PGE Portland General Electric Company

Bart D. Withers Vice President



July 17, 1981

Trojan Nuclear Plant
Docket 50-344
License NPF-1

Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Eisenhut:

On December 23, 1980, PGE's response to NUREG-0737 regarding implementation of the TMI-2 Action Plans was submitted to the NRC. This letter provides supplemental information to the following nine items as requested by NUREG-0737:

Action II.B.1 - Reactor Coolant System Vent

A general description of the Reactor Vessel Head Vent System (RVHVS) was submitted to the NRC in PGE letter dated January 2, 1980. PGE's item by item responses to the NRC Clarifications in NUREG-0737 were originally provided in PGE letter dated April 15, 1980 (Pages 20 through 23).

For your information and convenience, Attachment 1 is included for a summary of design descriptions and analytical considerations. Operating instructions for the RVHVS, including testing, are currently in preparation and will be completed prior to the design implementation date of July 1, 1982.

Action II.D.1 - Performance Testing of PORVs and Safety Valves

By letter dated July 1, 1981, R. C. Youngdahl (Consumers Power) transmitted the Interim Data Report for the EPRI PWR Safety and Relief Valve Test Program. This report summarizes the test data collected to date on relief valves; safety valve data are still not available. Trojan Nuclear Plant has 3-inch Copes-Vulcan relief valves, Model No. D-100-160, using 316 stainless steel valve body with stellite plug and 17-4PH cage. Trojan Nuclear Plant also uses Crosby safety valves, Model No. HB-BP-86, size 6M6. Results provided in Section 4.6 of the EPRI Interim Data Report are applicable to the Trojan Nuclear Plant. Our preliminary evaluation of the data provided in the EPRI report indicates that the Trojan valves will perform their intended function. PGE will submit a

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final evaluation and other Plant-specific data on the schedule consistent with R. C. Youngdahl's letters of December 15, 1980 and July 1, 1981.

A testing program for the block valves is addressed in the abovementioned R. C. Youngdahl letter to the NRC.

Action II.E.1.2 - AFW Automatic Initiation and Flow Indication

Modifications required to upgrade the AFW flow indication have been completed except for environmental qualification of equipment. Environmental qualification of the associated components is included in the ongoing IE Bulletin 79-01B electrical equipment qualification program and will be implemented as appropriate when that effort is complete.

Additional information regarding the AFW automatic initiation and flow indication was provided to the NRC in PGE letters of May 1 and June 30, 1981.

Action II.E.4.2 - Containment Isolation Dependability

NRC Position 5: Containment pressure setpoint.

During a telephone call on April 14, 1981, the NRC Staff requested that the Containment pressure setpoint, which initiates Safety Injection and Containment Isolation, be reduced to the minimum value compatible with normal operating conditions. As discussed in PGE letter of December 23, 1980 (Page 7) and the April 14 telephone call, we do not believe that changing the Containment pressure setpoint from 5.0 psig to 3.6 psig is warranted on a technical basis. However, as requested by the NRC Staff, the Containment pressure setpoint will be reduced to 3.6 psig with an allowable value of $\leq$4.1 psig to accommodate instrument fluctuations. Trojan Technical Specifications will be revised at a later date to reflect this change.

Action II.K.3.5 - Automatic Trip of Reactor Coolant Pump During Loss-of-Coolant Accident

As stated in Westinghouse Owners' Group letter (WOG-60) dated June 15, 1981 from R. W. Jurgensen to P. S. Check (NRC), utility action on this task is pending on NRC acceptance of the Westinghouse small-break model. The outstanding additional information relative to that model should be submitted by the Owners' Group during July 1981. In addition, the Owners' Group plans to provide, with that submittal, information to resolve the automatic RCP trip issue. Therefore, a proposed automatic RCP trip design will not be submitted until after NRC acceptance of the Westinghouse small-break model and notification by the NRC that such a design is needed.

Action II.K.3.25 - Effect of Loss of Alternating-Current Power on Pump Seals

The schedule for completion of this evaluation was inadvertently listed as July 1, 1981. The correct date should be January 1, 1982, which is in accordance with the NUREG-0737 schedule for PWR plants.

