhaust Fan		ES-46534	
Containment Ventilation Isolation Valves	RBV-1	ES-46598	**CLOSE/AUTO/OPEN** Spring return to **AUTO** from **OPEN**
	RBV-2	ES-46596	
	RBV-3	ES-46595	
	RBV-4	ES-46597	
Containment Purge & Vent Supply Inlet Damper TAV-10/ CD-34030		---	Is opened and closed in association with various temperatures and DPS
Turbine Bldg Makeup Air Winter Operations Damper TAV-11/ CD-34034		ES-46830	**OPEN/CLOSE** Two position maintained
Containment Purge & Vent Supply Unit Exhaust Damper TAV-12/CD-34033		ES-46532	**CLOSE/AUTO/OPEN** Spring return to **AUTO** from **OPEN**
Containment Exhaust Damper, RBV-5/CD-34006		ES-46531	
CRDM Cooling Fans	1A	ES-46823	**ON/AUTO/OFF** All three positions maintained
	1B	ES-46824	
Containment Dome Fans	1A	ES-46589	**ON/AUTO/OFF** Spring return to **AUTO**
	1B	ES-46590	
Reactor Support Cooling Fans	1A	ES-46535	**OFF/AUTO/ON** All three positions maintained
	1B	ES-46536	
Reactor Gap & Neutron Detector Cooling Fans	1A	ES-46537	
	1B	ES-46538	
Power Operated Vacuum Relief Valves VB-10A/CV-31337		ES-46828	**CLOSE/AUTO/OPEN** Spring return to **AUTO**
Power Operated Vacuum Relief Valves VB-10B/CV-31338		ES-46829	
Containment Fan Coil Units	1A	ES-46539	**PULLOUT/STOP/AUTO/START** With the switch spring return to **AUTO** from the **STOP** or **START**
	1B	ES-46540	
	1C	ES-46541	
	1D	ES-46542	

Mechanical Vertical Panel "A" Controls (Continued)			
Switch		**Switch No.**	**Function**
Containment Fan Coil Emergency Discharge Dampers	RBV-150A CD-34130	ES-40013	The switches spring return to **AUTO** from **CLOSE** position.
	RBV-150B CD-34131	ES-40014	
	RBV-150C CD-34132	ES-40015	
	RBV-150D CD-34133	ES-40016	
Post LOCA H_2 Containment Vent, LOCA-2B/MV-32146		ES-40017	**OPEN/AUTO/CLOSE** Spring return to **AUTO**
Post LOCA H_2 to Recombiner 1B LOCA-1OOB/CV-31725		ES-40019	
Post LOCA H_2 Recombiner 1B to Containment Iso. LOCA-201B/CV-31727		ES-40021	
H_2 Dilution to Containment Isolation SA-7003B/MV-32148		ES-40018	

NOTE: Each of the above control switches has a red (**RUNNING** or **OPEN**) and a green (**STOPPED** or **CLOSED**) light indication with the associated switch.

3.12 Controls - Local

The Post LOCA H_2 Control valves for:

- Train A are operated from Local Panel 1A.
- Train B valves are operated from Local Panel 1B.

Component	Switch No.	Function
Post LOCA H_2 Vent Isolation, LOCA-2A/MV-32145	ES-19577	**OPEN/NORMAL/CLOSE** Spring return to **NORMAL**
Post LOCA H_2 to Recombiner 1A, LOCA-100A/CV-31724	ES-19686	**CLOSE/ BLANK/OPEN** Spring return to **BLANK**
Post LOCA H_2 Recombiner 1A to Containment Isolation LOCA-201A/CV-31726	ES-19698	**CLOSE/AUTO/OPEN** Spring return to **AUTO**
Hydrogen Dilution to Containment Isolation SA-7003A/MV-32147	ES-19578	
Post LOCA H_2 Vent Isolation, LOCA-3A/CV-31386 (variable loader)	CS-19579	**OPEN/CLOSE**
H_2 Sample to Gas Analyzer LOCA-10A/CV-31387	CS-19580	
Post LOCA H2 Vent Isolation LOCA-2B/MV-32146	ES-19581	**OPEN/AUTO/CLOSE** Spring return to **AUTO**

Controls – Local (Continued)

Component	Switch No.	Function
Post LOCA H_2 to Recombiner B, LOCA-100B/CV-31725	ES-19687	CLOSE/AUTO/OPEN Spring return to AUTO.
Post LOCA H_2 Recombiner B to Containment Isolation, LOCA-201B/CV-31727	ES-19715	
Hydrogen Dilution to Containment Isolation SA-7003B/MV-32148	ES-19582	
Post LOCA H_2 Vent Isolation, LOCA-3B/CV-313889 (variable loader)	CS-19583	OPEN/CLOSE
H_2 Sample to Containment Air H_2 Analyzer LOCA-10B/CV-31389	CS-19584	
2" Containment Supply Blower	ES-19---	ON/OFF
Instrument Air to Containment Control Valves	IA-1001A IA-1001B	Local operation at the valve.

CFCUs A and B can be operated from the Dedicated Shutdown Panel (DSP).

Component		Switch No.	Function
CFCU A		ES-87167	OFF/AUTO/ON Spring return to AUTO from ON
		ES-87168	LOCAL/REMOTE
CFCU B		ES-87169	OFF/AUTO/ON Spring return to AUTO from ON
		ES-87170	LOCAL/REMOTE
SW Cooling Valve for CFCU A	SW-903A	ES-87123	Control Switches
SW Cooling Valve for CFCU B	SW-903B	ES-87125	

3.13 Indications

The Omniguard System is connected to monitor and alarm in the event of any abnormal temperature conditions at the following points:

- Outlet air at all Containment fan coils
- Outlet air from the CRDM shroud
- Outlet air from each pair of supports in the Reactor Support Cooling Subsystem
- Ambient space conditions in each of the RXCP Vaults A and B
- Ambient space conditions at Reactor Containment Bldg elevations 592', 606', and 626'

Six differential pressure (DP) measurements are taken between the Containment and the outside atmosphere.

- Three DP transmitters have a range of 0 to 30 psig,
- Three DP transmitters have a range of 0 to 60 psig.

All six transmitters provide signals to the engineered safeguards circuits.

Four DP signals are used to provide initiation of the Vacuum Relief Subsystem. Containment relative humidity is measured and indicated in the Control Room.

- One normal Containment DP transmitter provides indication in the Control Room of -0.5 to +2.5 psig.
- One DP transmitter having a range of 0 to 0.8 psid measures the difference in pressure between the Shield Bldg. Annulus and the Reactor Containment Bldg.
- Two DP switches measure the difference between Shield Bldg. Annulus and the Reactor Containment Bldg.

In the Post LOCA H_2 Control Subsystem:

- Independent flow indicators provide the Operator with air flow information to adjust the throttle valve through air loaders.
- A secondary instrument (DPI-11407) for the Annulus to Aux. Bldg. DP, provides a warning to the Control Room Operator if the air flow is causing a loss of normal Shield Bldg. Annulus DP.

An IA flow of 25 cfm is sufficient to limit H_2 concentration inside Containment. A maximum air flow limit of 100 cfm on the Kates valve ensures the Shield Bldg. Annulus DP is maintained.

Indication	Range	Normal	Location	Instrument
IA Supply (Kates)	0-100 cfm	0-25	1B MS Pen. Rm	IA-1002A
IB Supply (Kates)	0-100 cfm	0-25	1A MS Pen. Rm	IA-1002B
Emerg. Air Supply	---	--	1B MS Pen. Rm	SA-7001A
Emerg. Air Supply	---	--	1A MS Pen. Rm	SA-7001B
IA Air Flow to CNTMT	20-200 cfm	0	1B MS Pen. Rm	FI-18237
IA Air Flow to CNTMT	20-200 cfm	0	1A MS Pen. Rm	FI-18238
IA Press Upstream	0-160 psig	93 psig	1B MS Pen. Rm	PI-11405
IA Press Upstream	0-160 psig	93 psig	IA MS Pen. Rm	PI-11406
IA Press Downstream	0-160 psig	93 psig	IB MS Pen. Rm	PI-11403
IA Press Downstream	0-160 psig	93 psig	IA MS Pen. Rm	PI-11404
Containment Pressure	0-30 psig	0	Control Room	PI-41505-01
Containment Pressure	0-30 psig	0		PI-41505-02
Containment Pressure	0-30 psig	0		PI-41505-03
Containment Pressure	0-60 psig	0		PI-41505-04
Containment Pressure	0-60 psig	0		PI-41505-05
Containment Pressure	0-60 psig	0		PI-41505-06
Containment DP	0-0.8 psid	0		PI-41512-01
Shield Bldg DP	-.5-2.5 psid	0		PI-41512-02
Annulus to Aux. Bldg.	Psid	0		PI-11407

3.14 Alarms - RBV System and Post LOCA

NOTE: See Alarm Response Sheets in Control Room for Setpoint specifics.

Annunciator	Window Label
47005-32	CNTMT VACUUM BKR VALVES OPEN
47005-33	CNTMT VAC BRKR AIR RSVR PRESS LOW
47005-34	INLET DAMPERS AIR RSVR PRESS LOW
47005-41	REAC GAP AND NEUTRON DET FAN OFF
47005-42	REACTOR SUPPORT COOLING FANS OFF
47005-43	CRDM COOLING FANS OFF
47007-44	CNTMT FAN COIL EMERG DISCH DMPR ABNORMAL
47011-35	CNTMT EMERG DISCH DMPRS ACTIVATED
47016-33	POST LOCA H_2 1B VALVES IN LOCAL CONTROL
47022-13	TRAIN A CONT RM SWITCH IN PULL OUT
47022-14	TRAIN B CONT RM SWITCH IN PULLOUT
47023-15	CONTAINMENT VENTILATION ISOLATION
47030-33	BUS 1-51 FEEDER BREAKER OVERCURRENT TRIP
47032-33	BUS 1-61 FEEDER BREAKER OVERCURRENT TRIP

3.15 Sequence of Event Recorder (SER)

NOTE: See Alarm Response Sheets in Control Room for Setpoint specifics.

SER No	Window Label
49001202	RBV, VACUUM BKRR VLV 1A OPEN
49001203	RBV, VACUUM BRKR VLV 1B OPEN
49001204	RBV, AIR RSVR TNK 1A PRESS LOW
49001205	RBV, AIR RSVR TNK 1B PRESS LOW
49001288	RBV, INLET DMPRS AIR RSVR PRESS LOW
49001285	RBV, FAN COIL 1C SWITCH PULLED OUT
49001286	RBV, FAN COIL 1D SWITCH PULLED OUT
49001296	RBV, FAN COIL 1A SWITCH PULLED OUT
49001298	RBV, FAN COIL 1B SWITCH PULLED OUT
49001580	CONT FAN COIL 1D BKR 15104 TRIP
49001581	CONT FAN COIL 1D BKR 15104 0C TRIP
49001582	CONT FAN COIL 1C BKR 15105 TRIP
49001583	CONT FAN COIL 1C BKR 15105 0C TRIP
49001584	CONT FAN COIL 1B BKR 16104 TRIP
49001585	CONT FAN COIL 1B BKR 16104 0C TRIP
49001586	CONT FAN COIL 1A BKR 16105 TRIP
49001587	CONT FAN COIL 1A BKR 16105 0C TRIP

3.16 System Interrelationships

The Reactor Building Ventilation System interrelates with the following systems in performing its design functions:

Reactor Coolant System (RCS)	The RBV provided cooling to various instrumentation associated with the RCS.
Control Rod Drive Mechanisms (CRDMs)	Is provided with cooling to remove excess heat from the CRDMs to ensure proper operation without overheating.
Neutron Detectors	Are cooled to ensure the Nuclear Instrumentation System provides accurate indication of Reactor power.
Fuel Handling & Containment General	Is provided cooling to the general spaces to maintain temperature conditions suitable for equipment and personnel.
Instrument Air	Operates many pneumatic valves and dampers.
SW System	• Provides cooling water to the CFCUs and to the CRDM shrouds. • Provides deluge water for the charcoal filter banks if the heat detectors exceed 200°F.
Process Sampling	Connections are provided in various locations, for monitoring for particulate activity, radioactive iodine, noble gases (Xenon, Krypton), and tritium. (See Figure KNP-RBV4)
Hydrogen Recombiner (portable)	Connects into the Post LOCA H_2 Control Subsystem.
Electrical Distribution System	Supplies electrical power for fans, fan coil units, motor operated dampers, etc.
RBV System	Interconnects with the Shield Bldg Ventilation (SBV) System and the Aux Bldg Ventilation (ACA) System.

3.17 System Interlocks

The 2" Containment Supply Blower in the 2" vent subsystem is interlocked to prevent starting unless all four Train B local panel **LOCAL/REMOTE** switches for valves LOCA-2B, LOCA-100B, LOCA-201B, and SA-7003B are in **REMOTE**.

The CRDM Fans 1A and 1B are interlocked with the associated motor operated damper on each fan discharge. The damper automatically opens when the fan is started. Annunciator 47005-43 is actuated if both fans are off.

The Reactor Gap & Neutron Detector Cooling Fans 1A and 1B are interlocked with the associated motor operated damper on each fan discharge. The damper automatically opens when the fan is started. Annunciator 47005-42 is actuated if both fans are off. Only one fan can operate at a time.

The Reactor Support Cooling Fans A and B are interlocked with the electrical heating coils upstream of each fan. The associated fan must be running for the heating coil to cycle to maintain temperature. Only one fan can operate at a time.

