1/1/73

APPLICANT'S ANSWERS TO THE ASLB QUESTIONS
OF NOVEMBER 6, 1972

In the Matter of The

TENNESSEE VALLEY AUTHORITY

(Watts Bar Nuclear Plant Units 1 and 2)

DOCKET NOS. 50-390, 50-391

Environmental Question No. 1

"The first question is on page 3-5 of the environmental statement supplement. It is stated that the liquid radwaste disposal system is designed for 'a non-recyclable leakage of 20 gallons of primary coolant per day per unit excluding primary to secondary leakage.'

"Is this a reasonable design criterion and what is the experience basis for it? Does the Staff consider this to be acceptable and on what basis?" (Tr. 45)

Response

The design criterion of 20 gallons per day per unit of non-recyclable reactor coolant has been evaluated on the basis of operating plant experience and is considered to be reasonable for current plant designs, including Watts Bar Units 1 and 2.

The Liquid Waste Disposal System for the Watts Bar Plants is a modification of plants now in operation. These modifications enable the recycling of essentially all of the radioactive reactor coolant water from various sources in the plant with minimum release to the environment. This is accomplished primarily by the segregation of liquid drains.

The Watts Bar Liquid Waste Processing System consists of two main subsystems designated as drain channel A and drain channel B. Channel A normally processes all water which can be recycled and which in past systems was normally discharged. These effluents included equipment maintenance drains, excess samples, demineralizer backwashes, and piped-up valve and pump leakoffs. Drain channel B collects all effluents which cannot be recycled such as controlled area floor drains and laboratory equipment rinses. The primary reason that these effluents cannot be recycled is that for the most part they do not come from the primary systems. To recycle

An analysis of operating experience with waste leakage using an earlier waste disposal system design indicated that the portion of waste liquid attributable to leakage of reactor coolant was approximately 36 gallons per day. This leakage was from the fifteen sources listed in Column 2 of Table 1. Column 4 of Table 1 shows the leakage sources of reactor coolant which are processed and recycled in the Watts Bar waste processing system. Most of the sources of reactor coolant leakage which comprised the 36 gallons per day figure thus are recycled in the Watts Bar waste processing system leaving only minor sources of reactor coolant leakage which is non-recyclable. Although this amount is expected to be very small, the system nevertheless is designed for 20 gallons per day.

Examples of sources which could contribute to such leakage include the following:

(a) Valve stem packing leakage (negligible)

(b) Tubing and pipe fitting leakage (negligible)

(c) Inadvertent spillage during maintenance (negligible)

(d) Charging pump seal leakage (less than 1 gallon per day)

Some earlier plants experienced reactor coolant leakage in excess of 36 gallons per day primarily due to valve stem leakage from pressurizer spray valves and positive displacement charging pump seals. These sources of leakage have been essentially eliminated by incorporating in later plants such as Watts Bar new designs for the pressurizer spray valves using ball and bellows type valves and by providing a leak-off collection system for the positive displacement charging pumps.

them would result in a net accumulation which could not be accommodated. It is the Channel B waste liquid that is discharged after processing.

Table 1 below relates the disposition of the various sources of Liquid Wastes typical for operating plants using an earlier waste disposal system design as compared to current designs such as included in the Watts Bar plant.

TABLE 1

DISPOSITION OF PRIMARY LIQUIDS ENTERING WPS

(1)	(2)	(3)	(4)	(5)
	Typical Operating Plant System	Watts Bar System		
Source of Effluent	Processed & Discharged	Processed & Discharged (Channel B)	Processed & Recycled (Channel A)	Drummed Directly
Tank Drains	X		X	
Filter Drains	X		X	
HX Drains	X		X	
Demineralizer Drains	X		X	
Demineralizer Flushes	X		X	
Pump Leakoffs	X		X	
Valve Leakoffs	X		X	
Sample Sink Drains	X		X	
Excess Samples	X		X	
Laboratory Equipment Rinses	X	X		
Decontamination Drains	X	X		
Floor Drains	X	X		
Spent Samples	X			X
Laundry, Hot Showers and Hand Washes	X	X		
Non-recyclable Reactor Coolant Leakage	X	X		

Therefore, system modifications now included in plant designs such as Watts Bar coupled with leakage detection systems and a maintenance program assure that the 20-gallon per day design basis is reasonable and appropriate.

Environmental Question No. 2

"The second question, on the same page, is the statement that evaluations are based on continuous 20-gallon per day primary to secondary leakage in one unit. I have the same question with regard to that" (Tr. 46).

Response

The use of 20 gallon per day average annual primary to secondary leakage per unit for current plant designs is a highly conservative assumption based on the following considerations:

1. The operating data from Westinghouse plants presently in operation has resulted in a much lower average continuous primary to secondary leak rate. Table 1 shows the plant by plant tabulation of data which yields an average leak rate of 8.6 gpd on a continuous basis for presently operating plants. The future operation of Westinghouse plants is expected to improve this number significantly due to the experience gained to date and the remedial steps which have already been implemented in all present and future operating plants.

