Safety Evaluation Report

NUREG-0013
Suppl. No. 3

U. S. Nuclear Regulatory Commission

related to construction of

Pebble Springs Nuclear Plant Units 1 and 2

Portland General Electric Company

Supplement No. 3

Office of Nuclear Reactor Regulation

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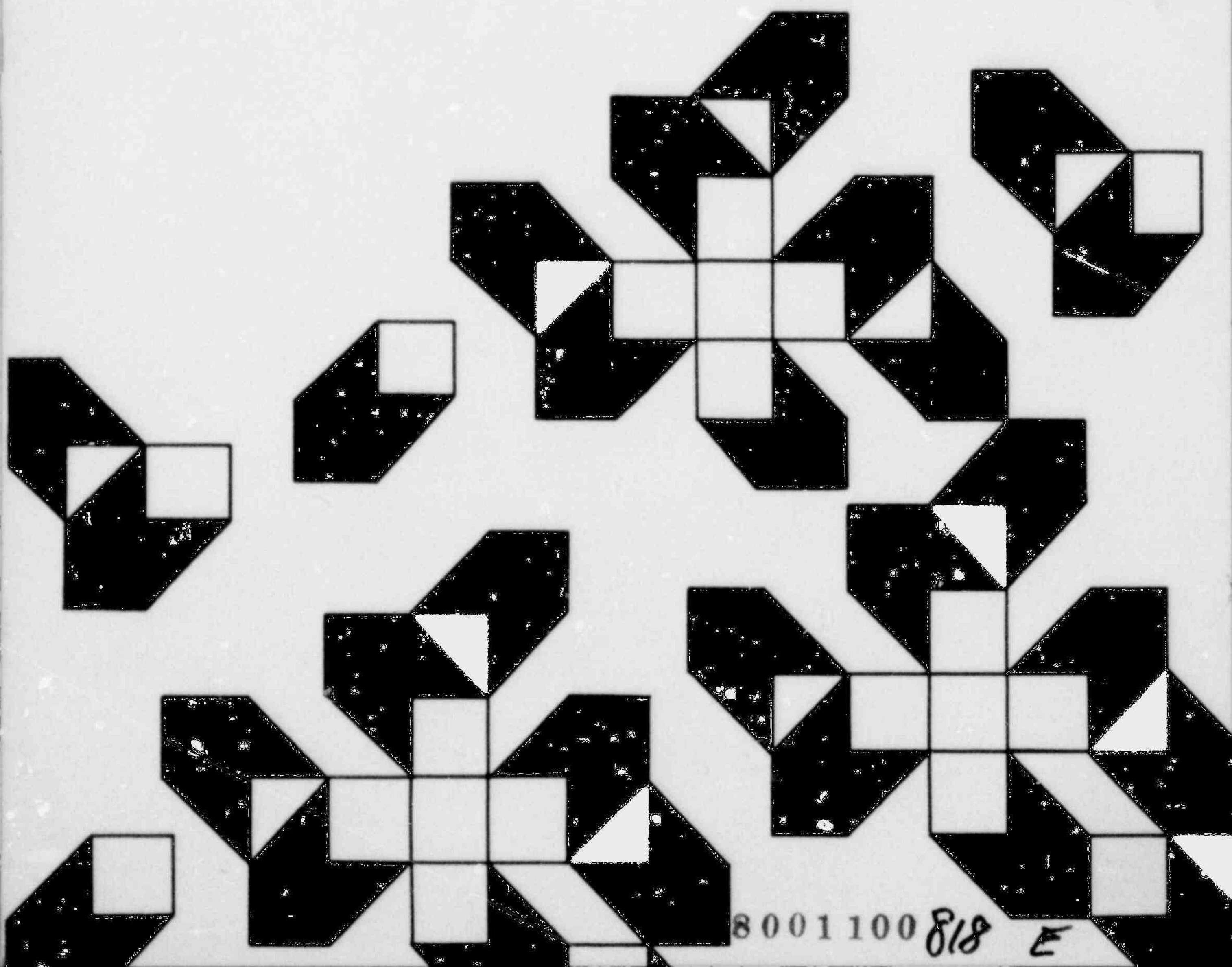

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SUPPLEMENT NO. 3
TO THE
SAFETY EVALUATION REPORT
BY THE
OFFICE OF NUCLEAR REACTOR REGULATION
U. S. NUCLEAR REGULATORY COMMISSION
IN THE MATTER OF
PORTLAND GENERAL ELECTRIC COMPANY
PEBBLE SPRINGS NUCLEAR PLANT, UNITS 1 AND 2
DOCKET NOS. 50-514 AND 50-515

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1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Nuclear Regulatory Commission's Safety Evaluation Report on the matter of Portland General Electric Company's application to construct and operate the proposed Pebble Springs Nuclear Plant, Units 1 and 2, was published in January 1976. The Safety Evaluation Report was based on information in the Preliminary Safety Analysis Report, supplemented by Amendments 1 through 8. As a result of further discussions with the applicant, a number of the issues identified in the Safety Evaluation Report as Outstanding Issues (Section 1.8) and Staff Positions with which the Applicant Does Not Presently Agree (Section 1.9) were resolved and reported in Safety Evaluation Report, Supplements 1 and 2, dated January 1976 and February 1976, respectively. In February 1976, we met with the Advisory Committee on Reactor Safeguards. Discussions on the geology and seismology aspects of the Pebble Springs site were deferred pending the resolution of several unresolved items that had been raise by the U.S. Geological Survey. The Committee report is attached as Appendix D to this supplement.

This supplement does not cover the geological and seismological aspects of the site. Since the publication of the Safety Evaluation Report and the Advisory Committee on Reactor Safeguards meeting, the generic issue of the 1872 earthquake has delayed the resolution of the geology and seismology of the Pebble Springs Nuclear Plant. Resolution will be reported in a future supplement to the Safety Evaluation Report.

This supplement discusses (1) resolution of a number of issues identified in the Safety Evaluation Report and not resolved in Supplements 1 and 2, (2) progress on the remaining issues as well as evaluations on fire protection, postulated fuel handling accident and turbine missiles, and (3) the Advisory Committee on Reactor Safeguards generic items and specific matters related to the Pebble Springs application.

Except for the Appendices, each of the following sections of this supplement is numbered the same as the corresponding sections of its Safety Evaluation Report. This supplement is an addition to and not in lieu of the discussion in the Safety Evaluation Report and Supplements 1 and 2.

1.8 Outstanding Issues

In the Safety Evaluation Report, we identified 10 issues in Section 1.8 for which we required additional information prior to proceeding to a public safety hearing to confirm that the proposed design will meet our requirements. The first three items that remain from the Safety Evaluation Report listing are summarized below and are discussed in the indicated sections of this supplement. Items 4 and 5 are new outstanding issues generated by additional analysis that show more protection is needed from turbine missiles; and further design changes are required in the containment exhaust system based on the results of the fuel handling accident analysis inside containment.

(1) Containment monitoring system to meet the single failure criteria (Section 7.3.10).

(2) Design of the decay heat removal system (Section 7.4.1).

(3) Evaluation of financial qualifications of applicant (Section 20.0).

(4) Turbine missiles (Section 3.5.1.2).

(5) Fuel handling accident (Section 15.8).

In addition to the above items, we are reevaluating the Pebble Springs Nuclear Plant nuclear steam supply system in view of numerous design features that are currently incorporated in or being considered for facilities of similar design. The results of our evaluation will be reported in a future supplement.

1.9 Staff Positions With Which the Applicant Does Not Presently Agree

In the Safety Evaluation Report, we identified eight items in Section 1.9 for which the staff required certain positions or acceptable alternatives be used in the design of the facility in the areas noted. Seven of the listed items were resolved in Supplement 2. The last item, the design basis for volcanic ashfall at the site, is discussed in this supplement (Section 2.5). This matter has been satisfactorily resolved.

1.10 ACRS Generic Items

The Advisory Committee on Reactor Safeguards periodically issues a report listing various generic safety-related matters applicable to light water reactors. Our discussion of these matters is provided in Appendix C to this report which includes references to sections of this report where more specific discussions of the status of the generic matters in relation to Pebble Springs design are presented.

2.0 SITE CHARACTERISTICS

2.3.5 Long-Term (Routine) Diffusion Estimates

In the Safety Evaluation Report, estimates of average atmosphere diffusion conditions were made. Further refinements were made of these estimates which were used in evaluating the radioactive waste management system and are presented below.

Using the onsite meteorological data collected by Portland General Electric Company, the staff has made reasonable estimates of average atmospheric dispersion conditions for the Pebble Springs site using an atmospheric dispersion model appropriate for long-term releases (Sagendorf and Goll, 1976). The model used by the staff is based on the "Straight-Line Trajectory Model" described in Regulatory Guide 1.111. As suggested in Regulatory Guide 1.111, we used vertical dispersion parameters based on atmospheric diffusion data developed at the National Reactor Testing Station, Idaho; these exhibit the decreased vertical dispersion encountered in desert climates (Yanskey, et al, 1966). In conformance with the criteria recommended in Regulatory Guide 1.111, releases from the turbine building vent and ejector exhaust were considered as ground-level, while releases through the containment vent were considered as partially elevated. The calculations also included considerations of intermittent releases during more adverse atmospheric dispersion conditions than indicated by an annual average calculation. Radioactive decay of effluents and deletion of the effluent plume were also considered as described in Regulatory Guide 1.111. We also included an estimate of maximum increase in calculated relative concentration (X/Q) and deposition (D/Q) due to the spatial and temporal variations of the airflow not considered in the straight-line model (as discussed in Regulatory Guide 1.111).

Table 2.3.5.1 contains pertinent X/Q and D/Q values which were used in Section 11.

2.5 Geology, Seismology, and Geotechnical Engineering

In the Safety Evlaution Report, we stated that the design basis is unresolved for volcanic ashfall at the site. The applicant postulated that a "worst case" ashfall would result in an accumulation of 6.5 inches of ash at the site. Based on advice from our consultant, the United States Geological Survey, our analysis showed that postulated volcanic eruption, modeled after the 1912 Katmai eruption, would produce 12.0 inches of ashfall at the site.