The Containment Purge Exhaust Fan trips on any one of the following:

- High DP, 0.3 psid, across Containment Purge & Vent Supply Inlet Damper TAV-10
- Freezestat temperature of less than 40°F
- Initiation of the exhaust filter deluge heat detectors
- Closure of any Containment isolation valve or damper
- High Containment radiation, Aux. Bldg. vent exhaust high radiation, or any CVI signal
- Stopping the Containment vent exhaust fan
- Loss of power or loss of IA
- Temperature of the condensate from either of the two preheat coils is below 100°F
- Rotating the control switch to **OFF**

The four CFCUs receive an automatic start signal from an SI sequence step 5. If both fans supplying a header are running, motor operated Dampers RBV-100AB or RBV-100CD (in the header) automatically open to permit direct air flow to the upper Containment elevations.

4.0 Precautions and Limitations

The Containment should be kept at approximately atmospheric pressure at all times.

During plant startup from refueling to hot shutdown, thermal expansion of the Containment air must be released by operation of the Containment Purge & Vent Subsystem, or via the 2" Post LOCA H_2 vent subsystem.

Discharge permits should be obtained from Radiation Protection Group when Containment pressure reaches 1.5 psig. A Discharge Permit is not required if using the 2" vent subsystem.

Do not initiate a Containment purge or vent if Annunciator 47016-42 (**AUTO CVI ON HI RAD DISABLED**) is in an alarmed condition.

Reactor Containment Building purge must be filtered through the purge filter (HEPA-charcoal) whenever the concentration of iodine and particulate isotopes is to be reduced on a batch basis. (TS 7.4.4)

When the Reactor is critical, NRC notification is required prior to ventilating Containment with the 36 inch RBV valves.

During release of gaseous wastes from Containment, the following conditions must be met:

- R-12 or R-21 monitor is operable, R-21 Sampler is operable, and R-21 flow rate measuring devices are operable.
- If less than minimum channels operable, take appropriate actions as described in Technical Specifications, TS Table 7.2.

While performing a gaseous waste release to the Aux. Bldg. Vent System using the 2" Containment vent release path, at least one train of Aux. Bldg. Vent System must be operating.

All Containment Vent Operations must be recorded on the Containment Vent Data Sheet and logged in the Control Operator Log. Upon completion of the Containment vent, the Containment Vent Data Sheet is forwarded to the Health Physics Supervisor.

Do not allow Containment pressure to exceed 21 psig or H_2 concentration to exceed 3.5%.

Do not allow the Shield Bldg annulus to become pressurized relative to the Aux Bldg as indicated on DPI-11407 at the Post LOCA H_2 Control Panel.

5.0 References

1.0 Summary	N-RBV-18-CL, N-RBV-18A, B & C, A-RBV-18
2.0 Functions	
3.0 Design Description	USAR 5.4 OPERM-601 E-1608 E-3104 OPERM-403 OPERM-602 E-1609 E-3310 OPERM-547 E-2068
	I & C Description #18, ASHRAE Std. 52.1-1992 Annunciator Response 47005-,47007, 47011-, 47022-, etc
	For complete listing of DCRs associated with System 18, see: Programs\KNPP applications\ Physical Changes\DCR Database\Sys 18
4.0 Precautions & Limitations	N-SBV-24-CL, N-RBV-18C TS 3.3.c.1.C, 3.3.c.2.B & C, 3.6.c. & d. Annunciator Response 47005-,47007, 47011-, 47022-, etc I & C Description #18
5.0 References	KNPP Technical Specifications
6.0 Procedures	Refer to a controlled copy of KNPP procedures.
7.0 Appendices	See attached Figures

5.1 Technical Specifications

NOTE: Refer to a Controlled Copy of the KNPP Technical Specifications for plant operations.

TS 3.3.c.1.A.2 The Reactor must not be taken critical unless two trains of Containment fan coil units are operable with two fan coil units in each train (all four fan coil units must be operable).

TS 3.3.c.1.A.3 During power operation or recovery from an inadvertent trip, any one of the four trains of Containment fan coil units or Containment spray train may be out of service for a period of up 7 days provided the remaining three trains are operable.

TS 3.3.c.2.C During power operation or recovery from an inadvertent trip, any two of the four trains of Containment fan coil units or Containment spray train may be out of service for a period of up to 72 hours provided the remaining two trains are operable.

TS 3.6.d & e — If the internal pressure of the Reactor Containment Building exceeds 2 psig., the condition must be corrected within eight hours or the Reactor must be shutdown. The Reactor must not be taken above the cold shutdown condition unless the Containment ambient temperature is greater the 40° F.

TS 3.8.a.8 — The Containment Ventilation & Purge Subsystem, including the capability to initiate automatic CVI, must be tested and verified to be operable immediately prior to and daily during refueling operations.

6.0 Procedures

NOTE: Refer to a cintrolled copy of KNPP procedures.

7.0 Appendices

7.1 Attached Figures

Figure	Title	KNPP DWGs
KNP-RBV1	Reactor Building Ventilation (RBV) 1	OPERM-602
KNP-RBV2	Reactor Building Ventilation (RBV) 2	
KNP-RBV3	Reactor Building Ventilation 3	OPERM-547
KNP-RBV4	Post LOCA Hydrogen Control – Train B	OPERM-403
KNP-RBV5	Containment Fan Coil Logic	E-3104
KNP-RBV6	Containment Vent & Purge Supply Fan/Rx Support Cooling	OPERM-601 OPERM-602
KNP-RBV7	RBV Fan Logics	E-1608 E-1609
KNP-RBV8	Containment Purge & Vent Supply & Exhaust Fans	E-1608
KNP-RBV9	Vacuum Relief & Containment Isolation Logic	E-1608 & E-1609

NOTE: The "Figures" (drawings) previously associated with the System Descriptions will not be revised and updated at this time. Instead (obsolete) copies of these Figures have been stamped HISTORICAL and temporarily attached.

NMC KNPP System Description	System No. 21	Rev. 1
Title Spent Fuel Pool Cooling and Cleanup System (SFP)	Date 07/03/02	Page 1 of 17

Author ______ James P. Brandtjen	Op's Review ______ David D. Mielke
Tech Review ______ David D. Mielke	Approved ______ Manager of Engineering

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INDEX

1.0 Summary

1.1 Overview (Figure KNP-SFP1)

The purpose of the Spent Fuel Pool Cooling and Cleanup (SFP) System is to:

- remove decay heat from the spent fuel,
- filter and demineralize the water in the spent fuel pools,
- maintain minimum boron concentration in the SFP water with boron addition via the Chemical and Volume Control System (CVCS),
- maintain the SFP temperatures below a maximum temperature of 150° F.

The Spent Fuel Pool currently has storage locations for 1205 fuel assemblies in high density storage racks. The Spent Fuel Pool Cooling System is designed to remove decay heat from spent fuel stored in the pools. The system is currently analyzed to ensure heat removal capability equivalent to the decay heat generated by 1205 fuel assemblies.

The SFP System is located in the northwest portion of the Auxiliary Building (Aux. Bldg.). This arrangement and location allows accessibility for maintenance and visual inspection of the components within the system during normal operation. The SFP area is surrounded by a cyclone fence to prevent accidents and provide additional security for Special Nuclear Material. The immediate area is restricted. Flotation devices are available if the need arises.

The basic SFP System components for normal cooling and cleanup of the spent fuel pools include:

- SFP Liners
- SFP Pumps A and B
- SFP Filters A and B
- SFP Heat Exchanger (HX)
- Associated valves, piping and instrumentation.

Each SFP pump supplies about 1/2 the total design flow of the SFP heat exchanger tube side.

The purification and cleanup components consist of:

- SFP Demineralizer (Demin.) Prefilter
- SFP Demin.
- SFP Demin. Post-Filter
- Refueling Water Purification Pump
- Purification Flow Control Valve FPC-204
- Associated piping, manual valves, and instrumentation

Redundancy of the components used in this system is not required because of the large heat absorption capacity of the water in the spent fuel pools.

- If one SFP pump should fail, the remaining duplicate SFP pump can handle the maximum heat load following a normal refueling.
- If a SFP pump failure should occur with the freshly discharged fuel from a complete core off-load in the pools, the water inventory is large enough to absorb heat released, allowing sufficient time to repair or replace the failed pump.

Since the possibility of failure of the SFP Heat Exchanger exists, the Residual Heat Removal (RHR) heat exchanger can be used as a backup by diverting SFP coolant through existing interconnections.

The return lines discharge into the pools above the top of the fuel. Loss of coolant from the spent fuel pools by line failure and subsequent siphon action is prevented by installation of check valves. The SFP suction piping connects to the SFP well above the fuel assemblies to prevent loss of cooling or shielding due to a system failure. The spent fuel racks are designed with neutron absorbing plates between the storage cells. The racks in pools A and B contain boron carbide plates. The Canal Racks contain Boral plates. The neutron absorbing plates prevent criticality with nonborated water in the pools. A leak detection system is installed and allows checking the liner welded seams for leaks.

1.2 System Operation versus. Plant Modes

Plant Mode	System Support
Startup	The Spent Fuel Pool Cooling and Cleanup (SFP) System is normally operating at all times regardless of the plant conditions. This manually controlled system may be safely shut down for reasonable time periods provided the KNPP Technical Specifications (TS) are reviewed and met.
Normal Power	
Shutdown	
Refueling	The SFP System may be aligned for refueling cavity purification.
Casualty Events	See Section 1.8 of this System Description
Infrequent Operations	Not applicable
Maintenance	Maintenance may be performed on the SFP System while the plant is operating as long as KNPP Technical Specifications are referred to and limiting conditions for operation are met.

1.3 Startup (OP's# N-SFP-21, N-SFP-21-CL)

With all valves aligned to allow drawing water from both pools and return to Pool A and/or B, with both pumps and filters aligned for service and with the heat exchanger (HX) properly aligned, the SFP loop is started by turning the Control Room control switch for one SFP pump to the **ON** position. Resins should be charged into the SFP demineralizer prior to this time. Flow through the purification bypass loop is established by manually setting the desired flow rate through the loop on the self contained flow regulating valve FPC-204, and opening the inlet and return valves, FPC-200 and FPC-206 respectively, for the purification loop. The SFP System is now in its normal operational mode. The SW System should be operable and aligned to provide water to the SFP HX.

Startup of the RWST purification loop is made by properly aligning valves to draw water from the RWST and to return it to the RWST, while at the same time isolating the SFP loop from the purification loop. The RWST purification pump is started locally.

1.4 Normal Operation (OP# N-SFP-21)

During normal operation of the SFP System, only one of the two SFP pumps is used to draw water from the pool and circulate the water through the filters, heat exchanger, and back to the pools. During full core off-load conditions, both pumps are used to provide additional cooling capacity. During this process, approximately 10% of the SFP pump discharge flow is passed through a pre-filter, demineralizer, post-filter loop and returned to the main flow path at the heat exchanger outlet. The purification loop is provided to remove fission products which can contaminate the pool as a result of leaking spent fuel assemblies. Also, the purification loop is capable of cleaning and purifying the water stored in the refueling water storage tank (RWST). Simultaneous purification of the spent fuel pool water and the RWST is not possible.

Normal operation of the SFP loop is with one pump operating (red light **ON**, switch in **ON**) and with one pump in **AUTO** (green light lit). Both SFP filters are in service. Water is taken from pools A and B through skimmers and returned to pools A and B. The purification loop is in service at the design flow rate of 60 gpm. SW is available to the SFP HX. Low SFP level alarms should be investigated for leaks from the SFP System.

Low level alarms are most probably caused by excessive seal leakage from the SFP pumps or through valve stems. Misalignment of the RWST purification loop could also be a cause.

Normal makeup is made from the RWST through the use of the RWST purification pump, or by the use of the boric acid blender in the CVC System. High level alarms are most probably the result of SW in-leakage through the SFP HX. Misalignment of the RWST purification loop could also be a cause. High temperatures are caused by loss of cooling, most probably by malfunction of the SW Flow Control Valve on the outlet of the SFP HX shell side.

1.5 Shutdown

The SFP cooling and purification loops may be safely shut down for reasonable times for the maintenance and/or replacement of malfunctioning components. The entire SFP System may be shutdown from the Control Room by turning both SFP pump control switches to **STOP**.

1.6 Refueling

During refueling, the SFP System also provides the capability of filtering refueling cavity water for improving visual clarity. Filtering is accomplished by passing water to the suction of the SFP pumps through a spectacle flanged interconnection with the RHR System. Water is returned to the RHR heat exchanger outlet or to the SFP. If water is returned to the SFP, water flow through the transfer tube is necessary to equalize water levels.

NOTE: This process is not proceduralized.

1.7 Abnormal Conditions (OP# A-SFP-21)

This procedure describes the action to be taken on a pump trip, pool level high/low level alarm, high temperature alarm, and a SFP HX DP low alarm. The procedure provides symptoms (annunciators), immediate actions and subsequent actions to correct the cause of the alarm and restore to normal conditions.

If a failure in the SFP HX occurs, the affected tubes could be plugged or RHR HX 1A can be made available through existing valving and piping. This action is accomplished by manually closing the valves FPC-7 and FPC-8 at the inlet and outlet of the SFP exchanger and opening the spectacle flanges thereby diverting the coolant through the RHR HX and returning the cool water to the SFPs.

1.8 Emergency (OP# E-SFP-21)

Assuming that a leak or rupture might occur in the SFP pump suction lines, these lines penetrate the pools 2' below the high water level, elevation 647'-4". This level assures that adequate coolant remains in the pools to allow time for repair.

If a SFP pump fails with the fuel from a freshly discharged full core off-load stored in the pools, the pools have large enough volumes to absorb the heat generated for a sufficient time to allow the failed pump to be repaired or replaced.

If high radiation alarm is received on Radiation Monitor R-5, Emergency Service Water to SFP Isolation Valve SW-1497 is opened to recover the SFP level.

2.0 Functions

Specific functions include the following:

- Cooling is provided by two SFP pumps and a SFP heat exchanger.
- Cleanup is provided by a pre-filter, demineralizer, post-filter, and a purification pump and flow control valve.