2. The 20 gallons per day annual average leak rate is based on the leak existing throughout the whole annual fuel cycle (i.e., the leakage develops on the very first day of operation after refueling and continues throughout the entire annual cycle). This is a highly unlikely occurrence and certainly during the annual cycle opportunities would arise to perform repairs during a scheduled maintenance shutdown.

TABLE 1

CUMULATIVE AVERAGE PRIMARY TO SECONDARY LEAK RATE

FOR OPERATING WESTINGHOUSE PLANTS

(Inconel Tube Steam Generators)

Plant	Operation Days To 9/1/72	Average Leak Rate (GPD)
Conn Yankee	1423	15.7
San Onofre	1303	1
Zorita	1153	0
NOK Beznau I	732	7.0
Ginna	892	0
Robinson	396	17.2
Pt Beach 1	528	43.3

Environmental Question No. 3

"The third question, on page 3.6, again, the draft environmental statement supplement, is a statement relating to tritium concentrations that can be tolerated during the refueling. The conclusion is stated that 2.5 microcuries per CC is acceptable for analytical purposes and this is based on the exposures that would be received by personnel during refueling.

My question is what are the estimated exposures that would be received under those circumstances?" (Tr. 46)

Response

For a tritium concentration in the reactor coolant system of 2.5 uCi/cc and a corresponding tritium concentration after mixing with water from the refueling water storage tank of 1.5 uCi/cc, the concentration of tritium in the air over the refueling canal is calculated to be 1.7 x 10^{-5} uCi/cc. The dose commitment to a person exposed to this concentration of tritium is calculated to be 3.4 mrem per hour of exposure. A person could be present on the bridge above the refueling pool for approximately 100 hours during the refueling process. The resulting dose for the exposure for this time period would be 340 mrem which is 6.8% of a worker's maximum permissible annual dose. Only two people would be exposed at this location.

To minimize the radiation exposure to plant personnel during the refueling process a purge system is arranged to draw exhaust air into a plenum around the periphery of the refueling canal effecting a ventilation sweep of the canal. If this ventilation system is completely effective a person above the refueling pool will not be exposed to tritium in the air. In the above analysis it has been conservatively assumed that the receptor is exposed to the same tritium concentration as in the exhaust air from over the refueling pool.

The ventilation system should be very effective in reducing the doses to persons located on the sides of the refueling pool. If it is assumed that the tritium concentration in this area is 10% of that in the exhaust air from over the refueling pool, the dose to a person in this area for a period of 100 hours would be 34 mrem. About four persons could be at this location.

The tritium concentration in the reactor coolant system will increase gradually. It is estimated that it will take about 8 years for this concentration to reach the 2.5 uCi/cc level. At the time of the first refueling period the tritium concentration in the reactor coolant system is calculated to be 0.6 uCi/cc and the concentration after mixing with water from the refueling water storage tank is calculated to be only 0.1 uCi/cc. The estimated maximum dose to a person located over the refueling pool for a period of 100 hours is only 23 mrem, which is less than one-half of 1 percent of a worker's maximum permissible annual dose.

At the time of the second refueling the tritium concentration in the reactor coolant system is calculated to be 1.1 uCi/cc and the corresponding concentration after mixing with water from the refueling water storage tank is approximately 0.25 uCi/cc.

Since the buildup of the tritium concentration in the reactor coolant system can be predicted, future increases in radiation doses can also be predicted. This will allow the opportunity to carefully evaluate the potential for doses from tritium during refueling operations.

Environmental Question No. 4

"The fourth question has to do with the discussion on pages 3-8 through 3-11 relative to extended treatment of steam generator leaks. On page 3-9 it is mentioned that in case of primary to secondary leakage iode noble gases will be released through the mechanical vacuum pump.

"Is this pump used during other than start-up operations and if so, under what circumstances will this pump be operated with a contaminated secondary system?" (Tr. 46)

Response

The condenser vacuum will be established and maintained with a continuously operating two-stage mechanical vacuum pump. During startup operations, the second stage of the vacuum pump will be by-passed to provide a hogging mode for rapidly establishing an intermediate condenser vacuum. During normal operation the noncondensibles from the pump are continuously discharged through a filter system consisting of a HEPA and two charcoal filters. The filter system will be by-passed during startup.

The estimated annual releases and doses are reported in the final environmental statement (Tables F-1, F-2, F-3, and F-4, respectively, for a 20-gallon per day primary to secondary leakage rate and a 6-gallon per minute blowdown rate per unit.)

Environmental Question No. 5

"Page 3-12 of the draft environmental statement supplement says that each containment will be purged twice a year. Page 9-5 of the Staff's safety evaluation says that the Applicant has estimated 12 purges per year and it is not clear whether that is for one unit or for two.

"My question is: Which is correct? If one of these is incorrect, does it change any conclusions that are drawn in the document as a result?" (Tr. 46-47)

Response

The numbers of purges per year reported in both the draft environmental statement and the Staff's safety evaluation report are for each containment and both are correct numbers. The difference is that the environmental statement uses a realistic number of two purges per year with the maximum expected value of 0.25% defective fuel pins whereas the Staff's safety evaluation quotes the design basis value of 12 purges per year with 1% defective fuel pins. Therefore, the conclusions drawn in both documents are correct for the two different types of evaluations. Even if twelve purges per year had been used in the environmental statement, the doses as reported in the final environmental statement in Tables F-1, F-2, F-3, and F-4 would remain essentially unchanged.