TABLE 2.3.5.1

RELATIVE CONCENTRATION (X/Q) AND DEPOSITION (D/Q) VALUES CALCULATED FOR THE PEBBLE SPRINGS NUCLEAR PLANT

Receptor Distance-Direct‹‹ (mi;es)	Source	X/Q No decay, undepleted (seconds per cubic meter)	D/Q (per square meter)
1.24 West-Southwest	A	2.8×10^{-6}	8.3×10^{-9}
	B	7.2×10^{-7}	4.9×10^{-9}
	C	2.8×10^{-6}	1.9×10^{-8}
	D	1.8×10^{-6}	1.6×10^{-8}
1.6 West	A	1.8×10^{-6}	2.8×10^{-9}
	B	2.5×10^{-7}	1.2×10^{-9}
	C	1.3×10^{-6}	6.5×10^{-9}
	D	9.7×10^{-7}	7.1×10^{-9}
3.1 South-Southwest	A	3.2×10^{-7}	3.2×10^{-10}
	B	1.0×10^{-7}	3.8×10^{-10}
	C	8.8×10^{-7}	3.3×10^{-10}
	D	8.7×10^{-7}	4.5×10^{-9}

NOTE: A - Turbine Building Vent/Air Ejector Exhaust
B - Containment Vent - Continuous Release
C - Containment Vent - Waste Decay Decay Tank Purge
D - Containment Vent - Containment Purge

The applicant reevaluated the potential ashfall at the site and developed other necessary data for design purposes. The results of this effort are reported in "Potential for Volcanic Ash Fall at Pebble Springs Nuclear Plant Site" (Revision 1, May 17, 1976). Based on our review and that of the U.S. Geological Survey, our position was accepted by the applicant for designing the plant to the following conditions:

1. Grain size distribution of the volcanic ash at the site shall be modeled in accordance with the data in Figure 10 of the Volcanic Hazards Study report.

2. Rate of ashfall shall be modeled generally in accordance with the 1912 Katmai eruption, assuming a maximum rate of 0.5 inches per hour for 9 hours, and a total accumulation of 8.5 inches of fresh loose ash. The Katmai eruption averaged about 0.44 inches per hour for approximately 9 hours. We have determined that a maximum rate of 0.5 inches per hour is a reasonable, conservative rate for design purposes. The maximum ashfall is based on recent work by the U.S. Geological Survey at Mount Saint Helens in which they measured 8.0 inches, 62 miles along the axial trace of the plume; and 2.0 inches, 174 miles along the near axial trace. When these data are applied to Figure 13 of the Volcanic Ash Fall Report and the upper bound curve reconstructed, the total compacted thickness at the site is about 5.5 inches. Applying a 35 percent compaction factor, as recommended by the U.S. Geological Survey, a total of 8.5 inches of loose ash would accumulate at the Pebble Springs site.

3. Acidity (pH) of the ultimate heat sink and reservoir water is to be determined by using the 8.5 inches of accumulated ash in conjunction with Figure 11 of the Volcanic Ash Fall Report. Further information will be needed in the Final Safety Analysis Report to justify the use of the buffered curves for determining the pH in the heat sink and reservoir water.

4. Steps must be taken to minimize the drift of volcanic ash. Drifting of volcanic ashfall at the site may occur from high winds during and after the postulated volcanic eruption. Consequently, steps must be taken to protect safety-related equipment and structures from this possibility. The applicant is required to factor this matter into the plant design and to develop a contingency plan for mitigating the consequences of drifting volcanic ash.

We conclude that the above conditions will provide a conservative basis for designing a plant at the Pebble Springs site for volcanic ashfall. We will review the plant design and appropriate procedures prior to the issuance of an operating license for this facility to assure compliance with its above cited conditions.

3.0 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.5.1.2 Turbine Missiles

Introduction

The turbine generators for Pebble Springs Nuclear Plant are in a nonpeninsular orientation with respect to the reactor containment. In the Safety Evaluation Report, it was stated that the applicant's commitment to provide a missile shield cover around the low pressure tubine stages was an acceptable solution, if it was determined that additional protection was necessary.

The applicant submitted a report giving an analysis of the turbine missiles hazards for the Pebble Springs Nuclear Plant supporting his determination that additional protection was not necessary. This analysis made use of a computer model which uses combined Monte Carlo and deterministic methods to track individual turbine missiles. Missiles of various sizes were selected and their speeds and directions were sampled from uniformly distributed parametric ranges. The missiles then were tracked through a mock-up of the plant geometry. Missile penetrations through barriers were calculated using the Ballistic Research Laboratories formulae and hits upon safety-related targets were recorded and statistically analyzed for a large number of missile trajectories. The applicant defined unacceptable damage to be a strike upon both redundant trains of a particular safety system by the same missile. Using these techniques, the applicant determined that the probability of unacceptable damage by turbine missiles was less than 10^{-7} per year, and concluded, therefore, that no additional protection against turbine missiles was required.

We have reviewed and evaluated the applicant's analysis. While we believe that a combined stochastic-deterministic computer simulation using a mock-up of the plant safety systems in a realistic geometry can prove to be very useful, the assumptions used can strongly influence the results obtained from use of such simulations. For example, the modeling used by the applicant was derived from particle tracking schemes and treats the missile as a point particle. This type of simulation may give rise to misleading results when applied to missiles of a finite size because dimensionless point particles can be recorded as misses whereas an actual missile could cause damage even if the geometric center of the missile does not intersect the target. In particular, the finite size of a missile can lead to situations where redundant trains of a system in relatively close proximity to each other could be damaged by the same missile. This limitation was not addressed in the applicant's analysis. In addition, the current barrier penetration and spallation equations used by the staff employs the modified National Resources Defense Council formulae. These are more conservative (predict greater penetration) for the typical velocity ranges associated with turbine missiles than the Ballistic Research Laboratories equations used by the applicant.

We have performed an independent analysis of the probability of damage from turbine missiles. We considered horizontal and elevation angles subtended by a target in determining the probability of a missile striking it, and used the modified National Resources Defense Council equations for barrier penetrations and interactions. Finite missile size effects were considered with respect to redundant systems in close proximity to each other. Our analysis and the results and recommendations are described below.

Analysis

Turbine Failure Probability

The low trajectory turbine missile risk assessment is based on a turbine failure at destructive overspeed since this represents the limiting case for potential damage of safety-related systems. The destructive overspeed turbine failure probability is taken to be 4×10^{-5} per turbine year on the basis of Bush's study (Nuclear Safety, Vol. 14, No.3, May-June 1973).

Missile Strike Probabilities

The strike probabilities with respect to the safety-related systems are based on the following assumptions:

(1) The turbine failure is limited to the bursting of a single low pressure turbine disc.

(2) Any one of the 42 low pressure turbine discs is equally likely to burst in the event of a destructive overspeed.

(3) The inner discs are constrained to ±5 degrees greater than or equal to the ejection angles with respect to the plane of rotation of the failed disc.

(4) The end discs are constrained to 0 degree - 25 degree ejection angles with respect to the plane of rotation of the failed disc.

(5) The missiles are assumed to be uniformly distributed within the ejection angle constraints of items (3) and (4), as well as in the angular direction about the turbine shaft.

(6) The missiles are assumed to be uniformly distributed in speed within the exit speed ranges presented by the applicant.

(7) The missiles are assumed to travel in straight line trajectories.

(8) High trajectory missile strike probabilities are a negligible contributor to the total turbine missile damage probability.

The strike probabilities were estimated by evaluating horizontal (θ) and elevation (ϕ) angles subtended by a target with respect to each turbine disc. The separate probability components P_θ and P_ϕ were used to calculate the total strike probability P_2 for each target, that is,

$$P_2 = P_\phi \cdot \frac{\sum_{i=1}^{N} P_{\theta,\, i}}{N}$$

where N is the total number of low pressure discs (N=42). The individual components P_θ and P_ϕ were obtained by ratioing the respective angles θ and ϕ subtended by the target to the total "allowed" angles in the horizontal and elevation planes.

Barrier Damage Probabilities

The analysis was factored in the primary concrete barriers (e.g., containment wall, steam line penetration enclosure). All other potential missile interactions, such as with the moisture reheater units, were neglected. The probability estimates for intermediate concrete barriers between the target and the turbine were evaluated on the basis of perforation. Spalling was the limiting condition used in evaluating the final barrier protecting a safety-related target.

The perforation or spalling probabilities P_3 were obtained by considering the distributed values of missile speeds and orientations. It was assumed that the missile speeds were distributed uniformly within the estimated ranges (assumption f in B.2). Also, the missile was assumed to be rotating within the plane of the disc, thus presenting a random impact area with respect to the barrier. These two distributions, speed and orientation, define a population of missile states which are available for interaction with a barrier. Only some portion of this population is capable of perforation or spalling. This subset of the population was determined by applying the modified National Resources Defense Council equations for perforation and spalling. The ratio of this subset to the total population was equated with the probability for damaging the barrier either through perforation or spalling.

Safety-Related Systems As Missile Targets

A review of the applicant's submitted turbine missile strike and damage probability analysis, including the plant structural features and relative locations of the safety-related equipment, indicated that the primary coolant system boundary within the containment and the steam line and feedwater penetration enclosure were the only safety-related targets that had significant "exposure" to low trajectory turbine missiles. Other plant areas, such as the control room, cable spreading rooms, spent fuel pool, and electrical penetration area will be sufficiently protected by a combination of relative location (such that none or a small fraction of the 42 low pressure discs could reach them) and redundancy and separation (so that a single missile would not be able to compromise both redundant systems).

The missile targets within the containment were defined as the steam generators and the areas within their concrete enclosures. Missiles which were postulated to enter the containment above the top of the steam generators were assumed to fall down randomly in free fall. The strike probability was obtained by ratioing the total steam generator enclosure plan area to the containment plan area. The reactor vessel enclosure was not included since it is covered by a missile shield.