All components in the system are **QA Type 3**. The piping between the pools and the isolation valve at the point where the piping enters the concrete walls are **QA Type 1.** Also, the SW piping and components that provide the emergency SFP cooling water supply are QA1. The piping and valves of the leak detection system of the spent fuel pools are **QA Type 2.**

A leakage collection and detection system allows checking the liner seam welds for leakage.

3.0 Design Description

3.1 System Arrangement and Flowpaths (See Figure KNP-SFP1)

The SFP System was originally designed to remove the heat released from 1-1/3 Reactor cores stored in both pools A and B. SFP System heat removal requirements were established on the assumption that:

- 1/3 Reactor core is stored with a 30 day decay time,
- 1/3 Reactor core is freshly discharged from the Reactor with 5 days decay time,
- 1/3 Reactor core with 10 days decay time,
- 1/3 Reactor core with 15 days decay time.

With the installation of high density storage racks, and the absence of commercial fuel reprocessing, the SFP heat removal capabilities have been re-evaluated. The revised heat removal requirement is based on the heat from a full core off-load that completely fills the 1205 cell rack capacity. Assuming this decay heat load and normal operating conditions, the SFP System can maintain the SFP water at less than 150° F. Following a normal refueling (partial core off-load) the SFP System maintains a maximum of 140° F.

The two SFP pumps deliver approximately 60 gpm to the SFP demineralizer (purification loop). The cleanup portion of the SFP System consists of two half-capacity filters that are sized to pass 425 gpm each. The filters remove particulate matter down to 15 micron absolute. The maximum differential pressure (DP) across the filters is 10 psid.

The coolant in the SFP System is a solution of boric acid in water consisting of about 2,200 ppm boron. Boric acid makeup is provided by the CVCS and is injected into the SFP System downstream of the SFP heat exchanger.

All material used in construction of the entire SFP System, including the SFP liners, is austenitic stainless steel (SST).

During normal operation, one SFP pump is operating and one SFP pump is in **AUTO** standby. The demin prefilter and post-filter are in operation. Surface skimmers are in service with water taken from both Pools A and B and returned to pools A and B. The purification loop is operating at a design flow rate of 60 gpm. SW is available to remove the heat from the SFP HX. Normal makeup is available from the boric acid blender in the CVCS.

Suction is taken on the spent fuel pools through 6" piping that penetrates the SFP less than 3' below the water surface. The location of the suction piping prevents a single active failure from uncovering the fuel assemblies.

The operating SFP pump discharges into a common 6" header. Flow is through the SFP filters, A and B in parallel, to the SFP HX. Some SFP flow (60 gpm) goes through the prefilter, demin, and post filter to remove fission products and reduce SFP water activity. Flow through the demin path is regulated by Flow Control Valve FPC-204/ CV-31293. A cross-connect between the RWST and the SFP demin is provided. Either the SFP or the RWST may be connected to the demin for purification. Normally the SFP is connected to the demin. The Flow Control Valve FPC-204 must compensate for the differing discharge pressures of the SFP pumps and the RWST purification pump.

The SFP HX receives the water from the outlet of the SFP filters. SW flows through the shell side of the SFP HX. Cross-connects with spectacle flanges are provided to the RHR 1A HX. If the SFP HX is out of service, RHR 1A HX is the alternate heat removal path. The outlet from the SFP HX goes to the return header.

The SFP return header splits and enters SFP's A and B above the top of fuel assemblies. Check valves in the return lines just before entering the pool prevent siphoning the pool in the event of a line failure. Tapping into the return header are the boric acid addition line from the CVCS, Reactor makeup water line, and 6" emergency fill line from the SW System.

The RWST purification loop is used to cleanup RWST water. This cleanup is normally performed after a refueling. A suction is taken from the RWST by the purification pump. The pump discharges to the SFP demin prefilter and to a RWST recirculation line. Purification is performed in accordance with chemistry specifications.

The SFP leak detection system functions to detect leakage through seam welds of the pool liner and the transfer canal. A total of ten detection zones are provided.

A solenoid operated valve, FPC-51, is a SFP flushing line. This connection can be used by chemistry technicians in a post-accident situation to flush primary system sample lines with a borated water source after obtaining a highly radioactive sample. This flush valve is controlled from the High Radiation Sample Room (HRSR). See System Description 37, entitled, "Primary Sampling System," for more details on the HRSR and sampling.

3.2 Spent Fuel Pool Liners

The SFP consists of four SST lined cavities.

- The north pool (A) has an almost square configuration and is used for spent fuel storage and for cask loading.
- The south pool (B) is rectangular and is used for spent fuel storage.
- The Canal pool is used for storage of old spent fuel assemblies.
- The fuel transfer canal is used to move fuel between the reactor vessel and the SFP.

Pools A and B may be isolated from each other by a removable SST gate fitted with a reinforced inflatable seal. Also, pool B may be isolated from the fuel transfer canal by a removable gate. The pools normally contain demineralized water containing approximately 2,200 ppm boron.

The pools are located in the Auxiliary Building refueling area. The top of each pool is at the 649'-6" elevation, and the bottom of the pool is at the 608' elevation. The pools are approximately 40' deep with fuel occupying the bottom 14'.

The pool liners are designed with leak tight integrity so that stresses caused by such effects as thermal expansion, hydrostatic and mechanical loads do not impair the SFP functions. Since the coolant in the SFPs is potentially contaminated with radioactive material, the SST liner is provided as a barrier between the concrete walls and the radioactive materials.

Boric acid addition to the SFPs is through a normally closed, manually operated diaphragm valve, CVC-414. Boric acid comes from the CVCS blender. The manual valve CVC-414 is located in the SFP HX room. Reactor makeup water is supplied via normally closed, manually operated valve MU-1009. The emergency makeup to the SFPs is through a normally locked closed, manual gate valve, SW-1497. SW1497 and MU-1009 are also located in the SFP HX Room.

Spent Fuel Pool Liners Data			
Manufacturer	Nooter Corporation		
Technical Manual	K-233		
Material	Type 304 SST, ASTM A240		
Nominal Plate Thickness sides/bottom	3/16" / 1/4"		
	Pool 1A (North)	**Pool 1B (South)**	**Canal Pool**
*Height	38'-6"	38'-6"	38'-6"
Width	19'	19'	4'
Length	17'	32' 6"	30'-8"
*Total Volume	12,436 ft^3 93,028 gallons	23,774 ft^3 177,842 gallons	4,728 ft^3 35,368 gallons
*Fuel Transfer Canal Volume (south of canal pool)	7,315 ft^3		

*Assumed water level = 646'-6"

3.3 **Fuel Transfer Canal**

The Fuel Transfer Canal is located just east of the Spent Fuel Pool B. The transfer canal provides a path for [illegible] transfer between Pool B and the Reactor via the fuel transfer tube. The Fuel Transfer Canal is the same construction as the SFPs.

A fuel upending rig is at the south end of the canal. The upender is powered from MCC 1-35D. The upender is normally operated during refueling operations by a Fuel Handler.

3.4 **Spent Fuel Pool Gates**

Two SFP gates are provided. Each SFP gate is designed to withstand the water pressure at a [illegible] water level 648'-2" with the adjacent pool dewatered. The SFP gates are supported in a guide slot mounted in the channel. The gates are lifted vertically by attaching a hoist to the two lifting eyes on the gates. Gate sealing is enhanced through application of air pressure to an inflatable seal on the sides and bottom of the gate.

Sealing is accomplished by inflating the rubber seal to expand it against the SST liner plate. The seals are premolded ethylene propylene fabric reinforced rubber. Inflation connection fittings are provided at the top of the seal and are the "snap-on" type. Nitrogen is used as a backup to the gate seals. A pressure regulator on the nitrogen bottle maintains a 60 psig pressure available to supply the seals if instrument air is lost. The seals are designed and constructed to be replaceable at the site.

SFP Seal Gate Data	
Manufacturer	Nooter Corporation
Technical Manual	K-233-26
Material	Type 304 SST, ASTM A240
Gate Width	2' 4"
Gate Height	27' 2"
Gate Thickness	3/4 "
Gate Weight	2700 pounds
Seal Manufacturer	Press-Ray Corporation
Seal Material	Material 60 Durro EPDM Compound No. E603
Inflation Pressure	90 psig

3.5 SFP Skimmers

Pools A and B have two skimmer boxes located at one end (Pool A at north end , Pool B at south end) to allow particulate matter floating on the pool surface to be drawn off and filtered out of the water. This filtering provides optical clarity to the water to enhance fuel handling operations. The flow rate through each skimmer is adjusted manually by raising or lowering the vertically adjustable gate on each skimmer box. Clean-out provisions exists for each skimmer box to allow removal of large entrapped particles. All materials are Type 304 SST.

A threaded T-handle rod is stored at the north end of the pools. This T-handle is used for adjusting the skimmer gate height. Normally in each SFP, one gate is full down and one gate is several inches beneath the SFP water surface.

3.6 SFP Leak Detection System (See Figure KNP-SFP2)

A leakage collection and detection system is provided for the SFPs by placement of channels behind the seam welds. The system is divided into 10 zones to allow localization of leaks. The 10 detection lines have normally open manual valves. The 10 detection lines drain to a collection trough which then drains to the floor drain piping.

3.7 SFP Pumps A and B (See Figure KNP-SFP01 and SFP03)

Two single stage horizontal, end suction centrifugal pumps are provided for the SFP System. Both SFP pumps are located in the SFP HX room. The SFP pumps provide the motive force for circulating SFP water through the cooling HX and purification flow paths.

SFP Pumps Data		
Manufacturer	Gould Pumps Inc.	
Technical Manual	K-141-3, thru -6	
Type / Model	Centrifugal / 3196	
Size	3 x 4 - 8G	
Suction O.D. / Disch O.D.	4" / 3"	
Rated Capacity / Total Disch Head	450 gpm / 200 ft TDH	
Pump Design Press / Nominal Disch Press	87 psig / 45-50 psig	
Impeller Size / Material	7-7/16" / 304 SST	
Pump Casing Material	304 SST, A-298, CF-8M	
Bearings	Two roller type, one thrust bearing, one inline bearing	
Pump Seals	Mechanical, John Crane # 9TQP1C1	
Motor Speed / Brake horsepower	3550 rpm /40.0 hp	
Motor Manufacturer	Allis Chalmers	
	1A	**1B**
Equipment Number	145-171	145-172
Power Source	MCC 1-52B	MCC 1-62E
Control Switch	ES-46521	ES-46522

Both SFP Pump A and B have control switches on Mechanical Vertical Panel "A" (See Figure KNP-SFP03). Each SFP pump has a three position maintained switch with **ON/AUTO/OFF** positions. These switches must be in the **AUTO** position to have automatic operation of the SFP pumps. The **OFF** position prevents pump operation.

The control logic for each of the two SFP pumps (See Figure KNP-SFP03) is identical. When a control switch is placed in the **OFF** position, the 480 VAC motor breaker trips open. As long as the switch is left in the **OFF** position, the pump does not respond to automatic or manual SI actuation signals.

When either Control Switch ES-46521 or ES-46522 is in the **AUTO** position, the SFP pump starts in response to a trip of the operating SFP pump or an SI sequence step 10 actuation signal. Once an SFP pump **AUTO** starts due to an SI sequence step 10 actuation signal or a loss of the **ON** pump, the pump continues to run until one of the following occurs:

- The control switch is manually placed in the **OFF** position.
- The **SI RESET** pushbutton is operated, and the control switch is turned to the off position.
- An Auto Inhibit signal is received.

When either SFP Pump Control Switch ES-46521 or ES-46522 is placed in the **ON** position, the pump starts and continues to run until an Auto Inhibit signal or Load shed is received. At step 10 of the SI or BO Sequence, the SFP Pump restarts.

Each SFP pump has indicating lamps directly above the control switch. The indicating lamps are red (**ON**) and green (**OFF**.)

3.8 Refueling Water Purification Pump (Figures KNP-SFP01 and SFP03)

The Refueling Water Purification Pump is a single stage, inline centrifugal pump that is located in the Auxiliary Building basement near the SI pumps. The pump is used to circulate the RWST water through the SFP demin for cleanup after refueling.

Refueling Water Purification Pump Data (145-161)	
Manufacturer	Ingersol Rand Co. (Cameron Pump Div.)
Technical manual	K-261-5
Model Number	1 1/2 VM-15
Serial Number	A70-8100
Rated Capacity	85 gpm
Total Discharge Head	275 ft at 70° F TDH
Nominal Discharge Press	125 - 130 psig
Suction O.D. / Disch O.D.	3" / 1-1/2"
Impeller Material	316 SST
Pump Casing Material	316 SST, A-296, CF-8M
Bearing	Ball Type, in motor
Seals	Mechanical, John Crane # 9TQP1C1
Lubrication	Grease
Brake horsepower	11.6 hp
Motor Manufacturer	Allis Chalmers
Electrical Characteristics	460 VAC, 3 phase, 60 Hz
Power Source	MCC 1-45A
START/STOP Pushbuttons	PB-19515

The refueling water purification pump has a seal water tank that provides the seal water and back pressure to the mechanical seals. The seal water tank is pressurized by demineralized water or station air. The pump capacity is nominally 85 gpm at 125-130 psig. The pump **START/STOP** pushbuttons (PB-19515) are mounted locally near the pump. Red (**ON**) and green (**OFF**) indicating lights are located with the pushbuttons.

3.9 SFP Filters A and B (Figure KNP-SFP01)

Two half capacity (50%) filters are provided in the SFP System. The design features of the filters include a single unit element sized to fit in a 55 gallon drum for disposal. The direction of flow in the filter elements is from the inside to out. This direction of flow collects the dirt and radioactive contaminants inside the element prior to reaching the outer surface of the filter. Other design features include conical head type bolts to fasten down the removable filter heads. Use of this type of head bolt allows the heads to be removed without special tools. Also, the filter vent is placed on the side of the filter housing, allowing the filter element to be removed without disconnecting any piping.

