Safety Evaluation Report

NUREG-0013

U. S. Nuclear Regulatory Commission

related to construction of

Pebble Springs Nuclear Plant Units 1 and 2

Office of Nuclear Reactor Regulation

Docket Nos. 50-514
50-515

Portland General Electric Company

January 1976

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SUPPLEMENT NO. 1

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

The Nuclear Regulatory Commission's (NRC or Commission) Safety Evaluation Report (SER) in the matter of the Portland General Electric Company Pebble Springs Nuclear Plant, Units 1 and 2 dated January 1976, stated that a supplemental report would be issued to update the SER in those areas where our evaluation had not been completed. The purpose of this supplement is to update the SER by providing our evaluation of additional information submitted by the applicant since issuance of the SER. As stated in the SER, we intend to issue another supplement to the SER after we receive the report from the Advisory Committee on Reactor Safeguards.

Each of the following sections in this report is numbered the same as the section of the Safety Evaluation Report that is being updated. Appendix A is a continuation of the chronology given in Appendix A of the SER.

2.0 SITE CHARACTERISTICS

2.1.2 Exclusion Area

In this section of the SER we stated that the applicant is to acquire the mineral rights in the exclusion area prior to the issuance of a CP. The applicant has obtained these rights and has the authority to determine all activities within this area as required by 10 CFR Part 100.3(e). This item is resolved.

2.2 Nearby Industrial, Transportation and Military Facilities

We identified in the SER a potential hazard due to the shipment to other industrial customers, significant quantities of chlorine and anhydrous ammonia on the Union Pacific railroad about 3 1/2 miles away from the site, which might affect personnel at the plant. The applicant has agreed to install non-seismic Category I ammonia and chlorine detectors to monitor the air supply to the control room. These detectors are designed to alarm and automatically trip and isolate the normal control room air supply. The detectors will be located so that the outside atmosphere can be monitored following isolation of the control room. Seismic Category I detectors would be required where significant quantities of chlorine were stored in the vicinity of the site. However, since these chemicals will not be stored at the site and since they will be at the railroad station for a short time, we conclude that non-seismic Category I detectors are acceptable. The probability of a seismic event causing a failure of the toxic gas detectors occurring simultaneously with a toxic gas release from a railcar is estimated to be of the order of 10^{-7} per year. Therefore, non-seismic detectors are adequate. We also believe that the distance between the site and the railroad (3.3 miles minimum), as well as the fact that the site is over 350 feet higher in elevation than the railroad, are factors that would significantly mitigate any consequences of a toxic gas release.

We conclude that the installation of non-seismic Category I ammonia and chlorine detectors is acceptable.

3.9 Mechanical Systems and Components

3.9.1.4 Analysis Methods Under LOCA Loadings

In this section of the SER we stated that the applicant must provide a commitment that the final design procedures will account for asymmetric loadings on the reactor vessel supports. The applicant has now made a commitment that the final design procedures will account for the resultant loadings on the reactor vessel internals and the exterior reactor pressure vessel supports, and we conclude that this commitment is acceptable for a CP review.

3.11 Environmental Design of Mechanical and Electrical Equipment

The applicant has revised its position on qualification. PGE now proposes a program to meet the requirements of IEEE Std. 323-1974 for all Class 1E electrical, instrumentation and control equipment by implementing a program identical to that submitted on Gulf States' River Bend Nuclear Power Plant application. We have reviewed the PGE program and conclude that it represents an acceptable method for meeting the qualification requirements of IEEE Std. 323-1974 as augmented by Regulatory Guide 1.89. The program consists of two parts. Part A discusses implementation and scheduling and commits the applicant to purchase aged and type-tested equipment wherever possible. However, wherever state of the art limitations on qualification technology exist, the applicant has agreed to develop alternate qualification techniques, which are discussed in Part B of the program. These techniques consist of:

(1) Justifying equipment qualification through previous operating experience and

(2) Qualifying equipment through some form of on-going qualification program.

Details of this program are included in a letter to NRC from PGE, dated January 20, 1976. We find this commitment to IEEE Std. 323-1974, as described in the above letter, an acceptable resolution for this CP Review.

6.2 Containment Systems

6.2.1 Containment Functional Design

In this section of the SER we stated that we had not completed our review of the applicant's minimum containment pressure analysis. The results of our findings are reported below.