Action II.F.1.5 - Containment Water Level Monitor

The Containment Water Level Monitoring System utilizes two indicating ranges: narrow range (0-50 in.) and wide range (0-200 in.). Due to a lack of available pressure transmitter locations for the wide range indicators, the transmitters were located at 188 in. above the bottom of the Containment sump (Elevation 42 ft) and were calibrated to measure 188 in. of water depth. This location of the transmitters is above the expected height of the maximum water level for LOCA conditions which is approximately Elevation 53 ft (132 in.). The transmitters will be relocated and calibrated for 200-in. level during the next scheduled refueling outage in 1982.

Action II.F.2 - Instrumentation for Detection of Inadequate Core Cooling

Reactor Vessel Water Level Indication System:

The necessary hardware installation for the Trojan Reactor Vessel Water Level Indication System will be completed by January 1, 1982 in accordance with NUREG-0737. However, the system cannot be made operational until the next scheduled refueling outage in 1982, at which time necessary preoperational tests, functional tests, and system calibration will be conducted. This is in accordance with the NRC requirement of preimplementation review prior to system operation.

Action III.A.2 - Improving Licensee Emergency Preparedness

On April 15, 1981, PGE responded to NRC Generic Letters 81-10 (February 18, 1981) and 81-17 (March 15, 1981) by committing to have two Chemistry and Radiation Protection (C&RP) Technicians on shift beginning September 1, 1981. The manning level of the C&RP Technicians has dropped significantly since this commitment was made, which prevents us from meeting the September 1 date. Therefore, we propose to meet the NRC requirement in NUREG-0654 alternatively by having one C&RP Technician on shift and one

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C&RP Technician on call within 60 minutes for the period of September 1, 1981 to July 1, 1982. From July 1, 1982, the on-shift requirement of having two C&RP Technicians will be satisfied.

Sincerely,

Bart D. Withers
Vice President
Nuclear

Attachment

c: Director of Nuclear Reactor Regulation
Mr. Robert A. Clark, Chief
Operating Reactors Branch No. 3
Division of Licensing
U. S. Nuclear Regulatory Commission

Mr. Lynn Frank
State of Oregon
Department of Energy

ATTACHMENT 1

REACTOR COOLANT VENT SYSTEM - TROJAN NUCLEAR PLANT

## Design Basis

The function of the Reactor Vessel Head Vent System (RVHVS) is to remove noncondensable gases or steam from the reactor vessel head.

This system is designed to mitigate a possible condition of inadequate core cooling or impaired natural circulation.

Venting capability of RCS hot legs is not required since the hot legs are not a high point in the system.

## Design Description and Evaluation

### General Description

The RVHVS is designed to remove noncondensable gases or steam from the reactor vessel via remote manual operations from the control room. This system provides additional flexibility to the plant operator during both normal and accident mitigation operations. Also, venting during plant startup, system venting during shutdown on post-accident mitigation of nondesign basis events are all possible periods of RVHVS usage. The RVHVS is designed to vent a volume of hydrogen at system design pressure and temperature approximately equivalent to one-half of the reactor coolant system volume in one hour.

The system discharges into a well ventilated area of the Containment to the refueling cavity.

The flow diagram of the Reactor Vessel Head Vent System is shown in Figure 1.

The RVHVS consists of two parallel flow paths with redundant isolation valves in each flow path. The venting operation uses only one of these flow paths at any one time. The equipment design parameters are listed in Table 1.

The active portion of the RVHVS consists of four 1-in. open/close solenoid operated isolation valves. The Reactor Vessel Head Vent System is connected to the existing 3/4-in. vent pipe which is located near the center of the reactor vessel head.

The system design with two valves in series in each flow path minimizes the possibility of reactor coolant pressure boundary leakage. The isolation valves in one flow path are powered by one vital power supply and the valves in the second flow path are powered by a second vital power supply. The isolation valves are fail closed, normally closed active valves. The isolation valves are also included in the Westinghouse valve operability program which is an acceptable alternative to Regulatory Guide 1.48. These valves will be qualified to IEEE-323-1974, -344-1975, and -382-1972.

If one single active failure prevents a venting operation through one flow path, the redundant path is available for venting. Similarly, the two isolation valves in each flow path provide a single failure method of isolating each of the venting subsystems. With two valves in series, the failure of any one valve or power supply will not inadvertently open a vent path. Thus, the combination of safety grade train assignments and valve failure modes will not prevent vessel head venting nor venting isolation with any single active failure.