Health and Safety Question No. 1

"Are the transmission lines at the point where they leave the switchyard sufficiently separated to avoid the possibility of a common accident damaging all of them?" (Tr. 47)

Response

As stated in Section 8.2 of the PSAR, the two transmission lines to and from the Watts Bar Hydro switchyard will be supported on separate towers and the separation will be adequate to ensure that the failure of any transmission line tower will not endanger the other line.

Health and Safety Question No. 2

"Page 1-6 of the Staff environmental statement [Safety Evaluation] says, and I quote, 'The proposed Watts Bar Nuclear Plant resembles the Sequoyah Nuclear Plant in every significant engineering sense important to safety.'"

"I would like to get some explanation to this statement in order to give some feel for the degree of resemblance and what is meant by 'every significant engineering sense' with 'significant' underlined" (Tr. 48).

Response

The applicant has discussed in detail the similarity between the Sequoyah and Watts Bar plants in Appendix E and Table 1.2-1 of the Preliminary Safety Analysis Report.

Health and Safety Question No. 3

"The third question I have is on page 2-6 [of the Safety Evaluation]; meteorological diffussion and relative concentrations are discussed. It is stated that NOAA's concentration values are in substantial agreement with those of the Staff's. Based on this page and on the NOAA report in appendix B, the Staff relative concentration for short-term accidents at 1200 meters is 3.4 times 10 to the minus third; and NOAA's value at 790 meters is three times 10 to the minus third. For the annual average the Staff has a value of 2.6 times 10 to the minus fifth at 1200 meters compared with NOAA's 1.4 times 10 to the minus fifth at 790 meters. If both calculations are taken at the same distance, are they still what the Staff would term as 'in substantial agreement'? I would like to know what the Applicant's values are for these two parameters, also. They are generally described on the same page, but I couldn't find any explicit statement of what the numbers are." (Tr. 48-49)

Response

The applicant has estimated the annual average dispersion value to be 2.34×10^{-5} sec/m^3 at the site boundary and has used this value in the evaluation of the environmental effects of gaseous effluents. The applicant's estimate of the short term dispersion factor is 3.5×10^{-3} sec/m^3 at 1200 meters which is in excellent agreement with the estimate of the staff.

Health and Safety Question No. 4

"On page 208 [of the Safety Evaluation], the statement is made that the Applicant has proposed constructing most facilities above all but 'the most severe flood levels.'"

"What does 'the most severe flood levels' mean? That is not clearly defined" (Tr. 49).

Response,

The features necessary to maintain the plant in a safe shutdown condition are protected to the probable maximum flood (elevation 743.5). The plant is designed to remain operational for any flood that does not exceed plant grade (elevation 728). For floods that could exceed elevation 728, the plant will be placed in a safe shutdown condition.

The normal full pool level of Chickamauga reservoir is elevation 682.5. The greatest historical flood occurred in March 1867, prior to the establishment of the TVA flood control system, and reached elevation 716 at the plant site. The same flood, regulated by the TVA flood control system, would reach elevation 701. These levels are considerably below plant grade which will be established at elevation 728. A discussion of Hydrology (including flood levels) is given in PSAR Section 2.7.

Health and Safety Question No. 5

"It is stated that plant grade will be 627--excuse me--728 or 729--both numbers appear--but that safety related structures will be protected to 743.5. This is about 15 feet above grade and I would be interested in a general description of how this will be done. Again, this is one that may be adequately answered in the PSAR" (Tr. 49).

Response

As described in various sections of the PSAR, provisions to protect the safety related features for the approximately 15 foot rise above plant grade (elevation 728 feet) by the maximum possible flood plus wave action (elevation 743.5 feet) are as follows:

1. Waterproofing and watertight bulkhead doors are installed in the reactor shield buildings and diesel generator building. These buildings are kept dry and anchored against floatation.
2. The intake structure pumping deck, where the safety related pumps are located, is above elevation 743.5 feet.
3. The lower portions of the auxiliary building, control building, turbine building, and service building are permitted to flood. (The TVA flood control and forecasting systems will provide a minimum of 36 hours of warning time before grade is exceeded to prepare for such a condition.)
4. All of the essential equipment in the control building and auxiliary building that is required to operate during a flood is located at elevations above 743.5 feet.

Health and Safety Question No. 6

"On page 5-12, it is stated that the Staff has reviewed the information presented in regard to the design basis performance and the effects of containment parameters on the hydrogen combiner performance."

"I would like to have an identification of the information that has been presented and if it is something other than what is in the documents that are presently available to the Board, I would like to get a copy of it if we could." (Tr. 49-50)

Response

The post-accident hydrogen control system description and performance capability is discussed in Section 6.7 of the PSAR. Figure 6.7-2 shows the hydrogen generation in the containment, for the AEC accepted release basis case. Figure 6.7-3 demonstrates the performance capability for one of the two available hydrogen recombiner units. The hydrogen concentration in the containment is controlled below the 4 volume percent based on the AEC TID release model and is in conformance with the guide limits indicated in AEC Safety Guide 7.