SFP Filter A/B Data	
Manufacturer	AMF Cuno Division
Technical Manual	K-214-2, -5
Model Number	GR-01 DB1
Inlet Connection / Outlet Conn.	3" Sch 10S/ 3"
Filter Housing Material	304 SST
Filter Element Material	347 SST
Filter Particulate Rating	15 micron absolute
Design Flow Rate	425 gpm
Pressure Drop (DP) Clean/Dirty	10 psid/35 psid
Maximum DP	70 psid
Filter Head Gasket Type	"0" Ring
Gasket Material	Ethylene Propylene

3.10 SFP Heat Exchanger

The SFP heat exchanger is a horizontal U-tube type HX located in the SFP HX room.

SFP HX Data		
Manufacturer	Struthers Well Corp	
Technical Manual	K-140-4	
Type	U-12-SH (CEU)	
	SHELL SIDE	TUBE SIDE
Coolant circulated	SW	SFP Water (2,100 ppm)
Total Flow Rate	550 gpm	850 gpm
Inlet Temperature	66.0 ° F	120° F
Outlet Temperature	96.0° F	100° F
Number of Passes	1	2
Pressure Drop	8.0 psi	8.0 psi
Fouling Drop	0.0010 psi	0.0005 psi
Material	Carbon Steel	304 SST

The SFP Heat Exchanger maintains the SFP below 140° F during normal fuel handling and storage conditions by means of control valve SW-1601. Valve SW-1601 controls the SW to the SFP heat exchanger and is located in the SFP Heat Exchanger Room. The SW-1601 temperature setpoint is manually adjustable from the SFP Hx Room.

3.11 SFP Demin Pre- and Post-Filters

Each filter is designed with features similar to the SFP filters with the exception that the maximum flow rate is 80 gpm.

SFP Demin Pre- and Post-Filters Data	
Manufacturer	AMF Cuno Division
Technical Manual	K-214-2, -5
Model Number	1DB1
Inlet Connection/Outlet Conn.	3" Sch 10S/3"
Filter Housing Material	304 SST
Filter Element Material	316 SST
Filter Particulate Rating	15 micron absolute
Design Flow Rate	80 gpm
Pressure Drop (DP) Clean/Dirty	10 psid/35 psid
Maximum DP	70 psid
Filter Head Gasket Type	"0" Ring
Gasket Material	Ethylene Propylene

3.12 SFP Demineralizer

The SFP demin is designed and supplied by the Westinghouse Electric Corporation. The demin is filled with nuclear grade resin and used in the SFP purification loop to remove fission products that are in the SFP coolant solution.

SFP Demineralizer Data		
Manufacturer	Edward E. Johnson	
Technical Manual	K-100-	
Drawing	541F109 X-K100-182	
Design Pressure/ Temperature	200 psig / 250° F	
Resin Volume for Flushed Beds	27 ft^3	
Maximum Flow Rate / Minimum Flow	72 gpm / 18 gpm	
Design Flow Rate	60 gpm	
Pressure Drop	**At 18 gpm**	**At 72 gpm**
Empty	0.5 psi	4.0 psi
With 27 ft^3	3.3 psi	15.5 psi

The SFP demin is located in the Auxiliary Building mezzanine, demineralizer room. The resin removes ionic impurities from the SFP water by absorbing the ions on the resin beads. The design flow through the SFP demin is regulated by Flow Control Valve FPC-204/CV-31293.

The Flow Control Valve FPC-204 is a self contained valve. The Flow Control Valve maintains a constant purification loop flow of 60 gpm even as system differential pressures (DPs) change during system operation.

3.13 Control Room Controls

The following SFP System controls are located on Mechanical Vertical Panel "A".

Red **(ON)**and green **(OFF)**lights are located with the control switches. One pump is **ON** and one pump is in **AUTO.** The **AUTO** SFP pump starts upon loss of the running pump.

Pump	Switch	Function
SFP Pump A	ES-46521 Control Switch	**OFF/AUTO/ON** Three Position Maintained
SFP Pump B	ES-46522 Control Switch	

3.14 Local Controls

Pump	Switch	Function
Refueling Water Purification Pump	PB-19515 Pushbuttons	**START/STOP**

Red **(START)** and green **(STOP)** indicating lights with the pushbuttons.

3.15 Local Indications

Indication	Range	Normal	Instrument
SFP Pump A Disch Press	0-200 psig		PI-11073
SFP Pump B Disch Press	0-200 psig		PI-11074
SFP Filter A DP	0-100 psid		DPI-11305
SFP Filter B DP	0-100 psid		DPI-11306
SFP Demin Pre-Filter DP	0-100 psid		DPI-11030
SFP Demin Post-Filter DP	0-100 psid		DPI-11015
SFP Demin DP	0- 30 psid		DPI-11014
SFP Heat Exchanger DP	0- 30 psid		DPI-11055
SFP Supply (Return Hdr) Temp	32-200° F		TI-12088
SFP A Pool Temp	0-220° F		TI-12007
SFP B Pool Temp	0-220° F		TI-12012
Refuel Wtr Purf Pump Press	0-200 psig		PI-11104

3.16 SFP System Alarms

NOTE: See Alarm Response Sheets in Control Room for Setpoint specifics.

Annunciator No.	Window Label	Setpoints
47055N	SPENT FUEL POOL ABNORMAL	SFP A/B Level High SFP Hx Primary to Secondary DP SFP A/B Temperature High SFP A/B Level Low

3.17 Sequence of Events Recorder (SER)

NOTE: See Alarm Response Sheets in Control Room for Setpoint specifics.

SER No.	Printout
155	Spent Fuel Pool A LEVEL HIGH
156	Spent Fuel Pool B LEVEL HIGH
157	Spent Fuel Pool A TEMP HIGH
158	Spent Fuel Pool B TEMP HIGH
159	Spent Fuel Pool A LEVEL LOW
160	Spent Fuel Pool B LEVEL LOW
161	Spent Fuel Pool Heat Exchanger Primary to Secondary DP Low

3.18 System Interrelationships

SW Supplies - The Service Water (SW) System supplies the cooling medium to the SFP heat exchanger and is the emergency makeup to the SFPs.

Demin Supplies - The Demineralized Water (DW) System supplies a back pressure for the Refueling Water Purification pump seals and can be used as makeup to the SFPs.

RWST - The RWST is also considered a part of the Emergency Core Cooling Systems (ECCS) including the Safety Injection (SI) System.

RHR HX - The RHR heat exchanger A can be used as an alternate heat exchanger if the SFP heat exchanger fails.

480 Vac Supplies - The 480 VAC Electrical Distribution System supplies power to the SFP pumps and the RWST purification pump.

Station Air - Station Air is used for the following: to inflate the SFP gate seals (with nitrogen as a backup), to pressurize the seal water tank, and for the RWST purification pump.

Misc Sumps - The SFP Leak detection and collection trough [illegible] discharge to the Miscellaneous Sumps and Drains System.

4.0 Precautions and Limitations

WR&RWP - The replacement of SFP demineralizer resin requires a Work Request and a Radiation Work Permit or Health Physics coverage.

Spilled Resin -All spilled resin shall be cleaned up and **NOT FLUSHED DOWN FLOOR DRAINS.** Ensure proper resin and volume is available for replacement.

High Rad Level - All personnel should be made aware of high radiation levels on the flush piping from the affected bed to the spent resin tank.

5.0 References

1.0 Summary	N-SFP-21 A-SFP-21	E-SFP-21-CL E-SFP-21	DCR 2946 DCR 2500
2.0 Functions			
3.0 Description Design	OPER M-218 E-1617, E-1636 FSAR 9.3	Annunciators 47004 P.O.'s K-233, K-141, K-214, K-140	
	For complete listing of DCR's associated with System 21, see: \Programs\KNPP Applications\Physical Changes\DCR Database\Sys 21		
4.0 Precautions & Limitations	Annunciators 47022, 47023, 47030, 47032		
5.0 References	KNPP Technical Specifications		
6.0 Procedures	For complete listing of proccedures associated with System 21, see: **H:\Procedures Issued\Proc Index\Sys #21**		
7.0 Appendices	See attached Figures		

5.1 Technical Specifications

NOTE: Refer to a Controlled Copy of the KNPP Technical Specifications for plant operations.

TS 3.8.a.2 During refueling operations, radiation levels in fuel handling areas, the Containment, and the spent fuel storage pool shall be monitored continuously.

TS 3.8.a.7 Heavy loads, are not to be transported over or placed in either SFP pool when spent fuel is stored in that pool.

TS 3.8.a.9.A The SFP Sweep System, shall be operating during fuel handling and when any load is carried over the pool if irradiated fuel in the pool has decayed less than 30 days

TS 5.4.c The SFP is filled with borated water at a concentration matching that in the Reactor Refueling Cavity and Refueling Canal during refueling operations or whenever fuel is in the pool.

TS 3.8.a.3 The Reactor will be subcritical for 148 hours prior to movement of its irradiated fuel assemblies.

6.0 Procedures

NOTE: For complete listing of proccedures associated with System 21, see:
H:\Procedures Issued\Proc Index\Sys #21

See System Description 32D for details on Solid Radioactive Waste System (RWS).

7.0 Appendices

7.1 Attached Figures

Figure No.	Title	KNPP DWG
KNP-SFP01	Spent Fuel Pool Cooling & Cleanup	OPER M218
KNP-SFP02	SFP Leak Detection System	
KNP-SFP03	SFP Cooling & Cleanup Logic	E-1617

NOTE: The "Figures" (drawings) previously associated with the System Descriptions will **not** be revised and updated at this time. Instead (obsolete) copies of these Figures have been stamped HISTORICAL and temporarily attached.

NMC KNPP System Description	System No. 45	Rev. 1
Title Radiation Monitoring System (RM)	Date 07/08/02	Page 1 of 27
Author ___ Eric E. Streich	Op's Review ___ David D. Mielke	
Tech Review ___ David D. Mielke	Approved ___ Manager of Engineering	

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INDEX

1.0 Summary

1.1 Overview (See Figures KNP-RM01, RM02, and RM03)

The Radiation Monitoring (RM) System provides continuous radiological surveillance of critical plant systems and work areas. The RM System performs the following basic functions:

- Warns operating personnel of radiological health hazards which have developed.
- Gives early warning of certain plant malfunctions which might lead to a radiological health hazard or plant damage.
- Prevents or minimizes the effects of inadvertent release of radioactivity to the environment by consequence-limiting automatic responses.
- Provides routine monitoring of controlled offsite plant releases.

The RM System is divided into the following subsystems:

- Fluid Process RM Subsystem, which monitors various process streams (liquid and gas) for indication of radiation levels within those streams. This subsystem includes additional monitors mounted on the outside of process piping and/or off-line sampling for greater reliability.
- Area RM Subsystem, which monitors radiation levels in various areas of the plant. This subsystem includes additional channels primarily for local and Control Room indication and alarm.
- Auxiliary (Aux.) Area RM Subsystem, which monitors radiation levels in the Auxiliary and Reactor Buildings and the Technical Support Center (TSC). This is a post-TMI addition to the plant (Eberline and General Atomic Systems).

Process radiation monitors which monitor "managed releases" to the environment also provide continuous records of actual releases to the environment.

The RM System consists of the following channels:

- Process Radiation Monitors: R-11 through R-23, R-31 through R-38, SPING-01 and SPING-02,
- Area Radiation Monitors: R-1 through R-10, (R-3 and R-8 were deleted) R-25 through R-30, R-39,
- Aux Area RM Subsystem: Instruments No. 29041 through 29046, 29048 through 29053, and 29055 through 29065, including two wide-range Containment monitors (R-40 and R-41). In addition, the Aux Area RM includes 5 Beta Air Monitors, Instruments No. 29066 through 29070.

The Nuclear Research Channels (R-2 through R-23) are configured with a field detector, which feeds a pre-amp, which provides local indication, which sends a signal to the Control Room. In the Control Room, the signal provides drawer indication recorder and PPCS trending, annunciator alarms and if necessary, automatic actuation's. Long term data retention functions are stored in the RADAC, located in the RAF.

The two process stack gas monitors (SPING-4) each contain three extended range noble gas channels and normal range particulate, iodine-131 and background channels. The DAMs and SPING-4 retain the data from each detector channel in local history files via a microcomputer. Each DAM in the Auxiliary RM System performs background subtraction, applies conversion factors and retains historical data. If AC power is lost to a DAM, battery backup for up to 8 hours is provided to preserve historical and initializing data and to allow continued operation. The RadServ Terminals, located in the RPO and the RAF (Master) have bi-directional communication with the DAM's via an optically coupled data highway system. A failure of a DAM or any detector does not affect other portions of the RMS. Operator interface with the remainder of the system is provided at the RadServ Terminal.

The RadServ Terminals have the capability to connect to a central computer. High radiation alarms are displayed at the RadServ Terminal. Two recorders in the Control Room record data from the wide range Containment monitors. Normal operation is in the full automatic mode with data displayed on the RadServ Terminals in accordance with programmed requirements. The RadServ Terminals provide RMS status of the DAM's and Sping's. Channel parameters and histories are also provided by the RadServ Terminals.

Usually, an indicator-alarm station is mounted adjacent to an area monitor detector. In some cases, such as the New Fuel Pit Criticality Monitor R-10, the detector is located in an inaccessible location. For these channels the indicator-alarm assembly is installed in an accessible location as close as possible to the detector.

1.2 System Operation versus Plant Modes

Plant Mode	System Support
Startup Normal Power Shutdown Refueling	The RM System safety related equipment (see Section 3.11, - Interlocks) is required to be operable during a reactor startup and plant operations. The RM System must be operable to provide monitoring and interlocks to prevent the inadvertent release of radioactivity to the environment.
Casualty Events	See Section 1.7 for RM System operation during casualty operations.
Infrequent Operations	Not Applicable
Maintenance	Maintenance can be performed on the RM System while operating at Power. The Control Room Operators must refer to a Controlled Copy of KNPP Technical Specifications to ensure that the Limiting Conditions for Operation (LCO's) and Radiological Effluent Technical Specifications (RET's) are met.