With respect to the steam line and feedwater penetration area, due to the physical dimensions of the postulated missiles, missile impact in the vicinity of the 18-inch partition wall separating Train A and Train B piping was assumed to compromise both trains. The target area on the penetration enclosure wall facing the turbine was sized in accordance with the dimensions of the largest missile. This aspect is contrary to the applicant's analysis, wherein the missile transport code tracks missiles as if they were point particles. Point representation of missiles is misleading, in that it excludes the simultaneous damage of Train A and Train B steam lines or feedwater lines unless the missile trajectory is not parallel to the 18-inch partition wall.

Total Probability for Compromising Safety

The total probability P was obtained by multiplying the individual components, so that

$$P = P_1 \, P_2 \, P_3$$

where P_1, P_2, and P_3 were described previously.

Conservative Versus Realistic Considerations

The results presented are separated into conservative and realistic estimates. With respect to the penetration enclosure area, the conservative analysis was based on the following assumptions:

(1) The missile sizes are such that penetration of any portion of the enclosure wall facing the turbine would cause damage to the 18-inch partition wall and thus lead to damage of both steam line or feedwater pipe trains. This assumption affects the value of P_2.

(2) Design and destructive overspeed missiles are equally capable of perforating the enclosure, so that $P_1 = 10^{-4}$ per turbine year and P_3 is assumed to be 1.0.

Realistically, unless the missile struck fairly close to the 18-inch partition wall inside the enclosure, compromise of both pipe trains is not envisioned. Thus, the conservatively estimated value of P_2 would be reduced by a factor of about two. Also, not all design overspeed turbine missiles will perforate the enclosure. This was accounted for in the realistic analysis by reducing the value of P_3 by a factor of three.

Results and Recommendations

Our evaluation of the turbine missile damage probabilities for the proposed Pebble Springs Nuclear Plant included a quantitative strike and damage probability analysis. The results of the analysis are summarized below.

Target	Probability for Unacceptable Damage by Turbine Missiles, Per Turbine Year	
	Conservative	Realistic
Steam Line and Feedwater Penetration Area	6.2×10^{-6}	1.0×10^{-6}
Primary System Pressure Boundary Inside the Containment	1.2×10^{-6}	2.7×10^{-7}
Control Room	1.7×10^{-6}	1.7×10^{-7}
Total	9.1×10^{-6}	1.5×10^{-6}

As indicated by the estimated probabilities, the main steam line and feedwater penetration area (situated betwen the containment and the turbogenerator) is the principal contributor to the overall risk from potential turbine missiles. Protection of this area from potential turbine missiles would reduce the overall risk substantially. Provision of missile shields above the steam generator enclosures within the containment would reduce the turbine missile risks for the remaining contributors. Implementation of the vendor recommended turbine steam valve testing procedures and frequencies would also reduce the turbine missile risks for all the contributors. We conclude that additional protection is required. We will require that the applicant reaffirm his commitment to provide a missile shield cover around the low pressure turbine stages, or submit specific plans for achieving the protection features discussed above. We will report on this matter in a future supplement.

7.0 INSTRUMENTATION AND CONTROLS

7.3.2 Engineered Safety Features Actuation System Interchannel Isolation and Devices

In the Safety Evaluation Report, we stated that since the outputs from the three analog subsystems supply the inputs to each of the digital subsystems, and a single failure at any point in the interface could propagate to render the engineered safety features actuation system digital subsystems inoperable, the isolation devices in this design have not been demonstrated to be acceptable.

The applicant has agreed to modify the design and to qualify each interchannel isolation device so that the maximum credible fault energy applied to either side of the isolation device will not propagate to the other side of the isolation device. We find this modified design acceptable.

7.3.8 Main Steam Line Isolation

In the Safety Evaluation Report, we stated that the staff is evaluating the design of this item and we would report on this matter in a supplement to the report.

There are two main steam lines from each steam generator with one main steam line isolation valve in each line. Each main steam isolation valve will be equipped with an independent pneumatic actuation system and actuated by redundant engineered safety features actuation system signals. The steam line break instrumentation of the engineered safety features actuation system also initiates feedwater isolation and actuates the auxiliary feedwater system coincident with the main steam isolation. In addition, redundant engineered safety features actuation system signals will initiate a turbine trip through buffered contacts. The engineered safety features actuation system signals interface with the electrohydraulic control system, which initiates turbine stop valve closing.

Based on our review, the applicant's instrumentation and control design for main steam line isolation function meets the requirements in the Standard Review Plan, Section 7.3, Appendix A.

7.3.9 Main Feedwater Line Isolation

In the Safety Evaluation Report, we stated that we were evaluating this isolation system design in conjunction with the steam line break accident, and we would report on this matter in a supplement to the report.

The main feedwater isolation system consists of two feedwater lines with a seismic Category I isolation valve in each line and a seismic category I check valve inside the containment. Main steam isolation and initiation of auxiliary feedwater flow occur coincident with feedwater isolation. Each feedwater isolation valve is actuated by redundant engineered safety features actuation system signals in the following manner: each engineered safety features actuation system signal controls a pair of solenoid-actuated pneumatic control valves; each pai5 is redundant to the other. The instrumentation and controls through the solenoid-actuated pneumatic control valves meet the single failure criterion.

The applicant's design also supplies redundant engineered safety features actuation system signals to the main feedwater control valves, bypass control valves and the main feedwater pumps in order to close the control valves and to trip the pumps. The applicant has stated that this Class IE trip signal went from seismic to non-seismic area with adequate signal isolation to prevent seismic disruption of engineered safety features actuation system.

The applicant's instrumentation and control design for the main feedwater line isolation meets the requirements in the Standard Review Plan, Section 7.3, Appendix A.

The staff finds this design acceptable.

7.3.10 Containment Monitoring System

In the Safety Evaluation Report, we stated that we were assessing the radiological dose significance of failure to isolate the purge system under accident conditions. We noted that a single failure in the monitoring and interlock system can prevent isolation.

The containment monitoring system consists of seismic Category I detectors which continuously monitor the gaseous iodine and air particulate activity levels in the containment atmosphere during normal plant operation; they will monitor the gaseous iodine and air particulate activity levels of the containment purge exhaust flow during containment purge operations. During containment purging the receipt of a high radiation alarm signal from the radiation detector will automatically de-energize a supply damper and an exhaust damper to isolate the purge lines.

As noted in Section 15.8 of this supplement, the applicant has agreed to modify the current design to assure prompt isolation and detection. The need for redundancy in the monitoring system will be considered in the modified design. The resolution of this matter will be reported in a supplement to the Safety Evaluation Report.

7.4.1 Decay Heat Removal System

In the Safety Evaluation Report, we stated that the motor-operated valves used to isolate the decay heat removal systems from the reactor coolant system do not meet our criteria as described in the Safety Evaluation Report. Additional information was provided since the publication of the Safety Evaluation Report to further justify the applicant's position.

The decay heat removal system suction valve interlock will protect the low pressure decay heat removal system from excessive pressure when the reactor coolant pressure exceeds 675 pounds per square inch gage. There are a total of four valves, two in series in each of the two suction lines. The two valves upstream will be interlocked by reactor coolant system pressure from the engineered safety features actuation system Channels A and B, and the remaining two valves are interlocked by pressurizer pressure from Channels A and B. Though redundancy and diversity are incorporated into the design, it does not meet the single failure criterion because of power supply assignment to valve motor operators. The two series motor-operated valves on one line are supplied 480-volt power from Load Group I while the series valves on the other line are supplied power from Load Group II. With this configuration, the system can not be isolated, given a single failure in any one of the power trains. Our position continues to be that the single failure criterion must be satisfied for the isolation function. We will report the resolution of this item in a supplement to this report.

7.4.2 Essential Control and Instrumentation Systems

In the Safety Evaluation Report, we stated that additional information was needed to complete our review of this system.

The essential control and instrumentation system is a collection of Class IE controls and indications which serve the following functions:

(1) To provide post-accident monitoring indication.

(2) To bring the reactor to a safe hot shutdown from inside or outside of the control room.

The essential control and instrumentation system is designed to meet the applicable criteria and IEEE Std 308-1971 for such functions as are required by General Design Criteria 19 and 34. Since the essential control and instrumentation system consists of manual and automatic controls for the auxiliary feedwater system, makeup, high pressure injection, and boron water storage tank suction valve controls, the applicant will ensure that an engineered safety features actuation system signal will override the essential control and instrumentation controls for these engineered safety features.

The applicant has stated that the essential control and instrumentation system has always included the post-accident monitoring indicators as non-protective safety-related Class IE indicators. Preliminary Safety Analysis Report Section 7.5, Table 7.5-1 of the Preliminary Safety Analysis Report, lists the post-accident monitoring instrumentation which indicates redundant sensors with at least one channel recorded. The power supplies for those instrumentations will be supplied from vital instrument buses which are from onsite emergency power supplies. The applicant has stated that essential control and instrumentation system complies with IEEE Std 379-1971 and IEEE Std 344-1971; in addition, the transmitters in the containment will be qualified to withstand the post-accident atmosphere in the containment.

The staff finds the proposed essential control and instrumentation system acceptable.

8.0 ELECTRIC POWER SYSTEMS

8.3 Onsite Power System

8.3.1 Alternating Current Power System

In the Safety Evaluation Report, we stated that additional information was required to determine if the proposed system design was acceptable.

Onsite standby power will be supplied by two diesel generators per unit. Each diesel will supply a 4.16 kilovolt emergency bus. Only one of the two diesels is required to provide emergency power for accident conditions.

The redundant engineered safety features and vital instrumentation and control loads are supplied from two 4.16 kilovolt emergency buses. This configuration is maintained through the alternating current and direct current subsystems. There is no automatic switching between redundant buses, and interlocks which satisfy the intent of Regulatory Guide 1.6 are provided. The applicant has identified cases where a third makeup pump, engineered safety features equipment room chiller, and component cooling water pump may be powered from either vital bus. The third pump will be mechanically interlocked to permit only one breaker to be closed at one time. Also, additional interlocks will be provided to prevent more than one of each type of pump from being connected to either emergency diesel generator at the same time.

