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PG&E Letter DCL-01-104

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
License Amendment Request 01-04
Revision To Technical Specifications 3.9.4 Containment Penetrations

Dear Commissioners and Staff:

In accordance with 10 CFR 50.90, PG&E is submitting an application for amendment to Facility Operating License Nos. DPR-80 and DPR-82 for Diablo Canyon Power Plant Units 1 and 2, respectively. The enclosed license amendment request (LAR) proposes to revise the Limiting Condition of Operation for Technical Specification (TS) 3.9.4, "Containment Penetrations," to allow the equipment hatch, both Personnel Air Lock doors and both Emergency Air Lock doors to remain open, and penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative control, during core alterations and movement of irradiated fuel assemblies. In addition, this LAR proposes to revise TS 1.1, "Definitions," for Dose Equivalent I-131, to allow the use of the thyroid dose conversion factors, listed in the International Commission on Radiological Protection Publication 30, "Limits for Intakes of Radionuclides by Workers."

Enclosure 1 provides a description of the proposed change, the supporting evaluation, and PG&E's determination that the proposed changes do not involve a significant hazard. Enclosure 2 provides a markup of the TS Changes. Enclosure 3 provides a markup of the TS Bases to reflect the proposed change (for information only). Enclosure 4 provides the new proposed TS pages. TS Bases changes will be implemented in accordance with TS 5.5.14 Bases Control Program as part of the implementation of this amendment, upon NRC approval of this amendment application.

This LAR is not required to address a safety concern. Therefore, PG&E requests that it be reviewed on a medium priority. PG&E requests that the review of this LAR be completed by April 15, 2002 to support its implementation during the Unit 1 eleventh refueling outage scheduled to begin April 15, 2002. Receipt of this amendment is not required to conduct the outage or to restart the unit following the outage. However, implementation of the requested TS changes during the outage will allow planned outage work to proceed in conjunction with critical path activities.

A001

Sincerely,

Greg Rueger

Gregory M. Rueger

Enclosures

cc: Edgar Bailey, DHS
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Diablo Distribution

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY	) Docket No. 50-275) Facility Operating License) No. DPR-80)
Diablo Canyon Power Plant Units 1 and 2	) Docket No. 50-323) Facility Operating License) No. DPR-82

AFFIDAVIT

Gregory M. Rueger, of lawful age, first being duly sworn upon oath says that he is Senior Vice President - Generation and Chief Nuclear Officer of Pacific Gas and Electric Company; that he has executed License Amendment Request 01-04 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

Greg Rueger
Gregory M. Rueger
Senior Vice President – Generation and Chief Nuclear Officer

Subscribed and sworn to before me this 17th day of October, 2001

Sandra Eatherly
Notary Public
County of SAN LUIS OBISPO
State of California

SANDRA EATHERLY
Commission # 1225422
Notary Public - California
San Luis Obispo County
My Comm. Expires Jul 18, 2003

EVALUATION

1.0 DESCRIPTION

This letter is a request to amend Facility Operating License Nos. DPR-80 and DPR-82 for Diablo Canyon Power Plant (DCPP) Units 1 and 2, respectively.

This License Amendment Request (LAR) proposes to revise Technical Specification (TS) 3.9.4, "Containment Penetrations", to allow the equipment hatch, both containment personnel air lock (PAL) doors, and both emergency air lock (EAL) doors to remain open, and penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative control, during core alterations and movement of irradiated fuel assemblies. In addition, this LAR proposes to revise TS 1.1, "Definitions," for Dose Equivalent I-131, to allow the use of the thyroid dose conversion factors, listed in the International Commission on Radiological Protection (ICRP) Publication 30, "Limits for Intakes of Radionuclides by Workers."

The proposed change will provide greater flexibility in outage scheduling, reduction of maintenance on equipment (such as PAL and EAL doors), and reduction in potential personnel overall exposure following a fuel handling accident (FHA) inside containment without significantly increasing dose to the public.