Westinghouse has also furnished the AEC Regulatory Staff with the following topical reports as background information in their evaluation of the Westinghouse Electrical Recombiner System:

WCAP-7820 "Electrical Hydrogen Recombiner for Water Reactor Containments" (Submitted to AEC 12-16-71)

WCAP-7820, Supplement 1 "Electric Hydrogen Recombiner for PWR Containments" (Submitted to AEC 5-31-72)

The above reports were made available to the AEC public document room.

Health and Safety Question No. 7

"Please discuss the basis for the Staff conclusion on page 5-18 [of the Safety Evaluation] that 'the Applicant has developed sufficient preliminary design information on which to base confidence that the EGTS will function as intended'" (Tr. 50).

Response

The Applicant has described the system design bases and the preliminary design in Section 5.1 of the PSAR.

Health and Safety Question No. 9

"The geological survey on appendix C, page 3 [of the Safety Evaluation], states that in some places the top surface of unweathered [Conasauga] shale may be several feet below the proposed foundation grade. The Newmark report on page E-3 in the appendix states that a major portion of the plant will be founded five to ten feet below the surface of the weathered rock which geological survey states is one to three feet thick.

"These two conclusions appear to be contradictory and I would like to have some explanation" (Tr. 50).

Response

Examination of PSAR Figures 2.8-64 and 2.8-65, Geologic Sections, Reactor and Control Areas, reveals that the foundations of all Class I rock supported structures are at least 5 feet below the elevation of the top of the shale, with the top of the shale as determined from the boring logs. This is as Dr. Newmark states in his report.

The U.S. Geological Survey, in Appendix C, page 3, incorrectly states that the top of the shale would generally be below foundation grade at holes 21, 29, 36 and 43 (Figure 2.8-59, Rev. 1, PSAR). This statement is based on the assumption for a final foundation grade which is not in agreement with Figures 2.8-64 and 2.8-65 referenced above. These figures reveal that hole 21 is not within the limits of the structures and that the structure foundations are 7, 12 and 6.5 feet below the top of shale at holes 29, 36 and 43 respectively.

If, by chance, there are local areas between drill holes where the top of shale is below a structure foundation, customary TVA practice is to excavate the areas to an elevation below the weathered zone and place concrete to the elevation of the structure foundation. This practice will be followed at the Watts Bar Nuclear Plant.

Health and Safety Question No. 10

"Again, this may be one that the Staff is already planning on addressing--the third paragraph on the second page of the ACRS letter discussed the ECCS system.

"Is there any question on the part of the Staff or as far as the Staff knows on the part of the ACRS that the present ECCS design satisfies the present interim acceptance criteria? Has the Applicant agreed to present the final design for review by the Regulatory Staff and the ACRS Staff prior to installation and compilation of major components as recommended by ACRS?

"Does the Staff believe there is a reasonable assurance that any changes that the Commission might be expected to make in ECCS criteria will be such that Applicant will be able to accommodate them by suitable changes or economically reasonable de-rating." (Tr. 50-51)

Response

The analysis presented in Appendix F of the Watts Bar PSAR shows that the ECCS performance meets the Interim Acceptance Criteria. Westinghouse is actively engaged in a study program to achieve an optimum solution to meet the AEC Interim Acceptance Criteria. These studies include such items as peak power density, model improvements, and system modifications. When the program is completed in late 1973, the final design of the ECCS will be submitted to AEC for review and approval. Meanwhile, the Watts Bar Nuclear Plant design will be kept sufficiently flexible to incorporate the essential features of the final solution as approved by AEC. (PSAR Appendix F, p. F1-19.)

Health and Safety Question No. 12

"On pages 5-3 to 5-5, the safety evaluation discusses independent analyses that they have required in connection with the ice condenser containment systems. These apparently have been completed during 1972.

"How are these analyses related to the ACRS recommendation, if at all? If they are not related, please explain the differences. Those two questions sort of go together" (Tr. 51-52).

Response

The Staff Safety Evaluation discussion of an independent analysis refers to an independent analysis of ice condenser containment performed by MPR Associates and reported in "Ice Condenser Containment Analysis Program Final Report", April, 1972, filed with the Commission as a part of the D. C. Cook Application (Docket Nos. 50-315 and 50-316). This report was incorporated in the Watts Bar license application by reference in the Watts Bar PSAR, page 14.5-10.

The ACRS letter dated September 21, 1972, page 2, 5th paragraph refers to the need for additional analytical studies of local and overall pressures in the ice condenser containment for various postulated loss-of-coolant accidents.

The results of recent additional analytical studies of ice-condenser containment transient pressure response have been submitted to the Staff by means of Amendment 17 to the PSAR.