The applicant has provided supplementary information which clarifies the sequencing of the third pump on the engineered safety features bus and show a method for assuring power lockout to the unused pump which meets the single failure criterion. In addition, the third equipment room chiller has been removed from the design. With these clarifications, the staff finds the design of the manual transfer scheme acceptable.

8.3.3 Physical Independence of Electrical Systems (Regulatory Guide 1.75)

In the Safety Evaluation Report, we requested additional information to determine the physical independence of the electrical systems.

The applicant has documented complete conformance to the provisions of Regulatory Guide 1.75. For isolation devices, the applicant has proposed using one circuit breaker between the Class IE system and the non-Class IE electrical equipment. This Class IE circuit breaker will be tripped by fault current sensing relays and an engineered safety features actuation system signal. In addition, the applicant has stated that there will be no associated circuits which do not conform to Regulatory Guide 1.75. The applicant has stated that if such circuits do become necessary, they will be identified as such in the Final Safety Analysis Report and an analysis or test will be conducted in accordance with Regulatory Guide 1.75. The applicant has committed to the identification requirements of Regulatory Guide 1.75. We conclude that this design is acceptable.

The applicant has indicated that routing redundant Class IE cables to remote structures will be accomplished through separate and independent seismic Category I duct banks (solid concrete structures with holes or ducts for cables) and manholes. In duct banks, non-Class IE cables and Class IE cables will be separated by installing them in separate ducts. In the manholes, Class IE and non-Class IE cable separation will be maintained by (1) routing Class IE and non-Class IE cables through rigid metal conduit, either inside or outside the manhole. The applicant has provided additional information conforming to the design basis of IEEE Std 308-1971, and stated that all duct banks and manholes containing Class IE circuits will be seismic Category I. The ducts will be sealed watertight or are sloped to drain. As an added precaution, all control and power cables less than 600 volts will be tested and qualified for operation under submerged conditions. All 5 and 15 kilovolt power cables will be specified as ethylene-propylene rubber insulated and has low moisture absorption and excellent di-electric stability while continuously submerged in water. On the basis of the duct and manhole design and cable qualification, we find the design acceptable.

We conclude that the alternating current power system design meets General Design Criteria 17 and 18 and Regulatory Guides 1.6 and 1.9, and is acceptable.

8.3.4 Direct Current Power Systems

In the Safety Evaluation Report, we stated that additional information was required to show that the direct current power system can supply all the load requirements.

Each direct current power subsystem will be located in separate rooms within a seismic Category I building. The ventilation for each room will be separate from that of the others. A backup vital battery charger is provided for each of two engineered safety features load group to supply the 125-volt vital direct current bus in the event either vital battery charger should fail. Interlocks are provided to prevent: (1) the backup vital battery charger from feeding two 125-volt vital direct current buses simultaneously and (2) both backup and primary vital battery chargers from feeding the same 125-volt vital direct current bus. Each vital battery is sized to supply the engineered safety features load requirements for two hours without vital battery charger support. The capacity of each vital battery charger is based on (1) the largest combined demands of all the various steady state loads, and (2) the charging capacity required to restore the vital battery from the design minimum charge state to the fully charged state in 12 hours under any plant operating conditions. In addition, the battery chargers are fed from the Class IE 480-volt emergency bus of the same power channel as the bus and Class IE battery to which it is connected.

Four redundant 120-volt alternating current vital bus systems will be provided to supply power to plant protection system instrumentation and related circuits, engineered safety features actuation System instrumentation, and related circuits. Each vital bus will be fed from an independent static inverter which in turn normally will be fed through a static battery charger from a 480-volt bus. Should the normal power source fail, the static inverter automatically will be powered from its associated battery. The 120-volt alternating current vital bus systems will be designed in accordance with IEEE Std 308-1971.

We have reviewed the design description, design criteria, design bases, and single line diagrams for the direct current onsite power system and the 120-volt alternating current vital bus system and the analysis regarding the adequacy of these criteria and bases. We conclude that the proposed design for the direct current onsite power and 120-volt alternating current vital bus system meets our requirements and, therefore, is acceptable.

9.0 AUXILIARY SYSTEMS

9.5.1 Fire Protection System

The applicant has responded to the new guidelines stated in Appendix A of our Branch Technical Position APCSB 9.5-1, "Guidelines on Fire Protection for Nuclear Power Plants," and their response is currently under review. We will review the evaluation along with revised design features of the fire protection system and provide the applicant with the results of our evaluation on a timely basis so that they can be effectively incorporated into the final design. The design as presently proposed meets General Design Criterion 3, "Fire Protection," and applicable guidelines in effect prior to issuance of Branch Technical Position APCSB 9.5-1. For the construction permit stage of the review, we find it acceptable. Final approval of the system will depend on the review of the applicant's submittal which will be completed after a decision on the issuance of the construction permit. However, based upon our current review of the facility, sufficient flexibility exists in the design to allow implementation of any design changes that may be necessary to assure compliance with Appendix A to Branch Technical Position 9.5-1.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Introduction

In the Safety Evaluation Report we indicated that a detailed assessment to determine conformance with Appendix I to 10 CFR Part 50 would be provided in a supplement. Section 50.34a of 10 CFR Part 50 requires releases of radioactive materials in liquid and gaseous effluents from nuclear power reactors to be "as low as is reasonably achievable". This term is defined in Section 50.34a to mean: As low as is reasonably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations, and in relation to the utilization of atomic energy in the public interest.

On September 4, 1975, the Commission amended Appendix I of 10 CFR Part 50 to provide persons who have filed applications for construction permits for light water cooled nuclear power reactors which were docketed on or after January 2, 1971, and prior to June 4, 1976, the option of dispensing with the cost-benefit analysis required by Paragraph II.D of Appendix I. This option permits an applicant to design the radwaste management systems to satisfy the Guides on Design Objectives for Light-Water-Cooled Nuclear Power Reactors proposed in the Concluding Statement of Position of the Regulatory Staff in Rule Making Hearing Docket RM-50-2, dated February 20, 1974. As indicated in the Statement of Considerations included with the amendment, the Commission noted it is unlikely that further reductions to radioactive material releases would be warranted on a cost-benefit basis for light water cooled nuclear power reactors having radwaste systems and equipment determined to be acceptable under the proposed staff design objectives set forth in RM-50-2.

In a letter to the Commission, dated September 24, 1975, Portland General Electric Company chose to comply with the Commission's September 4, 1975 amendment to Appendix I, eliminating the necessity of performing a cost-benefit analysis as required by Paragraph II.D of Appendix I.

11.2 Evaluation

We have evaluated the radioactive waste management systems for reducing the quantities of radioactive materials released to the environment in liquid and gaseous effluents. These systems have been previously described in Section 3.5 of the Final Environmental Statement, dated April 1975, and in Chapter 11 of the Safety Evaluation Report.

Based on more recent operating data applicable to the Pebble Springs Nuclear Plant, and on changes in our calculational model, we have generated liquid and gaseous source terms to determine conformance with Appendix I.

The source terms, shown in Tables 11.1 and 11.2, were calculated using the models and methodology described in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," April 1976. These source terms were used to calculate the doses as described below. The dispersion of radionuclides in and the deposition of radionuclides from the atmosphere were based on analyses performed by the NRC staff for this evaluation and are discussed in Section 2.3.5.

The mathematical models used to perform the dose calculations are contained in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents." Included in our analysis are dose evalutions of three effluent categories: (1) pathways associated with liquid effluent releases to the Pebble Springs cooling reservoir, (2) noble gases released to the atmosphere, and (3) pathways associated with radioiodines, particulates, carbon-14 and tritium released to the atmosphere. The dose evaluation pathways associated with liquid effluents were based on the maximum exposed individual. Since the reservoir water will not be used for domestic drinking water outside the restricted area or for recreational activities, these pathways are not discussed in this supplement. However, the applicant has an agreement with Kreb Brothers Ranch to supply reservoir water for use in watering livestock and irrigation of supplemental feed crops. As a result, we have estimates of the potential radiation dose commitments from these uses in this supplement. In the Final Environmental Statement, the staff provided estimates of dose commitments from ingestion of beef and vegetation produced by irrigation with reservoir water. These pathways have also been addressed in this supplement utilizing the most recent Commission source terms and dose models. For these dose assessments, it was assumed that the critical radionuclides (tritium, cesium-134 and cesium-137) were in equilibrium in the reservoir water, and that there would be no significant dilution of the liquid releases prior to agricultural use of the water. The maximum dose commitment (both units operating) resulting from ingestion of beef, milk, and leafy vegetables affected by use of reservoir water was estimated to be 0.061 mrem per year (total body) and 0.22 mrem per year (thyroid) for an infant.

The dose evalution of noble gases released to the atmosphere included a calculation of beta and gamma air doses at the site boundary and total body and skin doses at the residence having the highest dose. The maximum air doses at the site boundary were found at 1.24 miles west-southwest relative to Pebble Springs station. The location of maximum total body and skin doses were determined to be at a residence at 1.60 miles west of the Pebble Springs station.

TABLE 11.1

CALCULATED RELEASES OF RADIOACTIVE MATERIAL
IN LIQUID EFFLUENTS FROM
PEBBLE SPRINGS NUCLEAR PLANT, UNITS 1 AND 2

Nuclide	Curies per year per reactor	Nuclide	Curies per year per reactor
Corrosion & Activation Products		Te-129	1.6(-4)
		I-130	1.1(-4)
Cr-51	3.3(-4)[a, b]	Te-131m	1.8(-4)
Mn-54	6(-5)	Te-131	3(-5)
Fe-55	3(-4)	I-131	4.4(-2)
Fe-59	1.8(-4)	Te-132	3.4(-3)
Co-58	2.9(-3)	I-132	4.1(-3)
Co-60	3.6(-4)	I-133	2.9(-2)
Np-239	1.5(-4)	I-134	3(-5)
		Cs-134	5(-3)
Fission Products		I-135	5.8(-3)
		Cs-136	2.4(-3)
Br-83	4(-5)	Cs-137	3.6(-3)
Rh-86	2(-5)	Ba-137m	3.4(-3)
Sr-89	7(-5)	Ba-140	4(-5)
Sr-91	3(-5)	La-140	3(-5)
Y-91m	2(-5)	Ce-141	1(-5)
Y-91	1(-5)		
Zr-95	1(-5)	All Others	6(-5)
Mo-99	4(-2)		
Tc-99m	2.9(-2)	Total (except H-3)	1.8(-1)
Te-127m	5(-5)		
Te-127	7(-5)	H-3	200
Te-129m	2.5(-4)		

a = Exponential notation; 1(-4) = 1 x 10^{-4}

b = Nuclides whose release rates are less than 10^{-5} Curies per year per reactor are not listed individually, but are in the category "All Others."