Appendix K to 10 CFR Part 50 of the Commission's regulations requires that the effect of operation of all the installed pressure reducing systems and processes be included in the ECCS evaluation. For the evaluation, it is conservative to minimize the containment pressure since this will increase the resistance to steam flow in the reactor coolant loops and reduce the reflood rate in the core. Following a loss-of-coolant accident, the pressure in the containment building will be increased by the addition of steam and water from the reactor coolant system into the containment atmosphere. After initial blowdown, heat transfer from the core, reactor vessel, piping, and steam generators to the ECCS water will produce additional steam. This steam together with any ECCS water becomes energy that is released from the postulated LOCA break to the containment during both the blowdown and later ECCS operational phases, i.e., reflood and post-reflood phases.

Energy removal within the containment occurs by several means. Steam condenses on the containment walls and internal structures which serve as passive heat sinks that become effective early in the blowdown transient. Subsequently, the operation of the containment heat removal systems such as containment sprays and fan coolers will remove energy from the containment atmosphere. When the energy removal rate exceeds the rate of energy addition from the LOCA, the containment pressure

will decrease from its maximum value. The ECCS containment pressure calculations for Pebble Springs Units 1 and 2, were made with the Babcock & Wilcox evaluation model. We reviewed the Babcock & Wilcox model and published a status report on October 15, 1974, which was amended November 13, 1974. We concluded that Babcock & Wilcox's containment pressure model was acceptable for ECCS evaluation. We required, however, that justification of the containment plant-dependent input parameters used in the analysis be submitted for our review of each plant.

The containment input data relating to containment net-free volume, passive heat sinks and containment heat removal systems were submitted for Pebble Springs Units 1 and 2, in Amendment 7 to the PSAR. The staff has evaluated the containment input data and found that the data used for Pebble Springs are conservative for the ECCS containment pressure analysis. Data for the passive heat sinks are conservative in comparison with our recommendations contained in the Branch Technical Position (BTP) CSB 6-1; the passive heat sink data are based on measurements within the containment of similar nuclear plants. The containment heat removal systems were assumed to operate at their maximum capacities and minimum operational values for the spray water and service water temperature were assumed.

We have concluded that the plant-dependent information used for the ECCS containment pressure analysis for Pebble Springs. Units 1 and 2, is reasonably conservative, and therefore, the calculated containment pressures are in accordance with Appendix K to 10 CFR Part 50 of the Commission's regulations.

7.0 INSTRUMENTATION AND CONTROLS

7.2 Reactor Protection System (RPS)

The Reactor Protection System is currently being reviewed generically as RPS-II under Topical Report BAW-10085. As indicated in the SER issued January 12, 1976, the applicant has satisfied the staff's requirements by agreeing to comply with the generic resolution of this topical report. In addition, in the event that NRC has not approved the RPS-II design by the time of the FSAR submittal, the applicant will provide an RPS design that achieves at the least the same degree of safety that is provided by the previously reviewed and accepted RPS-I design. We find this additional commitment acceptable.

7.3.7 Core Flood Isolation Valve Interlocks

In this section of the SER we stated our concern that valve position indication and motive power to the valve operators for the Core Flood Isolation Valves would not be Class IE. In addition, no ESFAS signal was to be provided to open the isolation valves.

The applicant has proposed the following modifications. Class IE power to the valve motor operator control circuits and alarm indicator circuits will be provided. Specifically, each 480 V motor operated valve will be supplied motive power from separate Class IE emergency buses. In addition, redundant Class IE valve position indication will have separate and redundant position switches on each valve.

With respect to our position that an ESFAS signal should be provided to open the isolation valves PGE has presented acceptable proof

that such actuation is not necessary. Namely, that the core flood system is not required to mitigate the consequences of even the worst case LOCA below 750 psig and that the Class IE valve control circuitry has been designed so that the isolation valves open automatically at 750 psig during normal plant startup. Reactor coolant pressure interlocks are provided to prevent the valves from being inadvertently closed. The operator can then initiate the necessary action to disconnect the power supplies to the valve by locking out the relevant breaker. This step will electrically disable the isolation valve in the open position during normal operation. No credible single electrical failure can then close either core flood isolation valve.

In summary, PGE does not take credit for the core flood system to mitigate LOCA consequences at pressures below 750 psig. At higher pressures, the core flood isolation valves will automatically open and the operator will, by locking out breakers, electrically disable the valves in the open position. An ESFAS signal is therefore not needed to open the valves because the valves will already be open when needed. We therefore find this design acceptable.

7.4.3 Auxiliary Feedwater System (AFS)

The applicant has modified the design of the AFS to meet the applicable criteria of diverse power sources. The design includes two direct-drive diesel engine pumps and two electric motor driven pumps powered from separate, 480V emergency buses.

The diesels are water cooled with self-contained starting systems.

Both diesels and all auxiliaries are independent of all a-c power. Fuel oil is supplied directly to the fuel injection system of the diesel from a separate day tank.