1.3 Startup (OP's# N-RM-45, N-RM-45-CL)

The Radiation Monitoring (RM) System is normally maintained in an operational status during all plant modes. The RM System requires no special procedures for startup other than the normal operation checks. See Operations Procedures N-RM-45 and N-RM-45-CL for specific lineups.

1.4 Normal Operation (OP# N-RM-45)

During normal plant operation, the RM System is maintained fully operational. The daily check and monthly test procedures referenced in the TS assure the necessary RM System surveillance. See also the Surveillance Procedures associated with System 45. Channels that provide an automatic control function, fail in a manner that duplicates the high level alarm condition.

1.5 Shutdown

Intentional shutdown of the entire system is not contemplated during the plant life. Shutdown or failures of individual channels are expected to occur, however. Sometimes removing a channel from service activates that channel alarm, such as R-13 activates the Zone SV. Calibrations are normally performed on all channels at regular intervals in accordance with the associated Surveillance Procedure (SP) or Instrument & Control Procedures (ICPs).

1.6 Special Operations

Since the RM System is in continuous operation, there are no special operational modes.

Effluent discharges should **not** be conducted if the applicable RM is out of service without further evaluation. Containment Vessel Air Particulate Monitor R-11 and Radio-Gas Monitor R-12 channels are normally set for monitoring the Containment atmosphere. The vent position is used for monitoring releases to atmosphere in accordance with Procedure N-RBV-18B. For additional requirements, see System Description 18, entitled, "Reactor Building Ventilation System".

1.7 Abnormal (OP# A-RM-45)

Abnormal Procedure A-RM-45 describes the actions to be taken when an abnormal condition exists in the RM System. Critical channels of the RM System are designed to provide automatic consequence limiting action in the event of plant system or channel malfunction. These actions require no immediate Operator intervention.

A loss of coolant accident (LOCA) would render a large part of the RM System inoperable or unreliable due to excessive radiation levels or pressure and temperature excursions (R-2, R-7, R-11, R-12, R-21). It may be necessary to maintain the functional status of the following channels for the post-accident period:

- Channel R-16 ..... Containment Fan Coil Unit (CFCU) Water Monitor
- Channel R-22 ..... Residual Heat Removal (RHR) Pump Pit Monitor
- Instrument 29064 .. Containment Hi Level RM 1B
- Instrument 29065 .. Containment Hi Level RM 1A

Abnormal Procedure A-RM-45 provides the symptoms/annunciators, automatic actions that occur, and Operator actions necessary to minimize the consequences of a high radiation level or high radioactivity.

2.0 Functions

The purpose of the Radiation Monitoring (RM) System is to provide continuous radiological surveillance of critical plant systems and work areas.

The specific functions of the RM System are as follows:

- Warns operating personnel of radiological health hazards, such as abnormal radiation fields.
- Provides warning of plant malfunctions which could lead to plant damage and/or radiological hazards.
- Prevents or minimizes inadvertent releases of radioactivity to the environment via automatic action capability.
- Provides monitoring of controlled radiological plant releases.

3.0 Design Description

3.1 System Design Requirements

The Radiation Monitoring (RM) System is designed to provide reliable information required to meet the four functions specified in the Overview, Section 1.1. The ranges and sensitivities of the individual channels are consistent with meeting the requirements of Title 10, Code of Federal Regulations (CFR), Parts 10 CFR 20, 10 CFR 50, and 10 CFR 70. The channel electronics are of the fail-safe design such that channel failures cause an alarm. For channels R-1 through R-23, channel failures are alarmed and indicated at the corresponding IAT Unit in the RMS Panels, and [illegible] alarmed on the plant annunciator panel located in the Control Room.

The Aux. Area RM System was designed to meet post-TMI Nuclear Regulatory Commission (NRC) requirements for monitoring radiation levels in the Aux. Bldg. and the Reactor Containment Bldg. and the TSC.

Specific locations for area monitors were chosen with the following guidelines:

- An area monitor should be located in any area where personnel perform regular duties, once per day or more frequently. A rule of thumb with area monitors is each area monitor can effectively survey a 40' line of sight radius.
- An area monitor should be located in any area where personnel perform infrequent duties, but where a significant probability exists for hazardous dose rates.
- An area monitor may be located in an area where personnel perform infrequent duties and where there is an insignificant probability of hazardous dose rates, but where continuous, low maintenance surveillance is necessary.

Main radiation monitoring channel circuitry is of the non-saturating design that precludes falloff in indication in high radiation fields. Noninterruptible power is provided from inverters operating from the 125 VDC power supply. Redundant safeguard channels are isolated electrically and physically in accordance with IEEE Standard 279. Redundant safeguard channels have independent power sources from the safeguards buses.

Each RM System channel has been subjected to a minimum 100 hour electrical burn-in period by the supplier. Process monitor detectors are located in the process stream such that the detectors are not subject to temperatures in excess of maximum specified detector operating temperatures.

The process monitor channels are designed to provide information on the following:

- Radioactivity concentration levels present in the various systems.
- Leakage across closed system boundaries.
- Radioactivity concentrations in liquid and gaseous effluent paths that lead to release from the plant site.

The ion chamber detectors in the Aux Area RM System are seismically qualified as Class 1E equipment following 1.0E+6 R exposure. All electronic channels are seismically qualified as Class 1E equipment. The wide range Containment monitors (R-40, R-41) include redundant channels whose signals are fed to two recorders in the Control Room. These monitors are Safety Grade 1E and qualified under loss-of-coolant accident (LOCA) conditions. These detectors, which are required post-LOCA, have a rated integrated lifetime dose of greater than 1.0E+9 Rads.

Seven radiation monitoring channels, R-11 thru R-15, R-19 and R-21, actuate or initiate automatic sequences of engineered safety-related equipment. Four channels, R-16, R-22, and Instruments No. 29064 and 29065, are designed to operate in the Containment post-accident environment. R-16 and R-22 are located in the Auxiliary Building (Aux. Bldg). Instruments No. 29064 and 29065 are located inside the Reactor Containment Bldg.

Safety Related RM Channels			
Channel	**Number**	**Instrument**	**ESF Associated action**
Containment Vent Particulate Monitor	R-11	29020	Isolates Containment Vent (CVI) See also Section 4.4
Containment Vent Gaseous Monitor	R-12	29021	Isolates Containment Vent (CVI) See also Section 4.4
Aux Bldg Vent Monitor	R-13	29010	Isolates normal vent exhaust and shifts to HEPA filters
Aux Bldg Vent Monitor	R-14	29007	Isolates normal vent exhaust and shifts to HEPA filters
Containment Sys. Vent Activity Monitor	R-21	29025	Isolates Containment Vent (CVI) See Section 4.4

Each RM channel consists basically of a radiation detector, an amplifier with associated power supplies, indicators, and computer-meter outputs. The three basic detectors used in the RM Systems are:

- scintillation detectors,
- Geiger-Mueller (GM) tubes, and
- Ion chambers.

The scintillation detectors are plug-in units containing either a thallium activated sodium iodide (NaI) crystal optically coupled to a photo-multiplier tube, or a beta radiation sensitive plastic scintillator. The GM tubes are standard beta-gamma GM radiation counter tubes operating on the secondary ionization current principle. The ion chambers are standard ionization chambers which measure current flow produced by ions between two electrodes within a gas-filled enclosure. The ion chambers are used for fuel handling areas and potential criticality areas (R-5, R-10).

When exposed to a radiation field, the detector generates a small signal that is amplified by an integral preamplifier. This signal is fed into shaping and conversion circuitry and a final amplifier stage. For channels R-1 through R-10, a 2-second moving average filter is used for channels R-11 through R-23 a 30-second moving average filter is used to reduce the variation in the channel reading. The signal is then transmitted to the readout device. On high radiation alarm, a channel may be required to initiate a mechanical function such as valve closure, damper isolation, etc.

All channels are capable of operational verification through the use of radioactive check sources and electronic test signals.

The R-1 through R-23 channel Indicator-Amplifier-Trip Assemblies (IATs) with alarm indication are located in the main Control Room on Mechanical Vertical Panel "C". Alarms are annunciated when an Alert or high radiation condition exists or when the channel becomes deenergized or when a Check-source is actuated. Where applicable, alarms are provided to warn if a monitor sampling system malfunctions.

The sensitivities, ranges, temperature requirements, set points, and other design parameters are delineated for most channels on Logic Diagram E-2021.

In the Auxiliary Area RM System, the radiation detectors provide data to six field mounted data acquisition modules (DAMs), and to two RadServ Terminals. One terminal is located in the Radiation Protection Office (RPO) and the second terminal is in the Radiological Analysis Facility (RAF) in the Technical Support Center (TSC).

3.2 Process Radiation Detectors - Liquid

NOTE: Reference KNPP Drawing E-2021, entitled "Radiation Monitoring-Kewaunee -System 45" for current Channel specifics, such as descriptions, ranges, setpoints, meter scales, detector types, and additional details.

The Liquid Process RM System consists of 15 channels to monitor various process systems. Ten of the 15 channels are equipped with some level of automatic action upon receipt of a high radiation alarm. Two of the 15 channels R-19 and R-21 actuate or cause automatic sequencing of ESF equipment. Two channels, R-16 and R-22, are designed to operate in a post-LOCA process environment.

Liquid Process RM Detectors				
Channel Name	Number	Instrument	Vendor/Mfgr	Type
Containment Fan Coil Unit (CFCU) Water Monitor	R-16	29024	Nuclear Research Company	NaI / Scint.
Component Cooling Water (CC) System Liquid Monitor	R-17	29001		
Waste Disposal (WD) System Liquid Effluent Monitor	R-18	29008		
Steam Generator Blowdown (SGB) Liquid Sample Monitor.	R-19	29005		
Deleted per DCR 2172				
Service Water (SW) Monitor	R-20	29032		
Deleted per DCR 2172				

Auxiliary Process Monitors				
Steam Line 1A Low Range	R-31	29080	Eberline	GM Tube
Steam Line 1A High Range	R-32	29079		Ion chamber
Steam Line 1B Low Range	R-33	29082		GM Tube
Steam Line 1B High Range	R-34	29081		Ion chamber
Aux Bldg Vent Low Range	R-35	29078		GM Tube
Aux Bldg Vent High Range	R-36	29077		Ion chamber
Containment Bldg Low Range	R-37	29076		GM Tube
Containment Bldg High Range	R-38	29075		Ion chamber

All channels of the Liquid Process RM System use a thallium activated, sodium iodide crystal (NaI) as the radiation detection element and are sensitive to only gamma radiation. The NaI detector is optically coupled to a photomultiplier tube and is housed in a hermetically sealed container. A preamplifier and Control Room operated check source are included in the detector package. (See Figure KNP-RM04.)

The detectors are mounted either directly in the process piping or offline in a parallel flow path, depending on the specific channel. Sufficient lead shielding is provided around the detector to attain the required sensitivity for all detectors during normal plant operation. For those monitors required to operate following a LOCA (R-16, R-22, and Instruments 29064 and 29065), higher temperature operating requirements or additional shielding are provided to achieve the specified sensitivity.

The R-31 through R-38 high/low range monitors are actually area type monitors which have been installed on the outside of process flow piping. A pair of monitors, one high range and one low range, are located on the outside of the Containment vent ducting, the Aux Bldg Vent ducting, main steam piping 1A and 1B. Each high/low range detector has a local readout and a Control Room readout on multipoint strip chart recorders located on the Radiation Monitoring Control Console. The R-31 through R-38 detectors are all Eberline GM Tubes for low range and ion chambers for high range.

3.2.1 Containment Fan Coil Unit (CFCU) Monitor R-16 (See Figure KNP-RM04)

This channel monitors the Containment fan coil cooling water (SW) for radiation following a LOCA indicating a leak from the Containment atmosphere into the SW cooling. A sample connection is located on the 10" SW discharge header for the Containment Fan Coil Units (CFCUs) A and B. The C and D CFCUs have a similar connection. Sample lines from these connections are run into a common detector header to provide a representative sample and are monitored by a single NaI scintillation detector mounted in an in-line type sample volume chamber. Upon indication of a high radiation level, each CFCU is individually taken out of service to determine which unit is leaking. This procedure is achieved inside the Control Room by manually selecting the desired CFCU to be tested using SW Isolation Valves SW-903A/MV-32060, SW-903B/MV-32061, SW-903C/MV-32058, or SW-903D/MV-32059 as applicable.

3.2.2 Component Cooling Water (CC) System Monitor R-17

Note: See KNPP Drawing E-2021 for specifics to this channel.

This channel continuously monitors the CC System for radiation indicative of a reactor coolant leak into the CC System. A leak can occur into the CC System from the Reactor Coolant Pump (RXCP) thermal barriers, the Chemical and Volume Control System (CVCS) letdown heat exchanger, or from the Residual Heat Removal (RHR) heat exchangers. An in-line T-type monitor is located in the CC pumps 16" discharge header downstream of CC heat exchangers A and B. This channel, R-17, uses a NaI scintillation counter. An alert and high radiation level alarms actuates Control Room alarms.

3.2.3 Waste Disposal (WD) System Liquid Effluent Monitor R-18

Note: See KNPP Drawing E-2021 for specifics to this channel.

This channel continuously monitors all Waste Disposal System liquid releases from the plant. Monitor R-18 has an in-line detector assembly which is inserted in the 3-way 2" discharge header near Waste Condensate Pumps 1A and 1B. Automatic closure of air operated Valve WD-19/CV-31138 is initiated by this monitor to prevent further release after a high radiation level is indicated. An NaI scintillation counter and sample volume chamber assembly monitors these effluent discharges. Remote indication and annunciation are provided on the Waste Disposal System Panel and in the Control Room.