TABLE 11.2

CALCULATED RELEASES OF RADIOACTIVE MATERIAL
IN GASEOUS EFFLUENTS FROM
PEBBLE SPRINGS NUCLEAR PLANT, UNITS 1 AND 2

Curies per year per reactor

Radionuclide[d]	Reactor Building	Auxiliary Building	Turbine Building	Air Ejector	Decay Tanks	Total
Kr-85m	2	3	a	2	a	7
Kr-85	320	9	a	6	800	1100
Kr-87	a	1	a	a	a	1
Kr-88	2	5	a	3	a	10
Xe-131m	76	3	a	2	a	81
Xe-133m	48	5	a	3	a	56
Xe-133	8000	470	a	290	a	8,800
Xe-135	13	8	a	5	a	26
Xe-138	a	1	a	a	a	1
I-131	2.5(-3)	6.8(-2)	8.8(-4)	4.3(-2)	a	1.1(-1)
I-133	2.7(-3)	8.5(-2)	1.1(-3)	5.3(-2)	a	1.4(-1)
Mn-54	c	1.8(-2)	c	c	c	1.8(-2)
Fe-59	c	6.0(-3)	c	c	c	6.1(-3)
Co-58	c	6.0(-2)	c	c	c	6.1(-2)
Co-60	c	2.7(-2)	c	c	c	2.7(-2)
Sr-89	c	1.3(-3)	c	c	c	1.3(-3)
Sr-90	c	2.4(-4)	c	c	c	2.4(-4)
Cs-134	c	1.8(-2)	c	c	c	1.8(-2)
Cs-137	c	3.0(-2)	c	c	c	3.0(-2)
H-3	650	650	c	c	c	1300
C-14	1	a	a	a	7	8
Ar-41	25	c	c	c	c	25

a = less than one Curie per year per reactor for noble gases and carbon-14, less than 10^{-4} Curies per year per reactor for iodine.

b = Exponential notation; $1.4(-2) = 1.4 \times 10^{-2}$

c = Less than 1.0 percent of total for this nuclide.

d = Radionuclides not listed are released in quantities less than those specified in notes a and c from all sources.

The dose evaluation of pathways associated with radioiodine, particulates, carbon-14 and tritium released to the atmosphere was also based on the maximum exposed individual. One such individual assumed is an infant whose diet included the consumption of 330 liters per year of milk, produced at the location of the milk cow having the highest calculated dose from these and two other pathways noted below. This location is 3.1 miles south-southwest of the facility. The maximum exposed individual was also exposed to inhaled radionuclides in this category, as well as those deposited on the ground at the location described above.

Since the Guides on Design Objectives apply to all light water cooled reactors at a site, it is necessary to compare the total dose from Units 1 and 2 with the Design Objectives contained in the Concluding Statement of Position of the Regulatory Staff. Table 11.3 provides a comparison of the calculated doses, with the design objectives of Sections II.A, B and C of Appendix I and the proposed NRC staff design objectives set forth in RM-50-2.

As shown in Table 11.1, the expected quantity of radioactive materials released in liquid effluents from Units 1 and 2 will be less than 5 Curies per year per reactor (0.18 Curie per year per reactor), excluding tritium and dissolved gases, in conformance with the amendment to Section II.D. The liquid effluents released from Units 1 and 2 will not result in an annual dose or dose commitment to the total body or to any organ of an individual, in an unrestricted area from all pathways of exposure, in excess of 5 mrem (Table 11.3).

Based on the NRC staff's evaluation of the gaseous radwaste management systems, the total quantity of radioactive materials released in gaseous effluents from Units 1 and 2 will not result in an annual gamma air dose in excess of 10 mrads and a beta air dose in excess of 20 mrads at every location near ground level, at or beyond the site boundary, which could be occupied by individuals (Table 11.3). As shown in Table 11.2, the annual total quantity of iodine-131 released in gaseous effluents will be less than one Curie per reactor (0.11 Curie per year per reactor) in conformance with the amendment to Section II.D and the annual total quantity of radioiodine and radioactive particulates released in gaseous effluents from Units 1 and 2 will not result in an annual dose or dose commitment to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 15 mrem (Table 11.3).

11.3 Conclusion

Our evaluation demonstrates that the estimated doses associated with the normal operation of the Pebble Springs Nuclear Plant, Units 1 and 2, meet the design objectives of Sections II.A, B and C of Appendix I of 10 CFR Part 50, and that the expected quantity of radioactive materials released in liquid and gaseous effluents and the aggregate doses meet the design objectives set forth in RM-50-2.

TABLE 11.3

COMPARISON OF PEBBLE SPRINGS NUCLEAR PLANT, UNITS 1 AND 2 WITH APPENDIX I TO 10 CFR PART 50, SECTIONS II.A, II.B AND II.C (MAY 5, 1975)[a] AND SECTION II.D, ANNEX (SEPTEMBER 4, 1975)[b]

Criterion	Appendix I[a] Design Objectives	Annex[b] Design Objectives[c]	Calculated Doses
Liquid Effluents			
Dose to total body from all pathways (infant)	3 mrem per year per unit	5 mrem per year per site	0.031 mrem per year per unit
Dose to any organ from all pathways (infant thyroid)	10 mrem per year per unit	5 mrem per year per site	0.11 mrem per year per unit
Noble Gas Effluents[d]			
Gamma dose in air	10 mrad per year per unit	10 mrad per year per site	0.20 mrad per year per unit
Beta dose in air	20 mrad per year per unit	20 mrad per year per site	0.66 mrad per year per unit
Dose to total body of an individual	5 mrem per year per unit	5 mrem per year per site	0.12 mrad per year per unit
Dose to skin of an individual	15 mrem per year per unit	15 mrem per year per site	0.40 mrem per year per unit
Radioiodines and Other Radionuclides Released to the Atmosphere[e]			
Dose to any organ from all pathways (Infant-thyroid)	15 mrem per year per unit	15 mrem per year per site	1.06 mrem per year per unit

[b]Federal Register V. 40, p. 40816, September 4, 1975.

[c]Design Objectives given on a site basis. Therefore, these design objectives apply to 2 units at the site.

[d]Limited to noble gases only.

[e]Carbon-14 and tritium have been added to category.

The evaluation shows that the applicant's proposed design of Units 1 and 2 satisfies the criteria specified in the option provided by the Commission's September 4, 1975 amendment to Appendix I and, therefore, meets the requirements of Section II.D of Appendix I of 10 CFR Part 50.

Based on our evaluation, the proposed liquid and gaseous radwaste management systems for the Pebble Springs Nuclear Plant, Units 1 and 2, meet the criteria given in Appendix I and are, therefore, acceptable. We conclude that the liquid and gaseous radwaste treatment system will reduce radioactive materials in effluents to "as low as is reasonably achievable" levels in accordance with 10 CFR Part 30.34a.

15.0 ACCIDENT ANALYSIS

15.0 Fuel Handling Accident

This section constitutes a further analysis of a fuel handling accident as reported in the Safety Evaluation Report.

The applicant has submitted an evaluation of the consequences of a postulated fuel handling accident inside containment. The applicant has stated that the containment will be purged during fuel handling operations by the containment purge exhaust system which will exhaust 24,000 cubic feet per minute from the area of the refueling cavity. Although the exact exhaust duct arrangement is not available, the applicant plans to exhaust air from the refueling cavity area through three 8000 cubic feet per minute exhaust grills located about 18 feet above the pool surface. These join into a common duct which will be routed about 134 feet before joining other ducts exhausting 26,000 cubic feet per minute from the remainder of the containment just before passing through the containment purge exhaust system isolation valves.

The applicant plans to install a containment radiation monitor system and to arrange that a high radiation signal which will automatically cause the containment purge exhaust system isolation valves to isolate the containment. While the sample point for the radiation monitor has not yet been specified, it has tentatively been planned to locate the sample probe in the exhaust duct about 32 feet downstream of the isolation valve, with the detector itself to be located at the end of a sample line about 100 feet long.

The applicant has provided the sequence of containment isolation to be used following a postulated accident. In this sequence, the leading edge of any activity released passes the isolation valve about one second before reaching the sample probe. The transit time of activity along the sample line from the probe to the detector is approximately 9.0 seconds. The containment monitor response and actuation of an engineered safety features actuation system is assumed to occur in about 2.9 seconds while the isolation valves will be designed to close within 5 seconds upon receipt of an engineered safety features actuation system. Using the maximum values, the applicant calculates that for almost 18 seconds activity is released from the containment before isolation occurs.

We have determined that the consequences of a fuel handling accident should be well within the guideline values of 10 CFR Part 100, and should also allow adequate margin for uncertainties at the construction permit stage to assure that the doses will be well within the 10 CFR Part 100 guidelines at the operating license review stage. We interpret "well within" to mean that the calculated consequences should lie in the range of 10 to 25 percent of the guideline values of 10 CFR Part 100. Also, the consequences of a fuel handling accident inside containment can be effectively mitigated either by prompt detection and isolation of any activity release, or by providing an engineered safety features filtration system where this is not feasible.