The RVHVS has two normally deenergized valves in series in each flow path. This arrangement eliminates the possibility of a spuriously opened flow path due to the spurious movement of one valve. As such, power lockout to any valve is not considered necessary.

The reactor vent piping branches into two redundant flow paths through 3/8-in. orifices. These orifices form the Safety Class 1 to Safety Class 2 transition. The system is orificed to limit the blowdown from a break downstream of either of the orifices to within the capacity of one of the centrifugal charging pumps.

A break of the RVHVS line upstream of the orifices would result in a small LOCA of not greater than 1 in. diameter. Such a break is similar to those analyzed in WCAP-9600 (Reference 1). Since a break in the head vent line would behave similarly to the hot leg break case presented in WCAP-9600, the results presented therein are applicable to a RVHVS line break. This postulated vent line break, therefore, results in no calculated core uncovering.

All piping and equipment used in the Reactor Vessel Head Vent System from the existing vent pipe to the orifices is designed and fabricated in accordance with ASME, Section III, Class 1 requirements. From the orifices to the first anchor downstream of the second isolation valves, all equipment is designed and fabricated in accordance with ASME, Section III, Class 2 requirements. The piping downstream of the first anchors downstream of the second isolation valves is non-nuclear safety.

The RVHVS provides for venting the reactor vessel head by using only safety grade equipment. The RVHVS satisfies applicable requirements and industry standards including ASME Code classification, safety classification, single-failure criteria and environmental qualification.

## Supports

The vent system piping is supported to ensure that the resulting loads and stresses on the piping and on the vent connection to the reactor are acceptable.

The support design for attaching the head vent system piping to the reactor vessel lifting leg is a two-part clamp configuration called a double-bolt riser clamp. The clamp and associated bolts, nuts, spacers, and washers are made of stainless steel. A gap condition exists at the 1-in. head vent pipe and the support clamp to allow for thermal expansion in the vertical direction.

The support design for attaching the head vent system piping to the CRDM Seismic Support Platform constitutes a two-part clamp configuration called a double-bolt clamp bracket. This clamp support is used to rigidly support the piping in the vertical direction and the horizontal direction perpendicular to the pipe axis. The pipe is free to move in its axial direction. The clamp and associated nuts, bolts, spacers, and washers are made of stainless steel and with high strength hold down washers threaded into the deck of the CRDM Seismic Support Platform. An overall view of the entire vent system is shown in Figure 1.

All supports and support structures comply with the requirements of the AISC Code.

## Analytical Considerations

The analysis of the reactor vessel head vent piping is based on the following plant operating conditions defined in ASME Code, Section III:

1. Normal Condition:

   Pressure, deadweight, and thermal expansion analysis of the vent pipe during: a) normal reactor operation with the vent isolation valves closed and b) post-refueling venting.

2. Loads generated by the Operating Basis Earthquake (OBE).

3. Faulted Condition:

   Loads generated by the Safe Shutdown Earthquake (SSE) loads generated by valve thrust during venting. In accordance with ASME III, faulted conditions are not included . . fatigue evaluations.

The Class 1 piping used for reactor vessel head vent is 3/4 in. and 1 in. schedule 160 and, therefore, in accordance with ASME III, is analyzed following the procedures of NC-3600 for Class 2 piping.

For all plant operating conditions listed above, the piping stresses are shown to meet the requirements equations (8), (9), and (10) or (11) of ASME III, NC-3600, with maximum operating temperature of 620°F and a design pressure of 2485 psig.

References

1. WCAP-9600, "Report on Small Break Accidents for Westinghouse NSSS System", June, 1979 (specifically, Case F, Section 3.2).

# REACTOR VESSEL HEAD VENT SYSTEM EQUIPMENT DESIGN PARAMETERS

## Valves

| | |
|---|---|
| Number (includes one manual valve) | 5 |
| Design pressure, psig | 2485 |
| Design temperatures, °F | 650 |

## Piping

| | |
|---|---|
| Vent line, nominal diameter, in. | 3/4 and 1 |
| Design pressure, psig | 2485 |
| Design temperature, °F | 650 |
| Maximum Operating Temperature | 620 |

# REACTOR VESSEL HEAD VENT