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SUBJECT: Forwards addl info re Unit 2 boron injection tank (BIT) thermal relief valve,in response to NRC 871206 request.Info identifies util actions re monitoring of BIT thermal relief valve leakage prior to 871010 installation.

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JAMES D. SHIFFER
VICE PRESIDENT
NUCLEAR POWER GENERATION

November 6, 1987

PG&E Letter No.: DCL-87-267

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington D.C. 20555

Re: Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Boron Injection Tank Thermal Relief Valve

Gentlemen:

During discussions with the NRC Staff on October 6, 1987, regarding a small leak from the Diablo Canyon Power Plant (DCPP) Unit 2 boron injection tank (BIT) thermal relief valve, the NRC Staff requested further information regarding: (a) clarification of the Final Safety Analysis Report (FSAR Update) description of the BIT thermal relief valve installation, (b) potential valve leakage rates and doses from radioactivity associated with postulated post-LOCA (loss-of-coolant accident) conditions, and (c) the DCPP Leak Reduction Program for systems outside containment.

The enclosure is provided to respond to this information request and further to identify actions that PG&E took regarding monitoring of BIT thermal relief valve leakage prior to installation of a new BIT relief valve on October 10, 1987.

Kindly acknowledge receipt of this material on the enclosed copy of this letter and return it in the enclosed addressed envelope.

Sincerely,

W. B. Kaefer /for

J. D. Shiffer

Enclosure

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ENCLOSURE

DIABLO CANYON POWER PLANT
UNIT 2 BORON INJECTION TANK THERMAL RELIEF VALVE

During discussions with the NRC Staff on October 6, 1987, regarding a small leak from the Diablo Canyon Power Plant (DCPP) Unit 2 boron injection tank (BIT) thermal relief valve, the NRC Staff requested further information regarding: (a) clarification of the Final Safety Analysis Report Update (FSAR Update) description of the BIT thermal relief valve installation, (b) potential valve leakage rates and doses from radioactivity associated with postulated post-LOCA (loss-of-coolant accident) conditions, and (c) the DCPP Leak Reduction Program for systems outside containment. The following is provided to respond to this information request and further to identify actions that PG&E has taken regarding monitoring of BIT thermal relief valve leakage prior to installation of a new relief valve.

A. CLARIFICATION OF FSAR UPDATE DESCRIPTION OF THE BIT THERMAL RELIEF VALVE INSTALLATION

1. Thermal Relief Valve Design and Release Path

The control of potential leakage and discharges of overpressure protection devices installed on the post-LOCA recirculation system located outside of containment in the auxiliary building is based on the design criterion that discharge of such pressure relieving devices which can relieve during recirculation is piped to the pressurizer relief tank (PRT). All such pressure relief valves on the emergency core cooling system (ECCS) process lines in the auxiliary building are routed to the PRT. The BIT thermal relief valve has a setpoint of 2735 psig, which is above the design shutoff head of 2670 psig for the high head charging pumps, and is not intended to relieve pressure generated by operation of the ECCS.

The thermal relief valve is designed to relieve pressures generated by the thermal expansion of the fluid in the BIT if the tank were inadvertently isolated from the rest of the system in a water solid condition while the tank heaters and associated heat tracing were energized. In this event, only a small volume of fluid would be discharged by the thermal relief valve in order to prevent tank overpressurization. Since the volume of discharged fluid would be small and the conditions requiring relief valve operation are not associated with post-LOCA recirculation operation, this valve discharge is not piped to the PRT. The latest nuclear steam supply system vendor design recommends that the BIT thermal relief valve

discharge be directed to the liquid holdup tank. Due to plant layout considerations, DCPP used the auxiliary building sump in lieu of the liquid holdup tank.

2. Modified FSAR Update

The next DCPP FSAR Update will clarify the intent of the statement regarding relief valve discharge to a closed system in Section 6.3.3.2.6, and the second paragraph of Section 6.3.3.2.7 will be modified to read:

> Pressure relieving devices with setpoints below the shutoff head of the high pressure ECCS pumps, from portions of the ECCS located outside of containment that might contain radioactivity, discharge to the pressurizer relief tank. The boron injection tank relief valve, 8852, shown in Figure 3.2-09, Sheet 3 of 10, has a setpoint above the shutoff head of the ECCS pumps and cannot be opened by the ECCS pressures during post-LOCA recirculation. This valve relieves to the liquid radwaste system via an open drain.

If appropriate, the description will also identify the newly installed relief valve (596).