We have performed an evaluation of the applicant's proposed design in light of these criteria. We find that we cannot concur with the applicant that the radiological consequences would be well within the guideline values of 10 CFR Part 100. We believe that the proposed exhaust grill arrangement does not provide assurance of substantial mixing and dilution of any activity released above the refueling cavity surface. We also find that the proposed location and response time of the containment radiation monitor system to be such that prompt detection is not assured.

The applicant has agreed to suitably modify his proposed design either to provide an arrangement of exhaust venting that provides assurance of significant mixing and dilution, or to revise the placement and/or response time of the containment radiation monitor system to provide assurance for prompt isolation and detection. We will review the proposed changes and report our evaluation in a future supplement.

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

On February 5-7, 1976, the Advisory Committee on Reactor Safeguards reviewed the application of the Portland General Electric Company to construct and operate the Pebble Springs Nuclear Plant, Units 1 and 2. This application was previously considered at a Subcommittee meeting on January 30, 1976 at Portland, Oregon; and the site of the proposed plant was visited by the Committee on January 29, 1976. A copy of the Committee's interim report, dated February 11, 1976, is attached as Appendix D. A discussion of the current status of each item on which the Committee commented or made recommendations is provided in the following paragraphs.

(1) The Committee noted that the seismic design basis and matters related to the disposition of volcanic ash arising from a postulated volcanic eruption at Mount Hood or Mount Saint Helens are still under review. With respect to the seismic design basis for the Pebble Springs site, it is still unresolved pending the generic resolution of the 1872 earthquake and its impact upon the site. Resolution of this matter will be reported in a future supplement. Matters related to the potential disposition of volcanic ashfall are reported in this supplement (Section 2.5). This item is resolved to the satisfaction of NRC staff and the applicant.

(2) The Committee noted that it reserves judgment concerning the performance of the Mark C fuel design since a complete analysis of its performance is not yet available. Also, the Committee recommends that the applicant continue studies directed at further improvement in the capability and reliability of the emergency core cooling system and to be kept informed on progress. Since there is no operating experience on the new fuel design, we require that a supplemental fuel surveillance program be conducted. The supplemental fuel surveillance program is directed at monitoring the behavior of the actual fuel systems as they perform inreactor, thus demonstrating the adequacy of the conclusions reached in the design evaluation. We are, therefore, requiring a supplemental fuel surveillance program for the first two plants with Mark C fuel. Based on existing construction schedules, we expect the first two plants to be Bellefonte Unit 1 and Washington Nuclear Project Unit 1. The details of the surveillance program requirements are provided in our letter, D. F. Ross (NRC) to K. E. Suhrke of Babcock & Wilcox, September 20, 1976. Briefly, the program will consist of a visual inspection of all the peripheral rods in the initial-core fuel assemblies as they are discharged into the spent fuel pool. The visual inspection will include observations for cladding defects, fretting, rod bowing,

corrosion and deposition and geometric distortion. If any anomalies are detected during the visual examination, further investigation will be performed, including, under unusual circumstances, destructive examination of a fuel assembly or individual fuel rods as required.

With respect to improvements in the emergency core cooling system, the latest Status Report on Generic Items, dated January 31, 1977 described a program to explore new emergency cooling approaches for application to future plants. Included are NRC plans to test the engineering principles of new vendor-proposed concepts. This will provide a basis for independent review of designs that may be proposed in the future by vendors for incorporation into their plants.

(3) The Committee noted that the utilization of a proposed reactor protection system, designated RPS-II, should be resolved in a manner satisfactory to the staff. In a letter to Babcock & Wilcox dated January 8, 1976 concerning the review of topical report BAW-10085, we stated that "the hybrid design of RPS-II represents an acceptable concept for application in a reactor protection system." Our review of the topical report is being conducted as a generic item and is incomplete at this time.

However, it is our intent to complete our review and evaluation of BAW-10085 prior to the receipt of the Final Safety Analysis Report for the Pebble Springs Nuclear Plant. The applicant has committed to conform to the generic resolution of the BAW-10085 review (Section 7.2 of the Safety Evaluation System). We consider this commitment to be acceptable for the issuance of a construction permit.

(4) The Committee noted that loads on the vessel support structure for certain postulated loss-of-coolant accidents should be resolved in a manner satisfactory to the staff. The applicant has committed to employ acceptable procedures and analytical techniques to assure the adequacy of the reactor vessel support (Section 3.9.14 of the Safety Evaluation Report, Supplement No. 1).

Two topical reports concerning this matter have been submitted by Babcock & Wilcox, as follows:

BAW 10131	"Reactor Coolant System Structural Loading Analysis," received January 5, 1977.
BAW 10132P	"Reactor Coolant System Hydrodynamic Loading During a Loss-of-Coolant Accident," received March 9, 1977.

The analytical methods to be utilized for the analyses of reactor internals will consider loss-of-coolant accident loads which include asymmetric cavity pressure, internal differential pressure, core bounce, and thrust due to pipe rupture defined by force as a function of time. The reactor vessel support loads will include the loss-of-coolant loads, seismic, thermal, deadweight and dynamic loads produced by equipment attached to the vessel.

We are continuing to review the methods in more detail in the context of the two topical reports noted above. There is at this time reasonable assurance that the methods proposed will be demonstrated to be capable of predicting the combined stresses and strains in the components of the reactor coolant system and reactor internals. The applicant will provide final design information at the operating license stage of review that demonstrate that:

(1) Combined stresses and strains in the reactor coolant system, including reactor internals, will not exceed the allowable design stress and strain limits for the materials of construction as specified in Appendix F to the Boiler and Pressure Vessel Code, Section III, and

(2) The resulting deflections or displacements of any structural elements of the reactor internals will not distort the reactor internals geometry to the extent that core cooling is significantly impaired.

We have concluded that there is reasonable assurance that the use of the proposed analytical techniques will confirm the adequacy of the reactor vessel supports and will result in an acceptable structural design for the facility reactor internals.

(5) The Committee recommended that the design of the Pebble Springs Nuclear Plant be such that potential design changes to minimize serious consequences from anticipated transients without scram can be readily incorporated should they be deemed necessary. The status of this matter is that on December 9, 1975, we issued our staff status report which identified guidelines for further analyses, and in a staff letter of April 7, 1976, we required Babcock & Wilcox to provide analyses by June 30, 1976 and also to identify design changes to meet anticipated transients without scram limits. Subsequently, Babcock & Wilcox requested a delay for submittal of these analyses. In December 1976, Babcock & Wilcox provided a partial analyses.

We are continuing a generic review of this area of concern and the staff evaluation of the Babcock & Wilcox analyses are expected to be published this year. We will require that any changes indicated to be needed in the facility design as the result of approved Babcock & Wilcox analyses shall be incorporated into the design in a timely manner. We will issue a construction permit for the Pebble Springs Nuclear Plant on this basis.

(6) The Committee noted that further evaluation concerning the assumptions made by the applicant is required on missile energy and penetration capability. This supplement to the Safety Evaluation Report provides further analysis on this matter (Section 3.5.1.2).

(7) The Committee recommended that the staff and applicant review its design features that are intended to prevent the occurrence of damaging fires and to minimize the consequences to safety-related equipment. This supplement to the Safety Evaluation Report provides further information on this matter (Section 9.5.1).

20.0 FINANCIAL QUALIFICATIONS

In the Safety Evaluation Report, we stated that we would report on this matter in a supplement to this report.

The Commission's regulations which relate to financial data and information required to establish financial qualifications for an applicant for a facility construction permit are Paragraph 50.33(f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50. To assure that we have the latest information to make a determination of the financial qualifications of an applicant, it is our current practice to review this information during the later stages of our review of an application. We are continuing our review of the financial qualifications of the applicant and will report the results of our evaluations in a supplement to this report.

21.0 CONCLUSIONS

In Section 21.0 of the Safety Evaluation Report, we stated we would be able to make certain conclusions upon favorable resolution of the outstanding matters set forth in Sections 1.8 and 1.9 of the Safety Evaluation Report. We have discussed these matters in this supplement as well as Supplements 1 and 2. Although we have reduced the Safety Evaluation Report issues to three items, two additional items have been generated. Furthermore, our rereview of the Pebble Springs design may produce additional issues.

Subject to a favorable resolution of existing items and any new issues resulting from the rereview, we will be able to reaffirm our conclusion as stated in the Safety Evaluation Report.

APPENDIX A

CHRONOLOGY OF RADIOLOGICAL SAFETY REVIEW

February 4, 1976	Letter to Portland General Electric Company on Decay Heat Removal System.
February 4, 1976	Letter from Portland General Electric Company forwarding Volcanic Hazard Study.
February 9, 1976	Letter from Portland General Electric responding to NRC financial questions.
February 11, 1976	Letter to Portland General Electric transmitting ACRS Interim Report on Pebble Springs Construction Permit Application.
February 18, 1976	Letter to Portland General Electric on USGS status for Pebble Springs site.
February 18, 1976	Letter to Portland General Electric requesting additional ECCS information.
February 26, 1976	Letter from Portland General Electric on schedule of information submittal concerning spray pond grouting.
March 9, 1976	Letter from Portland General Electric responding to NRC Geology/Seismology positions on the Pebble Springs Nuclear Plant.
March 10, 1976	Letter from Portland General Electric on additonal information that was left out of 3/9/76 letter concerning Geology/Seismology of the Pebble Springs site.
March 12, 1976	Letter from Portland General Electric on NRC ECCS conversations of February 18, 1976.
March 18, 1976	Letter from Portland General Electric on Single Failure Analysis for DHR System Performing its Emergency (Low Pressure Injection) function.