3.2.4 SG Blowdown Sample Monitor R-19 (See Figure KNP-RM04)

Note: See KNPP Drawing E-2021 for specifics to this channel.

This channels monitor the liquid phase of the secondary side of the SGs for radiation. This indication could be used to indicate a possible primary to secondary Reactor Coolant System (RCS) leak, thus providing backup capability for the condenser air ejector gas monitor and the steam line N-16 monitors (R-42 & R-43). Sample connections are taken off the 3/8" sample lines from SG's A and B after the primary and secondary coolers. SG samples are continuously monitored by the NaI scintillation counters and a sample volume chamber

Each of the two 3/8" detector sample lines has a manual flow control valve, (BT-35A-1 and BT-35B-1), a check valve, (BT-35A and BT-35B), and rotameter upstream of the common detector header. It is possible to flush the sample volume chamber to the SGBT holdup tanks using demineralized water. See System Descriptions 7A/7B for more details on the SGB System operation.

Upon indication of a high radiation level on R-19 or a high radiation level on Condenser Air Ejector Gas Monitor R-15, the motor operated Blowdown Isolation Valves BT-2A/MV-32077 and BT-3A/MV-32078 for SG A and BT-2B/MV-32079 and BT-3B/MV-32080 for SG B are automatically closed. These motor operated isolation valves are located in the 2" SGB lines. Furthermore, air operated Control Valves BT-31B/CV-31270 and BT-32B/CV-31271, located in the 3/8" sample line for the SG B, are automatically closed. Similarly, the SG A air operated valves BT-31A/CV-31334 and BT-32A/CV-31335 are closed. R-19 serves as backup to R-15 to divert Turbine Building air ejector discharge vent valve AR-6/CV-31168 to the Aux Bldg exhaust duct. Refer to the System Description 09, entitled, "Air Removal System" for more details on AR-6 operation.

After SG isolation as discussed above, manual Valves BT-35A and BT-35B are closed locally and the normally closed gate Valves BT-36A and BT-36B in the two 3/8" sample room lines, branching off the 3/8" radiation detector sample lines, are opened. Following this isolation of the radiation monitor and the opening of these new sample lines to the sample room, the SG liquid sample lines are again opened using the previously mentioned Valves BT-31A, BT-32A, BT-31B, and BT-32B. The SGs can then be manually sampled in the sample room and the leaking SG can be identified.

Indication and annunciation of Channel R-19 is provided at the Secondary Sampling Analytical Panel. See System Description 29, entitled, "Secondary Sampling" for more details on SG sampling. R-19 also has indication and alarms in the Control Room.

3.2.5 Service Water (SW) Monitor R-20 (See Figure KNP-RM06)

Note: See KNPP Drawing E-2021 for specifics to this channel.

This channel continuously monitors the SW return path from the spent fuel pool and CC heat exchangers. A NaI scintillation counter and sample volume chamber assembly monitors these effluent discharges prior to discharge to the Aux Bldg CW 24" Standpipe. An increase above ambient radiation levels could indicate a leak across one of these heat exchangers and selective isolation could then be used to locate the malfunction.

One sample connection is located on the return line from the CC Heat Exchanger A and B. Another sample connection is located on the SW return header from the Spent Fuel Pool (SFP) Heat Exchanger. Sample lines from each of these connections have associated manually operated Flow Control Valves SW-1330A, SW-1330B, and SW-1640, respectively. An alarm and indication is provided locally at the detector and in the Control Room.

Deleted R-24 per DCR 2172

3.3 Process Radiation Detectors – Gaseous & Particulate

3.3.1 General

The main process gaseous monitoring system consists of several individual process subsystems:

1) Containment Vessel and Containment System Vent Monitoring Subsystem, R-11, R-12, and R-21.
2) Aux Bldg Vent Monitoring Subsystem, R-13, and R-14 with charcoal sample filters.
3) Condenser Air Ejector Monitoring Subsystem , R-15.
4) RHR Pit Exhaust Air Monitoring Subsystem, R-22.
5) Control Room Vent Monitoring Subsystem, R-23.
6) Steam Line N-16 monitors, R-42 and R-43.

The gaseous process monitors use either GM tubes or scintillation detectors using the NaI crystals, or beta sensitive plastic scintillators. The detector assemblies are similar to those units already discussed. Each channel assembly contains an integral preamplifier and a Control Room operated check source.

The following gaseous process monitors are provided:

Gaseous Process Monitors				
Channel Name	**Number**	**Instrument**	**Vendor/Mfgr**	**Type**
Containment Vessel or Cont. System Vent Air Monitor	R-11	29020 Particulate	Nuclear Research Company	NaI/Scint.
Containment Vessel or Cont. System Vent Air Monitor	R-12	29021 Gaseous		
Aux. Bldg. Vent Monitor A	R-13	29010 Gaseous		
Aux. Bldg. Vent Monitor B	R-14	29007 Gaseous		
Condenser Air Ejector Gas	R-15	29004		Scint./plastic
Contain. Sys Vent Activity Monitor	R-21	29025		Scint./plastic
RHR Pump Pit Monitor	R-22	29022		Scint./plastic
Control Room Vent Monitor	R-23	29006		Scint./plastic
Main Steam Line Monitor 1A	R-42	29113	MGP (Instruments)	Scintilation
Main Steam Line Monitor 1B	R-43	29114		Scintilation

NOTE: Some of the above monitors perform engineered safety-related functions. (R-11, 12, 13, 14, and 21).

3.3.2 Containment Vessel & Containment System Vent Monitoring Subsystem - R-11 and R-12 (See Figures KNP-RM02 and RM03)

Note: See KNPP Drawing E-2021 for specifics to this channel.

These monitors are provided to measure air particulate gamma radioactivity in the **Reactor Containment Building with the ability to monitor the Containment** System Vent. The latter application ensures that the release rate through the Containment System Vent during purging is maintained below specified limits. In the Containment sampling mode, the R-11 and R-12 channels provide qualitative information on incipient RCS leaks. This provides for personnel safeguard and early indication of equipment malfunctions.

High radiation level initiates closure of the Containment Purge Supply and Exhaust Duct Valves RBV-2/CV-31126, RBV-3/CV-31124, and TAV-12/ CD-34033, and RBV-5/CD-34006.

Channel R-11 takes a continuous air sample from either the Containment atmosphere via Valve AS-3, or the Containment System Vent through Valve AS-20. The channel is purged using the Purge Inlet Valve AS-10. The sample is drawn from the Containment or discharge ductwork through a closed, sealed system monitored by a scintillation counter and filter paper detector assembly. The filter paper collects particulate matter greater than 1 micron in size, on the constantly moving surface, and is viewed by a photomultiplier NaI scintillation crystal combination. The sample then passes through Gaseous Detector R-12 and exits the monitor. The sample is returned either to the Containment through AS-30 when AS-3 is open, or to the Aux. Bldg. Vent Stack through AS-40 when AS-10 or AS-20 is open.

An isokinetic sampling probe (See Figure KNP-RM08) is used in the case of the Containment System Vent to assure that a representative sample is drawn. The nominal maximum flow rate in the Containment System Vent is 33,000 cfm. The controls in the Sampling System are used to hold the sample flow rate constant at 2 SCFM. During times when the vent stream flow rate is lower, the isokinetic tip draws more than its proportionate share and produces a conservative sample.

The detector assembly is in a completely enclosed housing. The detector is a hermetically sealed photomultiplier tube and NaI scintillation crystal combination. The pulse signal is transmitted to the RMS Drawers in the Control Room. Lead shielding is provided to reduce the background level to where it does not interfere with the detectors sensitivity. The filter paper mechanism, an electro-mechanical assembly which controls the filter paper movement, is provided as an integral part of the detector unit.

The R-11 filter paper assembly contains a sprocket and chain mechanism with two speed capability. The operating speed is 1" per hour and the fast advance or paper clear speed is approximately 28" per minute. A paper speed selector switch is located on the Control Room RMS Control Panels. Fixed filter paper operation is also possible. The paper supply lasts a minimum of 25 days when used at normal operating speed of 1" per hour. The paper only lasts about 2.8 hours when kept on fast speed

The R-12 Monitor (See Figure KNP-RM04) is provided to measure gaseous gamma radioactivity in the Containment or Containment System Vent. When sampling the latter, it ensures that the radiation release rate during purging is maintained below specified limits. High radiation level initiates closure of the Containment Purge Supply and Exhaust Duct Valves RBV-2, RBV-3, RBV-5, and TAV-12.

A continuous air sample is taken from the Containment atmosphere or the Containment System vent, after the air passes through Particulate Monitor R-11, and draws the sample through a closed, sealed system to the off-line type gas monitor assembly. The sample is constantly mixed in a shielded volume chamber where it is monitored by a GM Tube. The air sample is then returned to the Containment or the Aux. Bldg. Vent System.

Detector Assembly R-12 is in a completely enclosed housing mounted in a constant volume gas container. The container is a welded steel tank designed for cyclonic flow around the detectors axis (4π Sample R) to preclude stagnation within the sensing volume. The detector is mounted exactly in the center of the tank volume, parallel to the length of the tank. Lead shielding is provided to reduce the background radiation level to a point where it does not interfere with the detectors sensitivity.

Containment Air Particulate R-11, and Radioactivity Gas Monitor R-12, have assemblies common to both channels. The flow control assembly includes a pump unit and selector valves that provide a representative sample or a "clean" sample to the detectors.

The sample pump unit consists of a pump to obtain the air sample, a flowcontroller to control and indicate the flow rate and to provide low and high flow alarms.

Selector Inlet Valves AS-3, AS-10, and AS-20 (See Figure KNP-RM03) are provided to direct the desired sample to the detector for monitoring and to block flow when the channel is in maintenance or purging condition. Valve AS-30 and AS-40 are the main sample and auxiliary sample/detector purge outlet valves. These inlet and outlet valves are automatically controlled by operating a sample selector switch on the R-11 monitor on the RM Control Console. This selector switch has three positions, **VENT/CNTMT/PURGE**. This switch aligns the Containment Vessel or the Containment System Vent sample, or "clean" purge loops automatically for monitoring. A pressure sensor is provided to protect the system from high pressure transients. This unit automatically closes the Containment Inlet and Outlet Valves, AS-3 and AS-30, and stops the sample pump upon a high pressure condition. Status lights on the RM Control Console indicate which inlet valve is open.

Control Room Selector Switch ES-46566 is used to select the sample discharge location for samples routed through the Main Sample Outlet Valve AS-30. Samples can be sent to the Aux. Bldg. Vent through AS-35/SV-33621 or to Containment through AS-31/SV-33622. Containment Air Sampling Inlet Isolation Valves AS-1/CV-31383 and AS-2/CV-31384 and Containment Air Sampling Discharge Valve AS-32/CV-31385 are controlled by Switches ES-46563, ES-46564, and ES-46565 respectively. These switches are three position, **CLOSE/AUTO/OPEN**, spring return to **AUTO** from the **CLOSE** or **OPEN** positions. These switches are located on the Control Room Mechanical Vertical Panel "B". Valve AS-1 is closed on a Train A Containment Isolation Signal and Valves AS-2 and AS-32 are closed on a Train B signal.

(See Figure KNP-RM03.) Also located on Monitor R-11 are the control switches for the sample pump, **OFF/RESET/LOCAL/ON**, and the filter speed selector – **OFF/OPERATE/FAST**. In addition, there is a local **START/STOP** switch for the sample pump on the 657' level of the Aux. Bldg.

Channel sampling system indicator lights for HI and LOW sample flow and HI pressure are located on the Monitor R-11 on the RMS Control Console. The form paper alarm is sensed from a microswitch on the paper path That is used to sense when the paper roll needs to be replaced or inspected.

3.3.3 Containment System Vent Activity Monitor R-21 (See Figure KNP-RM06)

Note: See KNPP Drawing E-2021 for specifics to this channel.

In addition to the flexibility of utilizing R-11 and R-12 for monitoring the Containment System Vent, another channel is provided, R-21. In order to meet KNPP Technical Specifications (TS) requirements of having two leak detection systems in service, one of which must be sensitive to radioactivity, manual Valves AS-5, AS-2-1, AS-50, AS-51, AS-34, and AS-30-1 are installed to allow R-21 to monitor Containment samples when R-11/R-12 is out of service. The R-21 channel consists of a fixed particulate filter, charcoal filter, a positive displacement sample pump, and off-line type radioactivity gas monitor mounted in series. This channel continuously monitors the filtered effluent in the Containment System Vent for activity. The sample pump is controlled by a local three position switch, OFF/RESET/LOCAL/ON . A red indicating light on the RM Control Console indicates when the sample pump is ON .

The gaseous detector is a beta sensitive plastic scintillation detector mounted in an off-line sampler. Essentially all the gamma emitting isotopes of interest also emits beta radiation. The use of the thin beta sensitive plastic scintillator reduces the shielding requirements around the detector while providing the necessary sensitivity and range. The sample tank design provides a cyclonic air flow around the detector axis to preclude stagnant air within the sensing volume. The detector output is transmitted to the RM System Drawer in the Control Room. An isokinetic nozzle (see Figure KNP-RM08) is provided for sample collection and is similar to that discussed for the R-11 monitor when sampling RBV.

The charcoal filter for this channel is for measuring the accumulation of iodine isotope activity. The particulate filter is for the collection of particulate activity such as Cesium-134 and 137. These filters additionally serve to protect the sensitive sample volume chamber of the radio-gas detector.

A high radiation signal from the R-21 monitor provides a closure signal to the Containment Purge Supply and Exhaust Valves RBV-1/CV-31125, RBV-4/CV-31123, RBV-5/CD-34006, and TAV-12/CD-34033.