B. POTENTIAL VALVE LEAKAGE RATES AND DOSE CALCULATIONS

1. Post-LOCA Recirculation Leakage Values Identified in the FSAR

FSAR Update Table 6.3-9 provides a listing of nonfaulted ECCS recirculation loop components external to containment and their maximum potential leakage rates during post-LOCA recirculation operation. The maximum potential leakage rates are defined as the design leakage rates from the various components. Combining these component leakage rates gives a conservative value of 1910 cc/hour. This leakage rate is based on the characteristics of the mechanical components and was not selected as a limiting value based on dose considerations. This leakage rate was used for the ECCS recirculation component design leakage given in FSAR Update Chapter 15 offsite dose calculations. It is not intended to be the maximum allowable leakage rate in the auxiliary building including a faulted component.

The results of conservative dose calculations recently performed by PG&E show that the limits of 10 CFR 100 and 10 CFR 50 General Design Criterion 19 are not exceeded when considering a post-LOCA recirculation loop leakage to the auxiliary building of approximately six times the 1910 cc/hr leakage in conjunction with a 30-minute duration 50 gpm leakage rate from residual heat removal (RHR) pump seal failure. This conservative calculation assumed no credit for the charcoal filtration in the auxiliary building. With charcoal filtration, the dose limiting leakage rate would be

considerably higher. PG&E's calculations are discussed in more detail below.

2. Currently Analyzed FSAR Update Cases

Chapter 15 of the FSAR Update describes two cases concerning leakage into the auxiliary building following a LOCA. These are a 50 gpm leakage rate from RHR pump seals, which assumes the resulting airborne material is filtered by the auxiliary building ventilation system in the safeguards mode, and the minor design leakage at a rate of 1910 cc/hr from various sources as identified in FSAR Update Table 6.3-9, which is not filtered prior to being released into the environment. These cases are described in detail in Section 15.5.17.8 of the FSAR Update.

The offsite thyroid doses from all sources for the two analyzed cases are shown in FSAR Update Tables 15.5-23 and 15.5-25 and are calculated for the 2-hour site boundary and 30-day low population zone (LPZ). Whole body doses are also included in the tables, but are a small fraction of the thyroid dose and are not limiting. The FSAR Update doses are summarized below:

Thyroid Doses, rem

	2-Hour Site Boundary	30-Day LPZ
Containment leakage	95.9	17.7
Aux. building releases (50 gpm leak)	26.9	1.1*
TOTALS	122.8	18.8
Containment leakage	95.9	17.7
Aux. building releases (1910 cc/hr leak)	7.5*	2.1
TOTALS	103.4	19.8

* Not in the FSAR Update. Value from the original EMERALD computer output for the analyses reported in the FSAR Update.

To determine the maximum dose to control room personnel, the 50 gpm auxiliary building leakage was used. The details are provided in Section 15.5.17.10 of the FSAR Update. It was assumed that the radioactivity would enter the control room through two pathways: 1) from the control room ventilation pressurization air intakes

through charcoal filtration, and 2) from air infiltration due to personnel ingress and egress. The resulting 30-day thyroid dose is the sum of airborne radioactivity from containment leakage and releases from the auxiliary building. A control room dose of 21.1 rem is shown in FSAR Update Table 15.5-33.

The maximum dose allowed during an accident is defined in 10 CFR 100 and 10 CFR 50 General Design Criterion (GDC) 19 for site boundary (300 rem), offsite populations (300 rem), and control room personnel (30 rem).

3. Additional Leakage Rate Allowed in Auxiliary Building

Since the accident cases analyzed for the FSAR Update result in doses substantially less than NRC established limits, some amount of additional leakage in the auxiliary building above that used in the original analyses is acceptable during a LOCA.

A computer code (LOCADOSE) was used to determine the site boundary, LPZ, and control room doses from additional leakage occurring in the auxiliary building. The assumptions used for the analysis were very conservative and were the same as those for the design basis accident small leakage case described in FSAR Update Section 15.5.17.8 and shown in FSAR Update tables 15.5-3, 15.5-6, 15.5-24, and 15.5-32.

With the difference between the 10 CFR limits and the FSAR Update doses previously given, the total additional leakages that can be sustained in addition to the 50 gpm RHR pump seal and containment leakages were calculated. The results of the calculation showed that the 10 CFR 50 GDC 19 control room limit of 30 rem was reached with a 11,200 cc/hr leakage rate into the auxiliary building in addition to the 50 gpm already analyzed in the FSAR Update.

The leakage from the BIT thermal relief valve was occurring at 30 psig pressure. Since the BIT will be at an average pressure of about 850 psig during recirculation (the sum of the RHR and charging pump discharge pressures), the maximum allowable leakage rate of 11,200 cc/hr was extrapolated from 850 psig to 30 psig. The maximum allowable leak at 30 psig was determined to be 35 cc/min from the BIT thermal relief valve.

In the discussion with the NRC Staff on October 6, 1987, PG&E reported that the maximum allowable leakage rate could be 41 cc/min from the BIT at 30 psig. This assumed the BIT would be at 600 psig pressure during recirculation. A more conservative evaluation of the BIT pressure resulted in a change to the average pressure of 850 psig during recirculation. This resulted in a reduction of the maximum allowable leakage rate from 41 cc/min to 35 cc/min at 30 psig.