March 22, 1976	Letter from Portland General Electric on Transmittal of Test Fill for concrete backfill for Central Plant Facilities for Pebble Springs.
March 30, 1976	Letter from Portland General Electric on reservoir dam grouting.
March 31, 1976	Letter from Portland General Electric on PSAR updates.
April 8, 1976	Letter to Portland General Electric on inspection of fuel assemblies with new B&W fuel design.
April 9, 1976	Letter from Portland General Electric on comments concerning Pebble Springs Safety Evaluation Report.
April 12, 1976	Letter to Portland General Electric on comments and questions regarding volcanic ash fall at the Pebble Springs site.
April 15, 1976	Letter to Portland General Electric concerning information needed to evaluate containment refueling accident.
May 19, 1976	Letter from Portland General Electric on responses to questions on volcanic ashfall at Pebble Springs site.
May 19, 1976	Letter from Portland General Electric providing additional financial information.
May 20, 1976	Letter for Portland General Electric on monthly sales and statistical data for PGE, PSP&L, and PRL requested April 23, 1976 by NRC.
May 21, 1976	Letter from Portland General Electric on Revision 1 of Volcanic Hazard Study - Potential for Volcanic Ash Fall at the Pebble Springs site.
May 21, 1976	Letter from PGE on Amendment 10.
June 1, 1976	Letter from Portland General Electric on "Supplemental Stability Analysis of Pavement Excavation Slopes in the Emergency Spring Ponds."
July 7, 1976	Letter from Portland General Electric on turbine missile hazards.

July 19, 1976	Letter from Portland General Electric to advise NRC of receipt of letter on turbine missile hazards.
July 20, 1976	Letter to Portland General Electric concerning anticipated transients without scram (ATWS).
August 27, 1976	Letter to Portland General Electric concerning additional information on Amendments Part 2, 50, 51 10 CFR (See May 11, 1976 letter).
September 8, 1976	Letter to Portland General Electric regarding antitrust information.
September 17, 1976	Letter from Portland General Electric on analysis of turbine missile hazard.
September 23, 1976	Letter to Portland General Electric requesting additional financial information.
September 30, 1976	Letter to Portland General Electric on fire prevention plans for Pebble Springs.
February 2, 1977	Letter to Portland General Electric transmitting USGS status reports dated 11/5/76 and 1/11/77 of Volcanic Hazard Study.
February 25, 1977	Letter from Portland General Electric on Fire Protection Review (PGE 2013).
April 11, 1977	Letter from Portland General Electric on Annual Reports from utilities involved in the Pebble Springs site.
April 15, 1977	Letter to Portland General Electric on the fuel handling accident at Pebble Springs.
April 19, 1977	Letter to Portland General Electric on Correction to ECCS evaluation model.
April 22, 1977	Letter to Portland General Electric on Standard Format for meteorological data on magnetic tape.
May 26, 1977	Letter to Portland General Electric concerning unresolved Pebble Springs issues.

June 28, 1977	Letter to Portland General Electric on Radiological evaluation of fuel handling accident inside containment.
July 10, 1977	Letter to Portland General Electric on Analysis of Postulated Main Steam Line Break Accident.
July 27, 1977	Letter to Portland General Electric on fire protection.
August 30, 1977	Letter from Portland General Electric on Analysis Postulated Main Steam Line Break Accident.
September 7, 1977	Letter from Portland General Electric on update of significant engineering developments.

APPENDIX B

ERRATA TO THE SAFETY EVALUATION REPORT FOR THE PEBBLE SPRINGS NUCLEAR PLANT, UNITS 1 AND 2

SER Page	SER Section	COMMENT
5-9	5.5.7	The design basis for the sizing of the reactor coolant drain tank is incorrect as stated. The correct design basis for this tank is a complete loss of main feedwater. Refer to PSAR Sections 5.2.2, 5.5.11.1, and 11.2.2.1.1.
6-4	6.2.2	Automatic backup is provided for the manual switching of the containment spray pump suction to the recirculation sump on low level in the BWST as a consequence of the automatic switchover of the DHR (low pressure injection) system from injection to recirculation.
6-11	6.4	The second sentence should read "In the event of a high radiation signal at the outside air intake to the control room, loss of power, or an engineered safety features actuation system signal..."
7-2	7.3	The last sentence on this page should read "When either of the two digital subsystems receives two trip inputs, one set of the minimum required ESF devices is actuated."
7-4	7.3.5	PGE has provided the required preliminary information regarding the automatic switchover from injection to recirculation. Refer to letter dated January 20, 1976 and to PSAR Section 7.6.1.2.
7-7	7.4.2	The scope of ECI has not been recently expanded. The ECI has always included the post-accident monitoring recorders as well as Class IE safety-related display instrumentation. ECI control of the Auxiliary Feedwater System is not required to mitigate the consequences of any accident ESFAS actation is provided for this

		protection. PSAR Section 7.4 will be expanded to include a description of all functions of the ECI system. Table 7.5-1 provides a list of the post-accident monitoring instrumentation.
8-3	8.3.1	In Section 8.3.1.1.7.4 (Amendment 8) the required information was provided regarding the details of our design with respect to those cases where a third pump can be connected to either ESF bus. Note also that there is no longer a third ESF equipment room chiller (Refer to PSAR Page A5-95).
9-6	9.2.4	The component cooling water system supplies water to the containment spray pump and decay heat removal pump seal coolers rather than to the seals themselves.
10-2	10.3	There is only one air-operated relief (atmospheric dump) valve for each steam generator, rather than one for each main steam line. Refer to PSAR Table 10.4-2 and Figure 10.3-1.
11-4	11.3	The first four items in Table 11.2-1 are designed to Quality Group D (Augumented) requirements.
11-5	11.3	Excess gas accumulating in the GRS surge tank will be vented through the Auxiliary Building vent exhaust. The GRS is designed to Quality Group D (Augmented), not Quality Group C. The third paragraph on SER page 11-5 should be rewritten as follows:

> "The Containment Building atmosphere will be continuously exhausted through HEPA filters and charcoal adsorbers using a mini-purge system. The main purge exhaust system will be used during refueling. The radwaste and fuel handling areas in the Auxiliary Building, the Containment Building, and the Turbine Building ventilation exhausts will be released to the environment without treatment. The facility ventilation systems will be designed to induce air flows from potentially less radioactive contaminated areas to areas having a greater potential for radioactive contamination."

11-6	11.4	The condensate demineralizer backflush wastes will not necessarily be solidified. Low specific activity sodium sulfate crystal produced by centrifugation of condensate demineralizer backflush waste will be packaged without further solidifications.
11-7	11.5	Table 11.5-1 should be expanded to include monitoring of the RCS letdown line, the Auxiliary Building ventilation exhaust, and the central and the control room outside air supply duct.
13-3	13.4	The PSAR section dealing with review and audit is Section 13.4.
15-6	15.5	The radioactive waste gas decay tanks are not designed as seismic Category I.
15.8	15.6	In Table 15.6-2, under Spray effectiveness, the effective volume should be 2.08×10^6 cubic feet.
S1-8	7.3.7	In the second paragraph, the second sentence should read: "Class IE Power to the valve motor operator control circuits and position indicator circuits will be provided."
S1-16	15.7	The containment will not be purged through the spent fuel pool area exhaust system filters. As described in PSAR Section 6.2 and 9.4, a separate non-ESF zeolite filter is provided for the hydrogen vent system.

APPENDIX C

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - GENERIC MATTERS

The Advisory Committee on Reactor Safeguards (the Committee) periodically issues a report listing various generic matters applicable to all large light water reactors. We believe each of these matters should be carefully considered and, as conclusions are drawn, each facility or reference design application should be evaluated with respect to those issues appropriate to that application. We recognize that this could result in a necessity for modification of a facility even after the facility is completed. This is consistent with our continuing efforts toward reducing still further the already small risk to the public health and safety from nuclear power plants. The most recent such report concerning these generic items was issued in a letter dated February 24, 1977 to Commission Chairman Rowden from Committee Chairman M. Bender.

The status of staff efforts leading to resolution of all unresolved generic matters is contained in our Status Report on Generic Items periodically transmitted to the Committee. The latest such Status Report is contained in a letter from B. Rusche to M. Bender dated January 31, 1977.

For several of the items we have provided in this report specific discussion particularizing for the proposed facility the generic status in the Status Report. These items are listed below with the appropriate section numbers of this report where such discussions are included. The group numbering corresponds to that in the February 24, 1977 report of the Committee.

Group II

(1) Turbine Missiles - This item is discussed in this supplement to the Safety Evaluation Report (Section 3.5.1.2).

(2) Effective Operation of Containment Sprays in a Loss-of-Coolant Accident - Resolved for this facility (Section 6.2.3).

(3) Possible Failure of Pressure Vessel Post-Loss-of-Coolant Accident by Thermal Shock - This item is under generic review as indicated in our status report to ACRS dated January 31, 1977.

(4) Instruments to Detect (Severe) Fuel Failures - This item is partly resolved, as reported in the February 24, 1977 letter from the Committee to the Commission. Instrumentation to detect fuel failures associated with normal operation and transients (limited fuel failures) has been shown to be adequate. The adequacy of instrumentation to detect failures associated with more rapid events during which substantial fuel failure could occur has not been demonstrated and this concern is considered unresolved. Further work is necessary to determine (a) the adequacy of current instrumentation of these rapid events, and (b) the need for additional instrumentation. Research administred by the Division of Reactor Safety Research and studies conducted under contracts administered by the Office of Nuclear Reactor Regulation should provide the information required to evaluate instrumentation limitations and needs. In the interim, we have not identified any credible event (transient or accident sequence) for which a rapid fuel failure detection system would prevent "substantial" fuel failure (including fuel melt) and loss of coolable geometry. We conclude that this ongoing generic matter should not preclude issuance of a construction permit for the Pebble Springs Nuclear Plant.

(5) Monitoring for Excessive Vibration or Loose Parts Inside the Pressure Vessel - The applicant is committed to install a loose parts monitor.

(6) Non-Random Multiple Failures - This item is under generic review as indicated in our status report to ACRS dated January 31, 1977.

(7) Behavior of Reactor Fuel Under Abnormal Conditions - This item is under generic review as indicated in our status report to ACRS dated January 31, 1977.

(8) BWR Recirculation Pump Overspeed During Loss-of-Coolant Accident - This item is not applicable to the Pebble Springs Nuclear Plant which is a pressurized water reactor.