1. Local pressure transients are discussed on pages 14.5-4 through 14.5-11(c) with the results summarized on Tables 14.5-1 through 14.5-5. (Note that a typographical error appears in Table 14.5-5. The pressure 112.21 psig should read 12.21 psig.) This additional analytical study

was performed primarily to incorporate the assumption that the liquid component of the break fluid is entrained in the steam phase during blowdown. This assumption is found to be more conservative than assuming no entrainment.

2. The overall pressure transient is covered by Appendix G of Amendment 17, entitled "Containment Pressure Response to a LOCA". This additional study considered the pump suction break (LOCA) and was found to predict slightly higher containment pressures. The steam generator was considered as an active heat source which would superheat the break fluid by transferring energy from the secondary to the primary.

Subsequent to these additional analytical studies and based upon earlier reviews, the Commission further questioned the adequacy with which ice-condenser design margins are identified. The Commission is requesting that the Applicant identify and quantify the existing margins in the following areas: (1) margins in the predicted energy release following a LOCA, (2) margins in the predicted peak containment pressure due to the transient pressure response following a LOCA, and (3) margins in the structural design of the containment. The Commission has emphasized the necessity for a complete and accurate evaluation of these margins.

Health and Safety Question No. 14

"I would like to know what the Applicant's estimates are of the radiation exposure that employees will get during the course of normal operation and maintenance. If possible, a breakdown of this by ranges. Zero to 1-R per year, 1 to 2, 2 to 3, this sort of thing. If possible an estimate of the total man rems that are anticipated. What measures have been taken to keep these values as low as possible and the basis for the estimates" (TR. 52-53).

Response

DESIGN CONSIDERATIONS FOR OPERATION

TVA is giving every consideration to protection of the employees with regard to exposure to radiation and radioactive materials. Prior to the design and construction of the WBNP, design criteria were established which considered aspects such as anticipated frequency and occupancy times, estimated radiation levels, and maintenance requirements. Radiation levels were limited considering occupancy times so that no individual would receive more than 5 rem per year in carrying out his assigned duties. However, it should be noted that the occupancy durations and shielding design are based on operation of the plant with a 1.0% failed fuel fraction, whereas operating experience with pressurized water reactors indicates that 0.25% failed fuel would be more realistic. All areas of the plant have been classified into zones according to anticipated access as follows:

Zone	Access Conditions	Occupancy	Shielding Design Dose Rate	Access Dose Rate Range
1	Unlimited	Continuous	<1.0 mrem/hr	0-1 mrem/hr
2a	Regulated	Continuous	1.0 mrem/hr	1-5 mrem/hr
2b	Radiation Area	Periodic	15 mrem/hr	5-100 mrem/hr
3	High Radiation Area (Controlled)	Unoccupied	20-1000 mrem/hr	100-1000 mrem/hr
4	High Radiation Area (Restricted)	Unoccupied	1000 mrem/hr	< 1000 mrem/hr

Certain administrative procedures are also followed when entering specific zones. These procedures will be discussed later in detail.

Typical Zone 1 areas will be the turbine building, offices, turbine plant service areas and control rooms. Zone 2a areas would include the local control spaces in the auxiliary building, and the operating deck of the containment during reactor shutdown. Areas designated Zone 2b include grade level areas adjacent to the containment structures, fuel handling areas, waste packaging areas and intermittently occupied work areas. Typical Zone 3 areas will be the pump, and tank cubicles in the auxiliary building. Typical Zone 4 areas will be demineralizer cubicles and areas adjacent to reactor coolant piping and equipment during operation.

All radiation and high radiation areas will be appropriately marked and isolated in accordance with 10 C.F.R. Part 20 and other applicable regulations.

The radiation sources which provide the design bases for radiation protection are subdivided into three categories according to their origin or location:

a. The reactor core, internals and reactor vessel sources consist of leakage neutrons, fission gammas, fission product gammas and gammas from interactions between neutrons and water or structural materials.

b. The reactor coolant sources consist primarily of gamma radiation from Nitrogen-16 decay during operation and activated corrosion products and fission products resulting from assumed 1 percent defective fuel during shutdown.

c. Auxiliary systems equipment sources result from processing reactor coolant containing activated corrosion and fission products.

Primary Shielding

Primary shielding will be provided to limit radiation from the reactor core, internals and the reactor vessel and will be designed to:

a. Attenuate leakage neutron flux to prevent excessive activation of components and structures in the containment.

b. Reduce the amount of radiation from the reactor to obtain a reasonable division of shielding function between primary and secondary shielding.

c. Reduce shutdown radiation from the reactor to allow access to the area between the primary and secondary shields within a reasonable time after shutdown. The primary shield consists of a reinforced concrete structure immediately surrounding the reactor vessel and extending up to the concrete cavity above the reactor vessel. The reactor cavity, which is approximately rectangular in shape, has concrete walls which extend upwards to the underside of the operating deck.

Secondary Shielding

Secondary shielding will be provided to limit radiation from the reactor coolant system outside of the primary shield and will be designed to reduce the radiation intensity at the outside surface of the containment building to a negligible level during normal plant operation.

The Shield Building and interior concrete walls and floor provide secondary shielding for the WBNP.