Normal flow to the auxiliary feedwater pumps is from the non-seismic Category I condensate storage tank through a single seismic Category I line. The pumps can also be supplied from the Category I backup service water system (BSWS) through two lines. Each train of the BSWS will supply water to one motor-driven and one engine-driven auxiliary feedwater pump. One auxiliary feedwater pump associated with each BSWS train will supply the A steam generator. The other pump associated with each BSWS train will supply the B steam generator. We find this design acceptable

The changeover of water supply from the condensate storage tank to the BSWS will be done automatically by opening one 480V Class IE motor-operated valve in each service water line. The automatic switchover will occur only on low level in the condensate storage tank concurrent with automatic initiation of the AFS. A two-out-of-three low level logic signal will be used for each channel. In addition PGE has agreed to design the switchover circuitry to meet the requirements of an engineered safety feature. We find this commitment and the conceptual design acceptable.

Each of four auxiliary feedwater pump discharge lines will have electro-hydraulic valves for flow control and isolation. These valves will be powered from separate uninterruptable vital 120V instrument a-c buses. The total power requirement for each valve is less than

1.6 HP

We sought assurances that the motor loads would not be allowed to adversely effect battery capability. PGE has provided this assurance by committing itself to size the d-c batteries so that they will be able to handle all required loads. We find this commitment acceptable.

PGE has addressed the problem of a main steam line break accident inside containment as follows. If one of the steam generators fails to repressurize after the initiation of auxiliary feedwater flow by the secondary system protection instrumentation (indicating a failure in the secondary pressure boundary of that steam generator) auxiliary feedwater flow to that steam generator will automatically be terminated by the closure of isolation valves in the lines to that steam generator.

We have sought assurance from PGE that the proposed design will meet the single failure criterion. PGE has provided this assurance by stating that the design will not only be modified where necessary to meet the single failure criterion, but will also meet all applicable criteria necessary to mitigate the worst case consequences of the postulated accident. Final design information will be reviewed during the FSAR review. We find this commitment acceptable.

8.0 ELECTRIC POWER

8.2 Offsite Power System

We stated in this section of the SER that the offsite power system as originally proposed did not comply with the intent of GDC 17 because one design basis event on the transmission system coupled with a single switchyard battery failure could have eliminated total offsite power to both units.

The applicant reviewed the design and has agreed to meet the intent of GDC 17 by modifying the offsite power system so that a design basis event coupled with any single failure would not eliminate offsite power to the plant. We find this commitment acceptable.

9.1 Fuel Storage and Handling

9.1.3 Spent Fuel Cooling and Cleanup Systems

In this section of the SER we stated that the spent fuel pool cooling system must be designed to seismic Category I requirements and Quality Group C to meet the intent of Regulatory Guides 1.26 and 1.29. The applicant has made a commitment to do so . This commitment constitutes an acceptable resolution of this item for the CP review.

9.2.2 Component Cooling Water Systems

In this section of the SER we stated that the proposed non-seismic Category I portion of the system was unacceptable. The cooling water system to the spent fuel pool heat exchangers must be seismic Category I to conform with Regulatory Guide 1.29. The applicant has agreed to make this change. We find this commitment acceptable.

15.0 ACCIDENT ANALYSIS

15.5 Radiological Consequences of Accidents

We stated in the SER that the radioactive waste gas decay tanks will be designed to seismic Category I requirements. The applicant has redesignated these tanks to be non-seismic Category I in conformance with our technical position on this matter.

We conclude that the total failure of these tanks is sufficiently improbable that 10 CFR Part 100 doses are applicable, and that, based upon our calculations for other recent facilities, the doses for failure of a single tank would be well within Part 100 guidelines.

We required that the air filtration system installed in the spent fuel pool exhaust area and intended to mitigate the radiological consequences of a postulated fuel handling accident, be designed to meet the requirements of an engineered safety features (ESF) system and to fully comply with Regulatory Guide 1.52, including design of the system to seismic Category I standards.

We also required that the area in the auxiliary building where the ECCS equipment is to be located, be served by an air filtration system which also fully meets the requirements of an ESF system.

The applicant, has redesigned system AB-4, which will serve as the spent fuel pool exhaust ventilation system, so that it fully complies with these requirements and is designated as seismic Category I. System AB-4 will also be suitably redesigned so that the same bank of filters used to exhaust the spent fuel pool area, with appropriate additional dampers and ductwork, will serve to filter any leakage from ECCS components in the auxiliary building. We find this acceptable. We will require that the

dampers and controls of system AB-4 be normally aligned so as to serve the spent fuel pool area in the event of a fuel handling accident. In the event of a LOCA, we will require that system AB-4 be realigned automatically to serve the ECCS equipment area prior to or at the onset of the switchover from safety injection to recirculation. The applicant has indicated the intent to redesign system AB-4 to satisfy our concerns for ESF filters in the auxiliary building and spent fuel pool exhaust system. We find this acceptable.