(9) The Advisability of Seismic Scram - A seismic scram is not proposed for Pebble Springs and we will not require such a scram - see letter dated May 19, 1977, from E. Case, Acting Director, Office of Nuclear Reactor Regulation, to Committee Chairman Bender; subject: "The Advisability of a Seismic Scram."

(10) Emergency Core Cooling System Capability for Future Plants - This item is under generic review as indicated in our status report to ACRS dated January 31, 1977.

Group IIA

(1) Control Rod Drop Accident (Boiling Water Reactors) - This item is not applicable to the Pebble Springs Nuclear Plant which is a pressurized water reactor.

(2) Ice Condenser Containments - The Pebble Springs Nuclear Plant does not utilize an ice condenser containment.

(3) Rupture of High-Pressure Lines Outside Containment - This item is resolved for Pebble Springs Nuclear Plant by compliance with criteria specified in the Standard Review Plan (Section 3.4.2).

(4) Pressurized Water Reactor Pump Overspeed During a Loss-of-Coolant Accident - This item is resolved for Pebble Springs Nuclear Plant by the applicant's commitment to submit the results of the overspeed evaluation as part of the Final Safety Analysis Report. This type of information is not required prior to the issuance of a construction permit.

(5) Isolation of Low Pressure from High Pressure Systems - This issue is not resolved for Pebble Springs Nuclear Plant. We will require additional information from Babcock & Wilcox to demonstrate compliance with current staff requirements (Section 7.4.1 of this supplement).

(6) Steam Generator Tube Leakage - This item is resolved by controls on secondary system chemistry and the design provisions for inservice inspection.

(7) ACRS/NRC Periodic 10-Year Review of all Power Reactors - This item is under generic review as indicated in our status report to ACRS dated January 31, 1977.

Group IIB

(1) Computer Reactor Protection System - This item is resolved for Pebble Springs Nuclear Plant by a commitment to the outcome of topical report review (Section 7.2).

(2) Qualification of New Fuel Geometries - This item is partially resolved by similarity to existing fuel geometries of proven performance and the Babcock & Wilcox ongoing test program for the Mark C fuel assembly. The continuation of these test programs and an industry-wide surveillance program provides an ongoing generic review at this item. We conclude this generic qualification program is directed at design confirmation and should not preclude the issuance of a construction permit for the Pebble Springs Nuclear Plant (Section 4.2).

(3) Behavior of Boiling Water Reactor Mark III Containments - This item is not applicable to the Pebble Springs Nuclear Plant which is a pressurized water reactor facility.

(4) Stess Corrosion Cracking in Boiling Water Reactor Piping - This item is not applicable to Pebble Springs Nuclear Plant which is a pressurized water reactor.

Group IIC

(1) Locking Out of Emergency Core Cooling System Power-Operated Valves - This item was covered in Supplement No. 2 of the Safety Evaluation Report.

(2) Design Features to Control Sabotage - This item is resolved for Pebble Springs Nuclear Plant (Section 13.7).

(3) Decontamination and Decommissioning of Reactors - This item is under generic review as indicated in our status report to ACRS dated January 31, 1977.

(4) Vessel Support Structures - This item is under generic review by the NRC staff. The applicant has made a commitment that their final design procedures will account for asymmetric loadings in the reactor vessel supports.

(5) Water Hammer - Section 10.2. This item is under generic review as indicated in our status report to ACRS dated January 31, 1977.

(6) Maintenance and Inspection of Plants - This item is resolved for the Pebble Springs Nuclear Plant.

(7) Behavior of Boiling Water Reactor Mark I Containments - This item is not applicable to Pebble Springs Nuclear Plant which is a pressurized water reactor facility.

Group IID Resolution Pending - Items Added Since April 16, 1976

(1) Safety-Related Interfaces Between Reactor Island and Balance-of-Plant - This item is not applicable to Pebble Springs Nuclear Plant which is a custom design.

(2) Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment - This item is addressed in Pebble Springs Nuclear Plant Preliminary Safety Analysis Report as a general requirement for environmental qualification of equipment (Section 3.11).

APPENDIX D

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NUCLEAR REGULATORY COMMISSION

WASHINGTON, D. 20555

February 11, 1976

Honorable William A. Anders
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: INTERIM REPORT ON PEBBLE SPRINGS NUCLEAR PLANT, UNITS 1 & 2

Dear Mr. Anders:

At its 190th Meeting, February 5-7, 1976, the Advisory Committee on Reactor Safeguards completed an interim review of the application of the Portland General Electric Company for permission to construct the Pebble Springs Nuclear Plant, Units 1 and 2. This Plant was previously considered at a Subcommittee meeting on January 30, 1976, at Portland, Oregon, and the site for the proposed plant was visited on January 29, 1976. During its review the Committee had the benefit of discussions with representatives of the Portland General Electric Company (PGE) and consultants, the Babcock and Wilcox Company (B&W), the Bechtel Power Corporation, and the NRC Staff. The Committee also had the benefit of the documents listed.

The proposed Pebble Springs Plant will be located on an 8650-acre tract of land near Arlington, Oregon, approximately 55 miles west-southwest of the Tri-Cities (Kennewick, Pasco, and Richland, Washington) area, the nearest population center (1970 population - 55,000). The exclusion radius is 800 meters; the low population zone radius is 2 miles. In 1970 there were 6 residents within the low population zone.

The seismic design basis and matters related to the deposition of volcanic ash arising from major volcanic eruptions at Mount Hood and Mount Saint Helens are still under review by the NRC Staff, the United States Geological Survey and the Applicant.

The ultimate heat sink will include a seismic Category 1 spray pond with a seismic Category 1 intake structure housing the two backup service water pumps. The system also includes the Pebble Springs reservoir, which is nonseismic Category 1 but is protected against tornado damage. Makeup to the reservoir will be from the Columbia River and makeup to the spray pond will be from the reservoir.

The Pebble Springs Nuclear Plant is currently planned to consist of two identical nuclear generating units. The nuclear steam supply systems (NSSS's) will be supplied by B&W and will be identical to other B&W 205 Mark C fuel assembly NSSS's, including Bellefonte Nuclear Plant, Units 1 and 2, on which the ACRS reported on July 16, 1974.

The NRC Staff and the Applicant report that, employing the currently accepted LOCA-ECCS B&W evaluation model, peak clad temperatures have a margin to the limiting condition of 2200° F.

The Committee recommended in its report of January 7, 1972, on Interim Acceptance Criteria for ECCS, that significantly improved ECCS capability should be provided for reactors for which construction permit applications were filed after January 7, 1972. This position was repeated in its report of September 10, 1973, on Acceptance Criteria for ECCS. The Mark C fuel assemblies are responsive to this recommendation inasmuch as they will be operated at lower linear heat generation rates and are expected to yield greater thermal margins to fuel design limits. An extensive program has been initiated for determining the mechanical and thermal/hydraulic characteristics of the new fuel assemblies. A program of control rod tests also is proposed, including testing of trip times and control rod wear. Should modifications become necessary as a result of the control rod tests, retesting of the entire control rod drive would be undertaken. While many of the details of the proposed design are available, complete analyses of the performance of the Mark C fuel are not yet available, and the NRC Staff has not completed its review. The Committee reserves judgment concerning the final design until the required performance information is presented and has been reviewed. The Committee recommends that the Applicant continue studies directed at further improvement in the capability and reliability of the ECCS. The Committee wishes to be kept informed.

The Applicant proposes to utilize a new reactor protection system designated as RPS-II. The system, a hybrid using both analog and digital techniques, represents an evolution from the analog system, RPS-I, currently in use in the Oconee reactors. The Applicant has proposed a series of environmental, reliability, and in situ tests for qualification of this system prior to its use in Bellefonte, Units 1 and 2, the lead plant. This matter should be resolved in a manner satisfactory to the NRC Staff.

A question has arisen concerning loads on the vessel support structure for certain postulated loss-of-coolant accidents in pressurized water reactors. This matter should be resolved for the Pebble Springs Nuclear Plant in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

Specific consideration of the question of anticipated transients without scram is now under way by the NRC Staff. The Committee recommends that the design of Pebble Springs Nuclear Plant, Units 1 and 2, be such that potential design changes to minimize serious consequences from ATWS can be readily incorporated, should they be deemed necessary. This matter should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

The Committee believes that the Applicant and the NRC Staff should continue to review the Pebble Springs Plant design for features that could reduce the possibility and consequences of sabotage.

The Committee recommends that the NRC Staff and the Applicant review the design features that are intended to prevent the occurrence of damaging fires and to minimize the consequences to safety-related equipment should a fire occur. This matter should be resolved to the satisfaction of the NRC Staff. The Committee wishes to be kept informed.

The Applicant has calculated that the probability of adverse effects on the ability to shut the plant down safely due to turbine-generated missiles is acceptably low. The ACRS believes this analysis requires further evaluation, particularly with regard to the assumptions concerning missile energy and penetration capability. The Applicant has stated that he has backup positions including a steel turbine-missile shield which can be implemented late in the construction phase. The Committee recommends that this matter be resolved in a timely fashion, during construction, in a manner satisfactory to the NRC Staff and the ACRS.

The exact schedule for construction of Pebble Springs Nuclear Plant, Units 1 and 2 remains to be determined. The Committee recommends that if appreciable delay arises in the initiation of construction of Unit 1 from the originally planned schedule, or if delays lead to a completion date for Unit 2 significantly more than 10 years from now, the Plant should be reevaluated in terms of new regulatory requirements which may have significant effects in further protecting the health and safety of the public.

Generic problems relating to large water reactors are discussed in the Committee's report dated March 12, 1975. These problems should be dealt with appropriately by the NRC Staff and the Applicant.

The ACRS will review the site-related aspects of the application for a construction permit when the appropriate information has been developed, and evaluation has been completed by the NRC Staff.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

REFERENCES

1. Pebble Springs Nuclear Plant, Units 1 and 2, Preliminary Safety Analysis Report (October 29, 1974) with Amendments 1 through 8.

2. Safety Evaluation Report NUREG-0013 related to construction of Pebble Springs Nuclear Plant, Units 1 and 2, January 1976, with Supplements Nos. 1 and 2.

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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