Fuel handling shielding will be provided to attenuate radiation from spent fuel, control rod clusters and reactor vessel internals and will be designed to facilitate the removal and transfer of spent fuel assemblies and control clusters from the reactor vessel to the spent fuel pit. The fuel handling shielding consists of both concrete and water. The reactor cavity, flooded during refueling operations, will provide a temporary water shield above the components being withdrawn from the reactor vessel. The refueling canal is a passageway connected to the reactor cavity and extending to the inside surface of the containment vessel. The canal, formed by two shielding walls, extends upwards to the same height as the reactor cavity. During refueling operations, the reactor cavity and refueling canal are flooded with water to the same height.

The spent fuel assemblies and control rod clusters are transferred remotely from the reactor containment to the spent fuel pit via the refueling canal and the horizontal spent fuel transfer tube. The shielding will be designed to protect operating personnel during the transfer operation and in the event of credible malfunctions. The reactor cavity and refueling canal walls will protect personnel working in adjacent areas. Those working in the area of the spent fuel pit will be protected by the water in the pit during fuel handling operations.

Auxiliary building shielding will be provided to protect personnel in the sampling room and in the vicinity of the waste disposal, chemical volume and control and auxiliary coolant systems. Where required, piping will be located in shielded pipe trenches. The auxiliary building will be compartmented so that equipment areas may be entered without having to shut down adjacent operating systems or equipment. Where possible controls will be placed outside compartments so that entry into those areas will be reduced to a minimum.

Additional shielding will be provided for the control rooms. This shielding and that which will be provided by the secondary shielding assures protected control areas for continuous operation of either unit, and for the operation of engineered safety features.

Consideration will be given to the possibility of increased employee exposure as a result of the addition of the extended radwaste systems that may be installed to reduce off-site exposure from normal plant releases. The additional sources of radiation are known and appropriate shielding, ventilation, and controls will be provided consistent with the philosophy expressed herein, thereby minimizing the potential for exposure to the employee from these additional sources of radiation.

DESIGN CONSIDERATIONS FOR MAINTENANCE

Consideration will be given to reducing radiation for maintenance activities by providing shield walls, decks, and floors inside primary containment to shield workers from the reactor vessel. Arrangement of equipment and shielding in the plant will also be considered to reduce radiation to the worker to the extent practical from radioactive equipment other than the particular piece of equipment being maintained.

Attention will be given in design to reducing the occupancy times for maintenance workers during repairs and inspections. An important consideration in this regard has been to provide, where possible, adequate space around equipment to permit access for efficient work. Also, provisions will be made for adequate and controlled laydown areas for repairs requiring equipment dismantling as well as effective decontamination facilities to reduce the radiation levels of the equipment. Easy access will be provided to most equipment to reduce travel time in areas where high radiation levels are expected.

To reduce radiation exposures during in-service inspections of the reactor system inside primary containment, special attention will be given to defining the locations of welds which will need to be inspected. Insulation will be provided which can be readily removed and replaced at those welds so as to reduce exposure time for preparing for the actual nondestructive testing. Because of the very high levels of radiation associated with ultrasonic inspection of the reactor vessel welds, provisions have been made to perform most of the ultrasonic scans remotely.

<u>RADIATION MONITORING SYSTEMS</u>

To further protect the employee from potential hazards associated with radiation and radioactive materials, a radiation monitoring system will be designed for the plant. Alarm points, which will be pre-established, will be set at radiation levels much below the point at which an employee, if exposed, would exceed the daily guide for limiting exposure.

The radiation monitoring system will be designed to perform two basic purposes:

a. Warn plant personnel of increasing radiation levels which might result in a radiation health hazard.

b. Give early warning of a plant malfunction which might lead to release of radioactive materials to the plant which could result in a health hazard or plant damage.

Instruments will be located at selected points in and around the plant to detect, compute, indicate, annunciate, and record the radiation levels. In the event the radiation level should rise above a predetermined setpoint, an alarm will be initiated locally and in the control room.

The system will be divided into the following subsystems:

a. The Process Radiation Monitoring System will monitor various fluid streams for indication of increasing radiation levels.

b. The Area Radiation Monitoring System will monitor radiation in various parts of the plant normally accessible to operating personnel. The system will consist of several channels which indicate radiation levels in various portions of the plant. Fixed gamma-sensitive detectors will monitor such areas as control rooms, containments, the radiochemistry laboratory and the auxiliary building for gamma radiation. Radiation levels will be indicated locally at the detector and in the main control room, and each channel will alarm locally and in the control room in the event allowable radiation limits are exceeded. In addition, air particulate monitors will be located in plant areas susceptible to the release of airborne radioactive material. The sample will be monitored by three separate systems: a particulate monitor, an iodine monitor, and a total gas monitor. The activity level associated with the three monitoring systems will be recorded locally and in the main control room. An annunciator and alarm system will be provided on high activity alarms.

ADMINISTRATIVE CONTROLS

To prevent inadvertent entry by personnel into radiation areas, rigid access control will be maintained, which may include locked and barricaded doors, interlocks, and a system of local and remote (control room) alarms. Administrative control will include the use of special work permits, health physics surveys, and a "supervisory key" issued at the control room. Access to other less hazardous areas will be controlled by administrative procedures, as shown in the following table.