3.3.4 Aux Bldg Vent Monitoring Subsystem

In order that the Control Room Operators be aware of the amount of activity being released from the plant, the Aux. Bldg. Vent Monitoring Subsystem is designed to perform the following:

- Monitor, indicate and record the radioactivity level of effluent gases discharged from the Aux. Bldg. Vent to the atmosphere.
- Alarm when the radiation level of the effluent gases being discharged exceeds a preset limit.
- Provide automatic initiation of the Aux. Bldg. Special Vent (ASV) System, the SFP Ventilation System - filtration mode only, and initiate isolation of all ducting to the Aux. Bldg. Vent with the exception of the ASV System

discharge and the SFP Vent System discharge.
- Provide automatic closure of the gas release valve from the gas decay tanks.

3.3.5 Aux Bldg Vent Monitors R-13 and R-14 (See Figure KNP-RM07)

Note: See KNPP Drawing E-2021 for specifics to this channel.

The Aux. Bldg. Vent Monitors detect radiation passing through the Aux. Bldg. Vent to the atmosphere. The R-13/R-14 monitors each consists of an off-line sampler. The gaseous detector is a beta sensitive plastic scintillation detector.

Remote indication and alarms are provided adjacent to the Waste Disposal System Panel. A high level alarm shuts down normal ventilation fans except the SFP Vent fans. A high level also activates the ASV System, activates the SFP Vent System in the filtration mode only, initiates isolation of all ducting to the Aux. Bldg. Vent, and closes the gas release valve from the gas decay tanks. If the signal for high radiation is from R-13, then the R-11/R-12 sample discharge is diverted to the Containment; AS-35/SV-33621 closes and AS-31/SV-33622 opens. The Gas Release Valve WG-36/CV-31215 is an unloading valve located in the 2" discharge header from the gas decay tanks. The 2" Containment Vent System isolates with LOCA-100B/CV-31725 closing and/or LOCA-2B/MV-32146 closing.

Aux Bldg Vent Sampler R-13A section deleted per DCR 2172.

3.3.6 Condenser Air Ejector Gas Monitor R-15

Note: See KNPP Drawing E-2021 for specifics to this channel.

R-15 channel monitors the discharge from the condenser air ejector exhaust header for gaseous radioactivity which is indicative of a SG tube leak or a primary to secondary leak. The noncondensible gases are drawn from the Condensers A and B by the steam jet air ejector and discharged to the Aux. Bldg. Vent for filtration prior to release. A 3-way control Valve AR-6/CV-31168 is located in the 4" condenser air ejector discharge header and is positioned to normally direct the discharge to the Aux. Bldg. Vent System. AR-6 can be positioned to discharge to the Turb. Bldg. during initial drawing of a vacuum. An R-15 or R-19 alarm, causes the AR-6 to be directed to the Aux. Bldg. Vent System rather than discharge to the Turb. Bldg. R-15 serves as a backup to isolate SG blowdown and blowdown sampling. A gamma scintilator detector is mounted in an in-line dry well and used to monitor gaseous radiation levels. Detector Assembly R-15 is positioned in the 4" condenser air ejector exhaust header. SG sample monitor R-19 serves as a backup to R-15 to position Valve AR-6 to the Aux. Bldg. Vent System and to isolate SG blowdown.

3.3.7 RHR Pump Pit Monitor R-22 (See Figures KNP-RM06 & RM07)

Note: See KNPP Drawing E-2021 for specifics to this channel.

Monitor R-22 provides continuous monitoring of the RHR Pump Pits exhaust air for indication of pump leakage which may contain high gaseous activity and high particulate activity.

Isokinetic nozzles are located directly in front of the discharge to RHR Fan Coil Coolers A and B. Each sample line has an associated loop flow control valve, flow meter, and hi/low flow alarm switches.

The flow control valves are controlled by a three position maintained, A/B/BOTH switch. Control Switch ES-46117 is located in the Control Room on Mechanical Vertical Panel "B". Red (**OPEN**) and green (**CLOSED**) indicating lights are located with the control switch. All of the R-22 equipment, along with the main flow control valve and sample pump, are contained in the RHR Monitor Panel. Local and Control Room control switches, flow alarms, and indicating lights are identical to those described for R-21 except there are two sample flow lines with different alarm setpoints for R-22. The radio-gas monitor is mounted across from the cabinet to provide space for additional shielding. Isokinetic nozzles are included for the possible future installation of a particulate monitor.

The R-22 gaseous detector is a beta sensitive plastic scintillation detector mounted in an off-line sampler. The sample tank design provides a cyclonic air flow around the detector axis to preclude stagnation within the sensing volume. The detector output is transmitted to the RM Drawer in the Control Room.

A high radiation alarm alerts the Control Room Operators so that a timely transfer to the standby RHR pump may be made.

3.3.8 Control Room Ventilation Monitor R-23

Note: See KNPP Drawing E-2021 for specifics to this channel.

Channel R-23 continuously monitors the Control Room ventilating air intake supply for indication of airborne activity in the Control Room Ventilation (ACC) System. The detector is a beta sensitive plastic scintillation detector that is mounted on the side of the 14,000 cfm air supply duct.

High alarm circuits actuate the post-accident recirculating fans and necessary dampers to isolate the Control Room environment. This places the ACC System (see System Description 25) in a recirculate mode and the air is recirculated through the post-accident recirculating filter system. Redundant air intakes are provided to allow for any makeup requirements. The air intakes are the Aux. Bldg. Air Conditioning (A/C) Intake and the Aux. Bldg. Vent System intake.

If a LOCA occurs, the background radiation field in the vicinity of the detector could be so high that Monitor R-23 would provide extremely unreliable information as to airborne activity levels in the Control Room.
In this case, post LOCA recirculation through the post-accident filter system is initiated on an Safety Injection (SI) signal or an Steam Exclusion Zone SV Area signal.

3.4 SPING-4 Off-Line Stack Gas Monitors

The SPING-4 monitors consist of a self contained microprocessor based radiation detection system used to monitor for particulate, iodine, and noble gas activity in the air. Two SPING-4 Process Monitors are installed off-line, one in the Containment Bldg. Vent Stack and one in the Aux. Bldg. Vent Stack. All SPING System detectors are tied to two central control consoles (CCCs) for monitoring. Each SPING-4 stack process monitor contains nine individual detectors. The two SPING-4 Process Monitors and associated nine individual detectors are as follows:

Channel 01 = Area Monitor "Aux. Bldg. Vent Duct SPING-4"

Channel 02 = Area Monitor "Containment Bldg Vent Duct SPING"

Address Code	Individual Detector	Address Code	Individual Detector
01-01	Beta particulate	02-01	Beta particulate
01-02	Alpha particulate	02-02	Alpha particulate
01-03	Iodine	02-03	Iodine
01-04	Iodine Background	02-04	Iodine Background
01-05	Lo-range gas	02-05	Lo-range gas
01-06	Gamma area monitor	02-06	Gamma area monitor
01-07	Mid-range gas	02-07	Mid-range gas
01-08	Gas Background	02-08	Gas Background
01-09	Hi-range gas	02-09	Hi-range gas

Information from the SPING-4 Process Monitors is available at either of the CCC's. Information from the SPING-4 monitors is also processed at the Radiation Protection Office and the Radiological Analysis Facility in the Technical Support Center (TSC).

3.5 General & Secondary Area Radiation Monitoring Subsystems

Note: See KNPP Drawing E-2021 for specifics to this channel.

All channels of the area radiation monitoring use GM tubes as the radiation detection element with the exception of R-5, R-10, and R-30 which uses ion chamber detectors. Each area radiation detector and integral preamplifier is packaged in a moisture proof assembly.

The following area radiation monitors are provided:

General & Secondary Area Radiation Monitor Detectors				
Channel Name	**Number**	**Instrument**	**Supplier Detector**	**Type**
Control Room Area Monitor	R-1	29011	Nuclear Research Corporation	GM tube
Containment Vessel Area	R-2	29012		
Radio-Chem Lab Monitor	R-3	29013		
Charging Pump Area Monitor	R-4	29033		
Fuel Handling Area Monitor	R-5	29015		Ion Chamber
Sampling Room Area Monitor	R-6	29016		GM tube
Incore Instrument Seal Table Area Monitor	R-7	29017		
Drumming & Waste Disposal	R-8	29018		
RCS Letdown Line Area	R-9	29019		

New Fuel Pit (Criticality)	R-10	29014		Ion Chamber
Radwaste Drumming Room Area	R-25	29034	Eberline	GM tube
SGBT Ion Exchanger Room	R-26	29037		
Spent Resin Storage Tank Rm	R-27	29038		
Compacted Radwaste Drum Drum Storage Area	R-28	29036	Eberline	GM tube
SGBT Holdup Tank Room	R-29	29035		
Containment Sump C Area	R-30	29039		Ion Chamber
Sludge Interceptor Area	R-39	29026		GM tube

The additional area monitors (R-25 through R-30 and R-39) were installed as a result of needs identified from experience. These channels provide only local indication. Channel R-30 has local alarm capability. The local alarm prevents inadvertent entry into a High Radiation Area when the incore thimbles are withdrawn for refueling or an incore detector is not in the proper storage location.

The main area radiation monitors are designed to measure gamma activity in the surrounding air over the range of 0.1 mR/hr to 100 R/hr for Channels R-1 through R-10.

The detector output is amplified and the log count rate is determined by the integral amplifier at the detector. The radiation level is indicated locally near the detector and in the Control Room. High radiation alarms are displayed in the Control Room and near the detector location.

A remotely operated, long half-life radiation check source is provided with Channels R-1 through R-23. The source strength is sufficient to produce up scale deflection.

The New Fuel Pit (Criticality) Monitor R-10 is a specialty device which actuates an audible evacuation alarm [illegible] in the New Fuel Pit Area. This monitor meets the requirements of 10 CFR 70.

3.6 Auxiliary Area Radiation Monitors (See Figure KNP-RM05)

Note: See KNPP Drawing E-2021 for specifics to this channel.

The original equipment suppliers for the Auxiliary Area Radiation Monitors were Eberline and General Atomic. This system was installed as a result of the lessons learned after TMI accident and because of increased NRC requirements. The Aux. Area Radiation Monitoring consists of 29 channels. The detectors are mostly ion chambers and 6 energy compensated GM tubes. Local readout is available at field mounted data acquisition modules (DAMs) and the two associated CCCs. An additional five portable, cart-mounted beta air monitors can be tied into the system and can be read out at either CCC. High radiation alarms are displayed at the two CCCs. Historical data is stored at the DAMs via a microcomputer. Two recorders in the Control Room are provided for the wide range Containment monitors.

The Aux. Area Radiation Monitoring Subsystem Detectors provide data to the local DAM's. Data is then passed on to the two CCC's. The CCC's have two way communications with the DAM's via an optically coupled data highway system. A failure of a DAM or any detector does not affect other portions of the system. Each CCC has its own keyboard, printer, and system status annunciator.

The following table lists the DAM locations:

DAM	Location
1	Maintenance Stairwell, El. 586', near MCC 1-46E
2	High Radiation Sample Room (HRSR)
3	Maintenance Shop, El. 605'
4	Loading Dock
5	Stairwell, El. 642', near SPING-4
6	TSC Electrical Equipment Room

The Aux. Area Monitors include the following monitors and areas:

Channel No. SPING	Channel Name	Inst. N0.	Associated DAM
03-01	Waste Disposal Panel Area	29043	2
03-02	Post-Accident Sample Room	29044	
03-03	CC Heat Exchanger Area	29051	
03-04	Beta Air Monitor – HRSR	29067	
03-05	Beta Air Monitor – HRSR	29067	
03-06	Beta Air Monitor Aux. Bldg.	29069	
03-07	Beta Air - El, 586'	29069	
03-08	Beta Air Monitor Aux. Bldg.	29070	
03-09	Beta Air - El, 606'	29070	
03-10	Containment Hi Level Rad B	29064	
04-01	Machine Shop	29049	3
04-02	Monitor Room Area	29050	
04-03	Makeup Demineralizer	29053	
04-04	Cold Chem Lab Area	29055	
04-05	Beta Air Monitor Aux	29068	
04-06	Bldg. El, 606'	29068	
05-01	RHR Pump Pit Area	29045	4
05-02	Radwaste Compactor(GM tube)	29046	
05-03	Aux. Bldg. Loading Dock	29052	
06-01	I & C Shop Area (GM tube)	29056	5
06-02	Shield Bldg. Vent Filter	29057	
06-03	Control Room A/C Vent	29058	
06-04	Containment Vent Exhaust Filter	29059	
06-05	Zone SV Exhaust Filter	29060	
06-10	Containment Hi Radiation A	29065	
07-01	RAF Count Room – TSC(GM Tube)	29062	6
07-02	Tech Support Center(GM Tube)	29063	
07-03	TSC Stairwell (GM Tube)	29061	
07-04	Beta Air Monitor TSC	29066	
07-05	Beta Air TSC El, 586'	29066	
08-01	Sulfuric Acid Storage Tank (GM Tube)	29041	1
08-02	Containment Spray Pump	29042	
08-03	Heating Steam Boiler	29048	

3.7 Tritium Sampling

3.7.1 Waste Gas Tanks Tritium Sampler

This sample unit collects a sample of water vapor from the waste gas tanks. The sample of water is then taken to the lab to be analyzed for tritium. The sample unit consists of a silica gel sample column to collect the water vapor, a sample pump, and a flow indicator. Flow Indicator FI-18032 measures the air flow through the silica gel column. The sample unit is mounted next to the Automatic Gas Analyzer. The sample unit draws a sample from the Waste Gas Tanks sample line and discharges to the Automatic Gas Analyzer discharge line.