15.7 Hydrogen Purge Dose Analysis

We have revised our analysis of the radiological consequences of a hydrogen purge of the containment, post-LOCA. Since the applicant will purge the containment through the filters of the spent fuel pool exhaust system, we have given credit for an iodine removal efficiency of 95% for all forms of iodine. The assumptions and input parameters we used in evaluating the consequences of this mode of operation are listed in Table 15.7-1 and the calculated doses are listed in Table 15.6-1. The calculated long-term LOCA dose plus the hydrogen purge dose are within the guideline values of 10 CFR Part 100, and are acceptable.

TABLE 15.7-1

HYDROGEN PURGE DOSE INPUT PARAMETERS

Power Level (MWt)	3760
Volume of Containment (ft^3)	2.45×10^6
Purge Duration (days)	30
Holdup Time in Containment (days) Prior to Purge Initiation	10
Purge Rate (SCFM)	43
4-30-day X/Q (sec/m^3)	9.3×10^{-6}
Iodine Filtration Efficiency (percent)	95

TABLE 15.6-1

POTENTIAL OFFSITE DOSES DUE TO DESIGN BASIS ACCIDENTS

	Two-Hour Exclusion Boundary (788 Meters)		Course of Accident Low Population Zone (3200 meters)	
Accident	Thyroid (rem)	Whole Body (rem)	Thyroid (rem)	Whole Body (rem)
Loss of Coolant	88	4	68	3
Post-LOCA Hydrogen Purge Dose	--	-	17	2
Fuel Handling	5	2	< 1	< 1
Rod Ejection*				
Case I	48	< 1	53	< 1
Case II	75	2	--	--

* See Section 15.9 of the SER for the different assumptions used for Cases I and II.

15.10 Postulated Radioactive Releases Due to Liquid Tank Failures

In order to mitigate the consequences of postulated liquid tank failures, the applicant has committed to provide special design features in the auxiliary building drain system (ABDS). The ABDS is described in Subsection 3.6.6.4.3 and shown in Figure 3.6-1 of the applicant's Preliminary Safety Analysis Report. The ABDS will be designed to control, collect, and confine contaminated liquid which may result from postulated failure of a single tank. Our evaluation model assumes that a tank will be filled to 80% of its capacity.

Except for the equipment drain tank compartment, all compartments housing tanks having a combined capacity of less than 6,000 gal. will be drained directly to either of two radwaste receiver tanks (39,000 gal. each).

The equipment drain tank compartment will house a tank having a capacity of approximately 4,500 gal. Waste from this compartment will be routed directly to the auxiliary building passageway sump which will have a capacity of 3,700 gal. From the sump, the waste will be transferred via the sump pump to the radwaste receiver tanks for subsequent processing in the dirty radwaste system.

In the case of compartments containing tanks having a combined capacity of more than 6,000 gal., the ABDS drain line isolation valves (manual or air-operated) will be opened to route the waste to the radwaste receiver tanks. In the event of a radwaste

receiver tank spillage or failure, the radwaste receiver tank compartment will be provided with a steel liner to confine the waste in a water-tight enclosure should this tank fail. The compartment isolation valve will be kept closed until the waste can be routed to the dirty radwaste system via the auxiliary building passageway sump.

The ABDS special design features and modifications include the following: (1) level instrumentation in all tank compartments housing tanks over 6,000 gal. with annunciation in the control room; (2) air actuated isolation valves, operated from the control room, in the drain lines of those tank compartments which could result in radioactive concentrations in unrestricted bodies of water above the limits specified in 10 CFR Part 20, Appendix B, Table II, Column 2; (3) removal of the drain line isolation valve from the equipment drain tank compartment; (4) reduction in size of the equipment drain tank so that 80% of the tank capacity can be accommodated in the auxiliary building passageway sump following postulated failure; (5) a water-tight steel liner in the radwaste receiver tank compartment, sufficient to accommodate 80% of the capacity of the largest tank (reactor coolant bleed holdup tank) following postulated failure; and (6) drain lines (4" diameter) which are sized to ensure rapid drainage of waste liquid.

Considering the above special design features, we conclude that the provisions incorporated in the applicant's design to mitigate the consequences of postulated component failures involving contaminated liquids are acceptable.

APPENDIX A

Continuation of the Chronology of Radiological Safety Review

January 13, 1976	Applicant letter concerning reactor vessel supports
January 20, 1976	Applicant letter concerning items identified in section 1.8 and 1.9