2.0 PROPOSED CHANGE

The proposed change would revise the TS 3.9.4 Limiting Condition of Operation (LCO) as follows:

1. TS 3.9.4 LCO part "a" would be changed from requiring "the equipment hatch to be closed and held in place by four bolts" to "the equipment hatch being capable of being closed and held in place with four bolts."

2. TS 3.9.4 LCO part "b" would be changed from "one door in each air lock closed" to "one door in each air lock capable of being closed."

3. A note would be added to TS 3.9.4 LCO part "c" allowing "Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls."

In addition, the proposed change will modify the definition of Dose Equivalent I-131 in TS 1.1, "Definitions," as follows:

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites"; or those listed in Table E-7 of NRC Regulatory Guide 1.109, Rev. 1, October, 1977; or those listed in International Commission on Radiological Protection Publication 30, "Limits for Intakes of Radionuclides by Workers," 1979.

In summary, the proposed amendment would allow the equipment hatch, both containment PAL doors, and both EAL doors to remain open (provided the equipment hatch, one PAL door, and one EAL door are capable of being closed), and penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative control, during core alterations and movement of irradiated fuel assemblies. In addition, the proposed change will allow the use of the thyroid dose conversion factors provided in ICRP-30 for determining DOSE EQUIVALENT I-131, which is consistent with current NRC expectations.

The TS Bases will be revised to reflect the changes to LCO 3.9.4. A markup of those changes is provided in Enclosure 3 for information. These TS Bases changes will be implemented in accordance with TS 5.5.14, "Technical Specification (TS) Bases Control Program," as part of the implementation of this amendment, upon NRC approval of this amendment application.

3.0 BACKGROUND

Equipment Hatch

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. Internal pressure loads following an accident would cause the cover to bear against the opening band enhancing the sealing function. The bolts do not resist the loss of coolant accident pressure loads, but merely hold the hatch in place during operation and resist seismic loads. After the bolts are disengaged, the hatch cover can be moved horizontally using a monorail mounted above the hatch. The opening and closing of the equipment hatch is done manually without any required electrical power. Therefore, the closure of the equipment hatch is not adversely affected by any loss of power.

Maintaining the equipment hatch open, but capable of being closed, makes it easier to maintain a clean, safe working environment inside containment. With the equipment hatch closed during refueling activities, radioactive materials, equipment and waste accumulates in containment. This accumulation provides a source of additional dose to personnel inside the containment and creates personnel safety issues. If the equipment hatch door were allowed to be open during refueling activities, materials, and equipment that cannot safely be move through the PAL can easily and efficiently be moved in and out of containment during fuel movement.

During a typical refueling outage, work scope and related critical path sequencing is related to the availability of the equipment hatch for movement of equipment. Maintaining the equipment hatch closed can cause equipment movement delays and can delay the final closure of the equipment hatch door prior to plant heat-up. If the equipment hatch were to remain open during movement of irradiated fuel or core alterations, this would allow equipment to be moved in and out of the containment more effectively and efficiently during fuel movement.

Personnel and Emergency Air Locks

The PAL and EAL are welded steel assemblies consisting of two doors with double gaskets in series. The PAL and EAL doors are mechanically interlocked so that one door cannot be opened unless the second door is sealed. A pressure-equalizing valve at each door is provided to equalize pressure across the doors when personnel are entering or leaving containment. Provisions are made to bypass the interlocks to permit both doors to be opened, when the containment pressure is zero psig and it is safe to do so.

From a practical standpoint, TS 3.9.4 currently will not prevent all radioactive releases from the containment following a postulated FHA. A large number of people are in containment during a refueling outage, including during fuel movement and core alterations. Should a FHA occur, it would take a number of cycles of the PAL or EAL doors to evacuate personnel from containment. With each cycle of the PAL or EAL doors, more containment air would be released. While waiting for their turn to exit, the workers would be exposed to the released activity. As the doors are cycled for the exiting personnel, there would be a release of activity out of containment. Under the proposed change, containment could be evacuated more rapidly and efficiently, and then sealed. This would reduce the dose to workers in the event of an accident while maintaining acceptable doses to the public.