Zone	Access Conditions	Occupancy	Access Dose Rate Range	Action Requirements to Enter Zone
1	Unlimited	Continuous	0-1 mrem/hr	None
2a	Regulated	Continuous	1-5 mrem/hr	Administrative control (Signs, placards, direct reading instruments)
2b	Radiation Area	Periodic	5-100 mrem/hr	Administrative control (Signs, placards, direct reading instruments)
3	High Radiation Area (Controlled)	Unoccupied	100-1000 mrem/hr	Special work permit; locked doors, signs, temporary barricades; health physics surveillance
4	High Radiation Area (Restricted)	Unoccupied	<1000 mrem/hr	Positive exclusion, including locked doors, special work permits, and continuous health physics monitoring

All radiation and contamination zones will be properly identified with signs and labels in accordance with requirements of 10 C.F.R. Part 20, Standards for Protection Against Radiation.

Throughout operation of the WBNP, routine and special radiation and contamination surveys will be conducted to delineate and evaluate radiation and contamination hazards and to determine the necessary work limitations and physical safeguards. These surveys will be performed by a trained staff of health physics personnel whose primary responsibility will be to limit unnecessary radiation exposure of plant personnel. This will be accomplished through several means, as outlined below.

1. Direct Radiation Limits

a. Exposure Guidelines for Planning Work Schedules

Whole Body Dose Rate	Approval Required to Exceed Guideline
50 mrem/single day or 100 mrem/single week	Health physics technician and employee's supervisor
1 rem/hour or 100 mrem/single day	Health physics supervisor and shift engineer
50 rem/hour	Above and the plant superintendent

b. Limiting Doses to Occupational Workers

In addition to the regulating limits set forth in 10 C.F.R. Part 20, exposure to non-TVA personnel will be limited in the following manner.

(1) 300 mrem/calendar quarter, or

(2) 1,250 mrem/calendar quarter if dose records are supplied for the individual(s) for the present calendar quarter. The exposure permitted shall be controlled so that the total dose received shall not exceed the 1,250 mrem/calendar quarter.

2. Airborne and Surface Contamination Limits in Nonregulated Areas of Plant

Type of Radiation	Airborne Contamination (uCi/cc Air Unidentified)[a]	Surface Contamination Direct Reading	Surface Contamination Transferable (dpm/100 cm^2)
Alpha	7×10^{-11}[b]	300 dpm/100 cm^2	30
Beta-Gamma	1×10^{-9}[c]	0.25 mrad/hr[d]	1,000

a. If the identity of the radionuclides is known the concentration limits in 10 C.F.R. Part 20, Appendix B, shall apply.

b. If no one of the alpha-emitting radionuclides, ^{230}Th, ^{231}Pa, ^{232}Th, Th-nat, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{242}Pu, and ^{249}Cf is present.

c. If no one of the beta-emitting radionuclides, ^{210}Pb, ^{227}Ac, ^{228}Ra, and ^{241}Pu is present.

d. As measured with a GM survey meter calibrated with a standard beta source.

It should be further noted that the limits established above for airborne contamination (unidentified activity) are established using the most restrictive maximum permissible concentration (MPC) for an isotope (strontium-90) known to be present in a pressurized water reactor system. Respiratory protection will be required in all areas where this value is exceeded until such time that the activity has been identified. After identification of the activity MPC values established in 10 C.F.R. Part 20 will be used. Respiratory protection will be worn in all areas where MPC's of radioactivity are known to be exceeded, regardless of occupancy time, with the exception of tritium.

3. <u>Special Work Permits</u>

The special work permit is a special form on which the appropriate supervisor describes the location and type of work to be done and the plant health physicist or his representative prescribes the work limitations and radiation protective measures to be applied.

a. A special work permit will be required:

(1) In advance of any work assignment where it is anticipated that an employee may receive a whole body radiation exposure greater than 50 mrem in any regular workday.

(2) In advance of work involving dose rates greater than 100 mrem/hr to the total body.

(3) For work in an area having airborne radioactivity greater than $(MPC)_a$ for a 40-hour week or where the contamination levels exceed the values given previously.

(4) When radiation or contamination hazards for a particular job are unknown or for other reasons for which the plant health physicist or his representative requires special precautions.

b. Each special work permit will require:

(1) The supervisor's description of the work to be performed, including location, list of employees involved, duration, and special tools or equipment required.

(2) Reference, by number, to related special work permits.

(3) Specification by the plant health physicist or his representative of protective clothing and equipment, work limitations, and time restrictions.

(4) Signatures of the supervisor seeking the special work permit, the plant health physicist or his representative specifying the limitations, and the shift engineer.

4. <u>Training</u>

a. Employees whose work involves potential exposure to ionizing radiation will be adequately trained in radiation protection methods. All regular WBNP employees will be given adequate training in the methods, practices, and procedures of radiation protection for the safe conduct of their work assignments.

b. All temporary employees and trainees whose duties involve potential exposure to radiation will be given adequate radiation protection training.

c. Short-term visitors will be given basic orientation in radiation protection as needed, depending on their prior background and training.

d. A record will be maintained by section supervisors of all formal training given each employee in radiation protection, including the names of instructors, subject matter included, and references.