3.7.2 Containment Vessel Tritium Sampler (See Figure KNP-RM02)

This sample unit collects a sample of water vapor from the Containment Vessel. The sample of water is taken to the lab for analysis. The Containment Vessel Tritium Sampler unit consists of a silica gel sample column to collect water vapor, a sample pump, and a flow indicator. Flow Indicator FI-18033 measures the air flow through the silica gel column. This sample unit is mounted on top of the Radiation Monitors R-11 and R-12 and uses the same Containment penetration piping.

3.7.3 Containment System Vent Tritium Sampler (See Figure KNP-RM02)

This sample unit collects a sample of water vapor from the Containment System Vent discharge. The sample of water is taken to the lab for analysis. The Containment System Vent Tritium Sampler unit consists of a silica gel sample column to collect water vapor, a sample pump, and a flow indicator. Flow Indicator FI-18034 measures the air flow through the silica gel column. This sample unit is mounted on top of Radiation Monitor R-21 and draws a sample from the vent discharge line and discharges back into the same line.

3.7.4 Aux Bldg Vent Tritium Sampler

This sample unit collects a sample of water vapor from the Aux. Bldg. Vent discharge. The sample of water is taken to the lab for analysis. The Aux. Bldg. Vent Tritium Sampler unit consists of a silica gel sample column to collect water vapor, a sample pump, and a flow indicator. Flow Indicator FI-18035 measures the air flow through the silica gel column. This sample unit is mounted on top of Radiation Monitor R-13A and draws a sample from the vent discharge line and discharges back into the same line.

3.8 RM System Panels - Mechanical Vertical Panel "C" (See Figure KNP-RM01)

The RM System Panels consist of three Control Consoles:

- CR-116 is the Nuclear Research Corporation process rack (containing R-11, 12, 14,15,16, 17, 18, and 19) and two multipoint recorders for monitoring R-31 through R-38.
- CR-117 is the Nuclear Research Corporation area RMS rack (containing R-1, 2, 4, 5, 6, 7, 9, and 10) one multipoint recorder, and two Containment High Range recorders.
- CR-118 is the Nuclear Research Corporation process rack (containing R-13, 20, 21, 22, and 23).

The four-channel strip chart recorder located in CR-117 can trend RMS channels R-1 through R-23.

Additional Process Monitors R-31 through R-38 are connected to the paperless recorders 81034 and 81035 and their associated ethernet connection Real Time Backbone (RTB) (Reference KNPP Drawing E-3826. (See Figure KNP-RM01.)

Each Indicator-Amplifier-Trip (IAT) unit contains the following:

- Level amplifier
- Log level amplifier
- Power supplies
- Test-calibration circuitry
- Analog radiation level meter indication
- Indicating lights for high radiation alarm, alert radiation alarm, channel failure, normal operation, process sample flow failure, etc.
- Bistable circuitry for alarms
- Check source circuitry

The level amplifier amplifies and discriminates the radiation detector output pulses to provide a discriminated and shaped pulse output to the log level amplifier.

The log level amplifier accepts the shaped pulse of the level amplifier output. The log level amplifier performs a log integration to convert the total pulse rate to a logarithmic analog signal. The signal is then amplified and the resulting output is provided for indication and recording.

Individual power supplies are contained in each drawer/unit for furnishing positive and negative voltages for transistor circuits. The power supply also supplies the detector high voltage and relay and alarm lights.

Each IAT has test and calibration circuitry. These circuits provide precalibrated pulsed and/or analog signals to perform channel tests. A solenoid check source is available for checking and verifying channel operation. The check source energy emissions are similar to the energies being monitored. The check source strength is sufficient to cause approximately mid-range indication on the indicator.

A radiation level meter (see Figure KNP-RM09) is calibrated logarithmically in counts per minute (cpm). Indicating lights on the IAT unit indicate High Radiation Level alarms and Loss-of-signal. A common annunciator is actuated on high radiation levels.

3.9 Alarms

NOTE: See Alarm Respnse Sheets in Control Room for actual Setpoints

Annunciator	Window
47011-B	RADIATION INDICATION HIGH
47012-B	RAD MONITOR [illegible] FAILURE
47013-A	RAD MONITOR SAMPLING FLOW HIGH/LOW
47014-B	RAD MONITOR CHECK SOURCE ACTIVATED
47015-B	RAD MONITOR POWER SUPPLY FAILURE
47013-A	RAD MONITOR SAMPLING FLOW HIGH/LOW

3.10 Sequence of Events Recorder (SER)

NOTE: See Alarm Respnse Sheets in Control Room for actual Setpoints

SER	Printout
49001206	R1 CONTROL ROOM RADIATION HIGH
49001207	R2-CONTAINMENT RADIATION HIGH
49001209	ROOM RADIATION HIGH
49001210	R5 SPENT FUELPOOL AREA RADIATION HIGH
49001211	R6 SAMPLING ROOM RADIATION HIGH
49001212	R7 IN-CORE SEAL TABLE AREA RADIATION HIGH
49001214	R9 LETDOWN LINE AREA RADIATION HIGH
49001215	R10 NEW FUEL PIT RADIATION HIGH
49001216	R11 CNTMT AIR PARTICULATE MONITOR RADIATION HIGH
49001217	R12 CONTAINMENT GASOUS
49001218	R13 AUX BUIDLING VENT RADIATION HIGH
49001219	R14 AUX BUILDING VENT RADIATION HIGH
49001220	R15 CONDENSOR AIR EJECTOR RADIATION HIGH
49001221	R16 CNTMT FAN COILS SERVICE WATER RADIATION HIGH
49001222	R17 COMPONENT COOLING RADIATION HIGH
49001223	R18 LIQUID WASTE DISPOSAL RADIATION HIGH
49001224	R19 S/G BLOWNDOWN SAMPLE RADIATION HIGH
49001225	R20 AUX BUILDING HEAT EXCHANGER SW RADIATION HIGH
49001226	R21-CONTAINMENT STACK MONITOR RADIATION HIGH
49001227	R22 RHR PUMP PIT RADIATION HIGH
49001228	R23 CONTROL ROOM VENTILATION RADIATION HIGH
49001232	RADIATION MONITOR CHECK SOURCE ACTIVATED
49001390	R11 CONTAINMENT VENT FLOW HIGH/LOW
49001391	R13 AUX BLDG SAMPLE FLOW HI/LO
49001396	R21 CONTAINMENT SAMPLE FLOW HI/LO
49001397	R22-RHR PUMP PIT SAMPLE FLOW HI/LO

Sequence of Events Recorder (SER) (Continued)

SER	Printout
49001399	R14 AUX BUILDING SAMPLE FLOW HI/LO
49001453	R19 BLOWDOWN FAILURE
49001454	R20 SERVICE WATER FAILURE
49001455	R21 CONTAINMENT VENT FAILURE
49001459	R22 RHR PUMP PIT [redacted] FAILURE
49001468	R23 CONTROL ROOM VENT FAILURE
49001469	A TRAIN PS-100/POWER SUPPLY FAILURE
49001485	R39 SLUDGE FLTR DRM AREA RAD HI
49001486	R1 CONTROL ROOM AREA FAILURE
49001487	R2 CONTAINMENT AREA FAILURE
49001488	AREA PS-100/POWER SUPPLY FAILURE
49001489	R4 CHARGING PUMP ROOM FAILURE
49001490	R5 SPENT FUEL AREA FAILURE
49001491	R6 SAMPLING ROOM FAIURE
49001492	R7 IN-CORE SEAL TABLE FAILURE
49001493	B TRAIN PS-100/POWER SUPPLY FAILURE
49001494	R9 LETDOWN LINE FAILURE
49001495	R10 NEW FUEL PIT [redacted] FAILURE
49001496	R11 CONTAINMENT VENT FAILURE
49001497	R12 CONTAINMENT VENT FAILURE
49001498	R13 AUX BLDG VENT [redacted] FAILURE
49001499	R14 AUX BLDG VENT [redacted] FAILURE
49001386	R15 CONDENSER AIR EJECTOR FAILURE
49001387	R16 CONTAINMENT FAN COIL FAILURE
49001388	R17 COMPONENT COOLING FAILURE
49001389	R18 WASTE DISCHARGE FAILURE

3.11 Interlocks

- **R-11 Containment Vent & Purge Particulate RM:**
  Actuates a CVI signal on high radiation level. [redacted] for Train B equipment. Containment Purge Supply Dampers TAV-12/CD-34033 and RBV-2/CV-31126 close. Purge Exhaust Dampers RBV-3/CV-31124 and RBV-5/CD-34006 also close. Post-LOCA H_2 Isolation 1B Valve LOCA-2B/MV-32146 and SA-7003B/MV-32148 close.
- **R-12 Containment Vent & Purge Gaseous RM:**
  Actuates a CVI signal on high radiation level. [redacted] Same dampers as R-11 are closed.
- **R-13 Aux. Bldg. Vent RM:**
  Closes the Waste Gas Discharge Valve WG-36/CV-31215 in the gas discharge line. Shuts down the Aux. Bldg. normal vent supply and exhaust fans. Activates Train A of the Aux. Bldg. Special Ventilation (SV) System. [redacted] Repositions SFP dampers by opening Charcoal Filters A and B inlet and outlet dampers and closing the filter bypass dampers. Shifts Sample Discharge R-11 & R-12 to Containment by closing AS-35/SV-33621 and opening AS-31/SV-33622. Closes Post-LOCA Valves: LOCA-100B/CV-31725, and LOCA-201B/CV-31727.
- **R-14 Aux. Bldg. Vent RM:**
  Performs the same actions as R-13 on the redundant Train B equipment at a high radiation level. [redacted] Does not shift R-11 and R-12 Sample Valves.
- **R-15 Condenser Air Ejector Gas RM:**
  On a high radiation level [redacted] routes the condenser air ejector discharge to the Aux. Bldg. Vent by de-energizing AR-6/CV-31168 if the Turb. Bldg. has been selected.

Also closes SG Blowdown and Sample Valves: BT-2A/MV-32077, BT-3A/MV-32078, BT-2B/MV-32079, BT-3B/MV-32080, BT-32A/CV-31335, BT-31A/CV-31334, BT-31B/CV-31270, and BT-32B/ CV-31271. This RM also closes Valve HS-17-1/CV 31770, Humidification Steam inlet to the Office/Warehouse Annex.

- **R-18 Waste Disposal System Effluent RM:**
  Closes Valve WD-19/CV-31138 to the Circulating Water System (standpipe) when high radiation levels is reached.
- **R-19 [R-19B]SG Blowdown RM:**
  On a high radiation level on SG Blowdown, performs the same actions as R-15.
- **R-21 Containment System Vent Activity Monitor:**
  High radiation level alarm, performs the same functions (CVI) as R-11 and/or R-12 on the redundant equipment Train A equipment.
- **R-23 Control Room Vent RM:**
  On a high radiation level, automatically opens Post-accident Recirc Dampers ACC-3A/ MD-32397 and ACC-3B/MD-32371. Non-accident Fresh Air Dampers ACC-5/CD-34007, ACC-1A/MD-32367 or ACC-1B/ MD-32368 go closed. The Post-accident Recirc Fans A and B auto start.

4.0 Precautions and Limitations

Monitoring Channels Maintained - All channels of the Process and Area Radiation Monitoring should be maintained in operation regardless of the plant mode of operation.

No Discharges - No gaseous or liquid discharges from the plant to the environment should be conducted if the applicable radiation monitor is not functioning properly, except as permitted by Chemistry and/or HP.

5.0 References

1.0 Summary	N-RM-45, N-RM-45-CL, A-RM-45
2.0 Functions	
3.0 Design Description	USAR 11.2 — E-2013, E-2018E-2019, E-2951, E2021 M-748 — I & C Logic Description #45 Annunciator Response 47023 series, and 47024-15 For associated DCR's see:DCR 841, DCR 2172, DCR 3182, DCR 2876 Programs\KNPP Applications\Physical Changes\DCR Database\Sys 45
4.0 Precautions & Limitations	N-RM-45
5.0 References	KNPP Off-Site Dose Calculation Manual (ODCM) See Technical Specifications as referenced Operation Manuals #118856
6.0 Procedures	For complete listing of procedures associated with System 45, see: **H:\Procedures Issued\Proc Index\Sys 45**
7.0 Appendices	See attached Figures

5.1 Technical Specifications

NOTE: Refer to a Controlled Copy of KNPP Technical Specifications for plant operations.

TS Table 3.5-1 #8	Specifies the set point limit for the Containment Purge & Vent System detectors to be operable.
TS Table 3.5-4, #3.a	Requires that the Containment Ventilation Isolation (CVI) signal must be operable.
TS 3.1.d.5	Provides minimum RCS leak detection systems requirements for when the Reactor is critical and above 2% power.
TS 3.8.a.8	Identifies requirements for when the Containment Vent & Purge System isolation must be operable during and just prior to refueling.
ODCM	Requires specific process liquid effluent monitoring instruments to be operable or actions implemented and routine surveillance performed.
Formerly in TS	Requires specific radioactive gas monitoring instruments to be operational or actions implemented and routine surveillance checks performed.

6.0 Procedures

For listing of Procedures specific to System 45, see:
N\Procedures Issued\Proc Index\Sys #

7.0 Appendices

7.1 Attached Figures

Figures	Title	KNPP DWG's
KNP-RM01	Mechanical Vertical Panel "C" RM System Control Consoles	M-748
KNP-RM02	Containment Bldg Vent Radiation Monitoring	OPERM-602
KNP-RM03	Containment System Vent Air Particulate Monitor R-11	E-2013
KNP-RM04	RM Process Monitors	
KNP-RM05	Aux Area Radiation Monitoring Subsystem	E-2951
KNP-RM06	RM Process Channels	E-2018
KNP-RM07	RM Area Channels	
KNP-RM08	Isokinetic Probe	SD Specific
KNP-RM09	R-11 Monitor	

NOTE: The "Figures" (drawings) previously associated with the System Descriptions will **not** be revised and updated at this time. Instead (obsolete) copies of these Figures have been stamped HISTORICAL and temporarily attached.