Penetrations

The proposed change to allow the containment penetration flow path(s) to remain open, while under administrative controls, implements the NRC approved TS traveler TSTF-312, Revision 1 (Ref. 3). Furthermore, this approach is consistent with the administrative controls currently allowed by DCPP TS for higher operational modes. Current provisions in TS 3.6.3, "Containment Isolation

Valves," allow penetration flow paths to be unisolated under administrative controls in modes 1 though 4. These modes are more significant than during refueling operations due to the Reactor Coolant System energy and potential to provide a significant motive force for the expulsion of radionuclides subsequent to a design basis accident.

Based on the acceptability of administrative controls during higher modes of operation, a similar allowance should be acceptable for penetrations that are open during fuel movement or core alterations provided appropriate administrative controls are utilized. In addition, during core alterations and irradiated fuel movement inside containment the refueling cavity water level is 23 feet or greater for TS 3.9.7, "Refueling Cavity Water Level." Under these conditions the potential for an accident resulting in containment pressurization from core boiling is minimal and has been analyzed not to take place prior to the 4-hour required containment closure of TS 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level."

During the performance of Local Leak Rate Testing (LLRT), certain containment isolation valves (i.e., those subject to Type C testing) are required to be opened in order to drain the penetration piping, providing direct access from the containment atmosphere to the outside atmosphere. Therefore, under current restrictions, LLRT tests cannot be performed during core alterations or fuel movement inside containment. This restriction complicates the logistics for performing LLRT and reduces overall refueling outage efficiency. The proposed change to TS 3.9.4 would allow containment penetrations to be open during core alterations or the movement of irradiated fuel assemblies within containment, provided that the penetrations are under administrative controls and capable of being closed by a manual or automatic isolation valve, blind flange, or equivalent.

Dose Equivalent I-131

The proposed amendment would allow, as an additional option, the use of the thyroid dose conversion factors listed in the ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers," dated 1979, for the determination of the DOSE EQUIVALENT I-131. As stated in TS 1.1, "Definitions," "The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present." Currently TS 1.1 allows the use of conversion factors from Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," and those listed in Table E-7 of NRC Regulatory Guide (RG) 1.109, Rev. 1, October, 1977.

The allowance of use of the thyroid conversion factors listed in ICRP-30 is considered a change in the analysis methodology that requires prior NRC review and approval. The use of these ICRP-30 thyroid conversion factors is consistent with current industry and NRC expectations.

The new FHA radiological consequences analysis for inside containment, which was performed in support of this proposed amendment, utilizes the thyroid dose conversion factors listed in the ICRP–30. The current FHA analysis utilized the thyroid dose conversion factors listed in RG 1.109, consistent with TS 1.1.

4.0 TECHNICAL ANALYSIS

The proposed changes would allow the equipment hatch, containment penetrations, and the EAL and PAL doors to be open under administrative controls during core alterations and/or during movement of irradiated fuel assemblies within containment, provided they are capable of being closed. In allowing these doors and penetrations to be open during core alterations or movement of irradiated fuel concerns that radioactive materials could potentially be released to the outside environment during applicable accidents must be addressed. The applicable postulated accidents that could result in a release through these openings include a FHA and a loss of residual heat removal (RHR) cooling event. Discussed below is the justification for the proposed changes considering these two accident conditions.

Fuel Handling Accident

During movement of irradiated fuel assemblies within containment, the most severe radiological consequences will result from a FHA. Historically, to limit the potential occupational and public radiological exposure to a potential FHA inside containment, TS 3.9.4, "Containment Penetrations," has required containment closure. However, recent evaluations of the FHA indicate that containment closure is not necessary to maintain any release from a FHA to well within the requirements of 10 CFR 100. (Ref. 1)

The allowance to have the equipment hatch, PAL and EAL open, and penetration flow paths unisolated during core alterations and fuel movement, is based on (1) a new dose calculation of a FHA which does not take credit for the containment, and (2) commitments to administrative procedures to ensure that in the event of a FHA the closure of the open equipment hatch, PAL, EAL, and open penetrations will be initiated immediately upon identifying a containment closure requirement to further limit any radioactive release to the outside atmosphere.