5. <u>Exposure Records</u>

Radiation exposure records will be maintained on a current day-by-day basis. Film badges will be worn by all plant personnel. These badges will be processed on a monthly basis or as deemed necessary to evaluate each individual's exposure history. Pocket dosimeters will be worn by all personnel entering areas of the plant where there is potential exposure to sources of radiation. These dosimeters will be processed daily. A summary of weekly exposures, as determined from the pocket dosimeter records, will be distributed to each supervisor so that he can limit the accumulation of high individual exposure by distributing workloads and assignments. The exposure due to inhalation or ingestion of radioactive materials in contaminated work areas will be determined through routine bioassay and whole body counting, and made a part of the employee's overall exposure history.

ESTIMATE OF PERSONNEL EXTERNAL EXPOSURE

The anticipated maximum external radiation exposure to an employee is expected to reach the limit allowed by the Atomic Energy Commission (AEC) regulations as stipulated in 10 C.F.R. Part 20. TVA has, however, established administrative controls within its own organization to limit exposure of employees to no more than 5 rem in any one year, regardless of the individual's past radiation exposure history. Even though the anticipated maximum exposure to an individual is expected to reach 5 rem per year, TVA will make every effort to limit this type exposure to as few individuals as possible. This type exposure would only be received if major repair or inspection were necessary.

1. Estimate of External Exposure to Operating Personnel

Estimates of exposures to operating personnel at the plant during operation are reported in the table below. It should be noted that during this early design stage the occupancy time estimates are very preliminary and the exposure is based on design dose rates, which are design goals based on estimated maximum radiation levels and may vary somewhat from the data presented.

The estimates in the table are based on a working time of 2000 hours per year for each person considered and as such include the exposure to these persons during refueling and maintenance. The personnel classified as "operators" includes only those persons directly involved in plant operations. Those persons listed in the "other" category include health physics personnel, laboratory technicians, guards, technical staff, and administrative staff. The staffing used in these estimates is consistent with the organization as presented in figure 12.2-4 of the Preliminary

Safety Analysis Report (PSAR).

Zone	Access Conditions*	Occupancy Man-hrs/yr	Design Dose Rate	Est. Man-Rem Dose/Year
1	Unlimited	56,700 operators 20,000 others 76,700 total	0.1 mrem/hr	5.67 operators 2.00 others 7.67 total
2a	Regulated	19,600 operators 23,775 others 43,375 total	1.0 mrem/hr	19.60 operators 23.80 others 43.40 total
2b	Radiation Area	5,700 operators 2,225 others 7,925 total	15 mrem/hr	85.50 operators 33.80 others 119.30 total

*Dose estimates for Zones 3 and 4 will only be entered under very unusual circumstances during normal operation; therefore, dose estimates are not made for these classifications.

The estimated total man-rem dose per year for a 2-unit plant is 2.70 rem/yr/man for operators and 1.00 rem/yr/man for all other personnel. The average dose/yr/man is 1.70 rem for a 2-unit operation.

2. Estimate of External Exposure to Maintenance Personnel

Exposure to employees performing maintenance functions at the WBNP is most difficult to predict at this point in time. There is very little experience from operating plants available for use in projecting exposures to maintenance personnel. The projection of expected exposures based on such little information would be meaningless at this time.

Because of the large pool of maintenance personnel available in TVA, individual exposures will be minimized by distributing workloads and assignments. During normal operation and maintenance, no employee would be allowed to receive greater than 5 rem in any one year.

ESTIMATE OF PERSONNEL INTERNAL EXPOSURE

The anticipated maximum internal exposure to an employee is expected to be far below the AEC regulatory limits as stipulated in 10 C.F.R. Part 20. This is accomplished by controlling ventilation air in the buildings such that the supplied clean air is exhausted from clean areas to the more contaminated atmospheres. Therefore, those areas normally occupied by personnel are supplied with fresh outside air and would contain very small quantities of radioactive material. For work in contaminated atmospheres, protective clothing and respiratory protection will be required for any work where the MPC_a for a 40-hour week or any sizeable fraction thereof would be exceeded for all isotopes except tritium. Because of the low respiratory protective factors associated with work in tritium atmospheres (2 for most respiratory equipment), exposures will be controlled primarily by limited occupancy times. Where practicable, positive air-supplied plastic suits will be worn against tritium uptake. In summary, internal exposures to employees will be limited by:

1. Maintaining positive control of ventilation air in contaminated work areas,
2. Surveying work areas to identify the type and concentration of radioactive materials present in breathing zones of workers,
3. Providing and requiring the use of special respiratory equipment and protective clothing, and
4. Conducting periodic and special bioassays and whole body counts on employees who work in contaminated atmospheres.

Continuing records will be maintained on internal exposures accrued by plant employees and will be made a part of the employees' overall radioactivity exposure history.