This analysis assumes single passive failures on both the RHR pump and the charcoal filter. PG&E is currently recalculating the maximum allowable leakage in the ECCS recirculation path.

4. Access Path Dose Rates with Additional Leakage

The DCPP Units 1 and 2 radiation shielding review, dated June 1984, was re-reviewed to determine if the leakage from the BIT relief valve would affect access to vital areas of the plant.

The maximum whole-body dose calculated for sample collection from the postaccident sampling system (PASS) was 1.38 rem, compared to the 5 rem limit. A factor for equipment leakage, such as the BIT relief valve leakage, was included in the calculation of 1.38 rem. This factor contributed one percent to the total whole-body dose. Consequently, additional leakage would not have any significant effect on the calculated whole body dose rate.

Access to other vital areas is possible since, as described above, the whole body dose from direct radiation is the limiting factor. The additional iodine in the auxiliary building from the BIT valve leak can be managed in accordance with established procedures by means of self-contained breathing apparatus and potassium iodide.

5. FSAR Update Changes

The analysis discussed above assumed single passive failures of both the RHR pump seal and the auxiliary building charcoal filtration. PG&E is presently completing a comprehensive reanalysis of the maximum allowable leakage from ECCS components to the auxiliary building during post-LOCA recirculation operations. The next FSAR Update will describe the recalculated leakage that can occur in the auxiliary building during a LOCA before the regulatory limits are reached.

C. DCPP LEAKAGE REDUCTION PROGRAM AND SURVEILLANCE FOR SYSTEMS OUTSIDE CONTAINMENT

1. DCPP Leakage Reduction Program and Relationship to FSAR

PG&E has established and implemented a program of preventive maintenance to reduce leakage to as-low-as-practical levels from systems outside of containment likely to contain highly radioactive fluids during a serious transient or accident. This program was implemented in response to NUREG-0578, "Clarification of TMI Action Plan Requirements." PG&E committed to performing operating pressure leak tests on appropriate portions of the safety injection system, residual heat removal system, and nuclear steam supply sampling system. This commitment is implemented by Surveillance Test Procedure (STP) M-86, "Leak Reduction of Systems Outside Containment Likely to Contain Radioactive Materials Following an Accident (NUREG-0578, TMI-9)." This STP, performed during every refueling

outage, requires leak tests of portions of the safety injection system and boron injection tanks (STP M-86A), residual heat removal system (STP M-86C&D), and centrifugal charging pumps (STP M-86G). Pressurized systems are visually inspected for leakage into the auxiliary building environment. Where feasible, systems containing liquid are pressurized with a hydrostatic pressure test pump, and leakage is determined by measuring level changes of makeup water in a graduated tank. While each system is pressurized, it is visually inspected to identify sources of leakage and estimate the rate of leakage from the system.

The current acceptance criteria for STP M-86 require that the total external leakage allowable for the pump rooms, safety injection, residual heat removal, centrifugal charging pumps, and post-LOCA sampling system room is less than or equal to 1 gpm, and the total external leakage for the post-LOCA sampling system (sentry) room is less than or equal to 2.3 gpm. This leakage rate is based on operator accessibility of these areas during post-LOCA conditions. However, the STP suggests that external leakage should be as low as possible with all visible leaks stopped. At the conclusion of PG&E's reanalysis of ECCS leakage to the auxiliary building during post-LOCA recirculation operation, STP M-86 will be revised so that the outage based test program will reflect the leakage allowable to prevent exceeding offsite and control room dose limits required by 10 CFR 100 and 10 CFR 50 GDC 19, respectively.

2. Installation of a New Thermal Relief Valve

On October 10, 1987, PG&E installed a new thermal relief valve (596) on the BIT, providing a replacement relief path. The leaking relief valve (8852) has been gagged. There is no observable leakage from the new relief valve or the gagged relief valve.

3. Procedure and Frequency for Surveillance of BIT Relief Valve Leakage

To quantify leakage from the BIT relief valve (8852) prior to installation of the new relief valve, a temporary procedure, "Measurement of Leakage from BIT Outlet Relief Valve Discharge Line," was prepared for monitoring the leakage from the discharge line. The temporary procedure states that the monitoring frequency will be determined by DCPP management. This leakage was measured at the point at which it enters the drain scupper by taking samples to determine an average leakage rate in cc per minute. From the conservative calculations performed by PG&E, it was determined that corrective actions would have been required at a leakage rate extrapolated to accident conditions of 11,200 cc/hour. When the BIT thermal relief valve leakage was added to other known auxiliary building ECCS recirculation piping systems leakage and the leakage extrapolated to expected accident pressures, the total measured leakage never exceeded the 11,200 cc/hr value.