Fuel Handling Accident Analysis

As a part of these proposed changes PG&E performed a new FHA inside containment analysis to ensure that the possible radiological releases resulting from the changes to the TS are still "well within" twenty-five percent of the 10 CFR 100 values of 300 Rem thyroid and 25 Rem whole body. This is the acceptance criteria provided in NUREG-0800, Section 15.7.4 (Ref. 5) and equivalent to 75 Rem thyroid and 6 Rem whole body. In addition, although not previously provided specifically for this accident, the new analysis evaluated the control room dose consequences to assure that they also remain well below the GDC 19, "Control Room" (Ref. 6) equivalent limits of 30 Rem thyroid and beta skin, and 5 Rem whole body.

The following tables show the previous and new offsite boundary and new control room dose consequences calculated for the FHA inside containment. The previous values were calculated with the containment closed. The new dose consequences were calculated with the EAL/PAL doors, equipment hatch and penetrations open. No previous values for the control room dose are given because there was no requirement for the control room dose to be determined for the previous FHA analysis.

Table 1 – Previous Dose Consequences

Dose (rem)	Thyroid		Whole Body	
	Analysis	Limit	Analysis	Limit
2 hr Exclusion Area Boundary	18.4	300	0.31	25
30 day Low Population Zone Boundary	0.76	300	0.013	25

Table 2 – New Dose Consequences

Dose (rem)	Thyroid		Whole Body	
	Analysis	Limit	Analysis	Limit
2 hr Exclusion Area Boundary	60.62	300	0.4281	25
30 day Low Population Zone Boundary	2.521	300	0.0178	25
30 day Control Room	11.56	30	0.0072	5

Fuel Handling Accident Analysis Methodology

The new FHA inside containment analysis was performed using the Bechtel Standard Computer Program LOCADOSE, NE319, Release 6.0. This analysis provides the resultant radiological doses for the control room, Exclusion Area

Boundary, and Low Population Zone. The LOCADOSE, NE319 program has been accepted for use in the industry and has been verified and validated.

The code used is designed to calculate radioactive material activities within regions in the plant, radioactive releases from regions of the plant, and doses and dose rates within regions of the plant and offsite locations. The solutions are obtained by solving a system of coupled differential radiation transport equations with boundary values. The assumptions used in this analysis are consistent with RG 1.25 with the exceptions of pool decontamination factor and the ICRP-30 dose conversion factors. The assumptions and exceptions are discussed below.

FHA Analysis Assumptions

The DCPP design basis FHA is defined as the dropping of a spent fuel assembly in the fuel handling building or inside containment. Both accidents assume the rupture of the cladding of all the fuel rods (264 rods) in one assembly. In addition, the inside containment case assumes that:

1. With the PAL doors and the equipment hatch open during fuel movement, a radiological release path during the accident would be through these openings to the environment. RG 1.25, Section C.1.i, assumes that the radioactive material that escapes from the pool to the building is released from the building over a two-hour period. However, for the DCPP analysis, it is assumed that all of the radioactivity from the FHA will be instantaneously released to the environment through these openings. Therefore, the control room and offsite dose consequences are calculated based on this instantaneous release assumption. This maximizes the potential dose at the exclusion area boundary, low population zone boundary, and the control room.
2. The exfiltration rate from containment with the equipment hatch open is conservatively assumed to be 2.55E6 cubic feet per sec (the entire containment air volume will be released in one second) and the open PAL doors rate has been determined to be 1400 cubic feet per minute. As a result, the PAL exfiltration rate has no impact on the result of the dose consequence since the entire containment volume is already assumed to be released through the equipment hatch. Based on the smaller cross-sectional areas of the open EAL doors and any of the containment penetrations that may be open, the effects of these openings are bounded by the equipment hatch exfiltration rate and will have no impact on the analysis results.
3. The Thyroid Dose Conversion Factors used to calculate the doses from this event are included in LOCADOSE, NE319, Release 6.0. The Thyroid Dose Conversion Factors used in this calculation are based on ICRP-30 values as documented in Federal Guidance Report 11 and 12.

I-131	1.08E+06 (Rem/Ci)
I-132	6.44E+03 (Rem/Ci)
I-133	1.80E+05 (Rem/Ci)
I-134	1.07E+03 (Rem/Ci)
I-135	3.13E+04 (Rem/Ci)

4. The following assumptions are obtained from RG 1.25 (Ref. 7):

    a. The accident occurs 100 hours after shutdown. Fuel movement is not allowed prior to 100 hours after shutdown per the TS Bases for TS 3.9.4 and TS 3.9.7. Radiological decay and daughter product build-up was taken into consideration during this time to generate the source terms.

    b. The values assumed for individual fission product inventories are calculated for a composite source term assuming 105% full power operation (3580 MWt) at the end of core life immediately preceding shutdown, and include a radial peaking factor of 1.65.

    c. The iodine gap inventory is composed of inorganic species (99.75%) and organic species (0.25%).

    d. The effective overall refueling cavity decontamination factors (DF) for the inorganic and organic species is assumed to be 200. The previous DCPP fuel handling accident analysis inside containment used a refueling cavity DF of 100. The use of DF 200 is consistent with RG 1.183, titled "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors," Appendix B, which states that if the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively. This gives an overall effective decontamination factor of 200 (i. e., 99.5% of the total iodine release from the damaged rods is retained by the water). Based on DCPP TS 3.9.7, which requires at least 23 feet of water be maintained above the reactor vessel flange during fuel movement, a refueling cavity DF of 200 is a reasonable assumption. (Ref. 9)

    e. The retention of noble gases in the refueling cavity is negligible.

    f. The radioactive material that escapes from the refueling cavity to the containment building is released from the building through the purge line, PAL/EAL doors, and equipment hatch openings.

    g. All of the gap activity in the damaged rods is released, and consists of 10% of the total noble gases other than Kr-85, 30% of the Kr-85, and 10% of the total radioactive iodine in the rods at the time of the accident.

NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," is concerned with high fuel burnup, which may have a potential adverse impact on the gap inventory available for release in a FHA. The NUREG concluded that, "a slight increase in inventory and fuel-cladding gap-release fractions will occur for some fission products in those rods at extended burnup." As a result, it indicates that, in the case of Iodine-131 (I-131), the gap release fraction could increase from 0.04 to 0.12 as fuel burnup increased from 33,000 MWD/MTU to 60,000 MWD/MTU. The NUREG states "Regulatory Guide (RG) 1.25 procedures may be used for extended burnup fuel. These procedures give conservative values for noble gas release fractions that are above calculated values for peak rod burnup of 60 GWd/t, except for I-131, which may be up to 20% higher."

Although this NUREG indicates that the iodine levels could be higher in the high burnup fuel, the findings are based on very conservative assumptions that do not consider fuel assembly power levels that have a major effect on the available fission products. Based on the discussion below, the assumptions utilized in the DCPP analysis are sufficiently conservative to assure a specific penalty for high-burnup is not required, and that the 20% increase in I-131 does not need to be specifically considered as indicated in NUREG/CR-5009.

RG 1.25 provides established and acceptable assumptions for use in the FHA analysis, and DCPP has used these assumptions in their FHA analysis. Included in these assumptions for DCPP are assumed gap release fractions and radial peaking factors for individual fission product inventories. In addition, the source term used in the DCPP FHA inside containment analysis is conservatively based on a 105% of full power operation as compared to the 100% full power operation required by RG 1.25.

The use of a radial peaking factor allows for an uncomplicated treatment of the effects of power variations within the core. At DCPP, a peaking factor of 1.65 is assumed for operation. This peaking factor in its simplest form represents a multiplier to the fission product inventory of the fuel assembly and conservatively accounts for assemblies which operate at a higher than average power level. The highest-power assemblies in the DCPP cores are new fuel assemblies that operate at relative power levels between 120% and 140%, which is considerably below the equivalent 165% power level assumed in the peaking factor. Assemblies that have accumulated substantial burnup (e.g., greater than 40 GWD/MTU), physically cannot operate at these high power levels due to depletion of the uranium inventory and corresponding reactivity reduction within the assembly. Consequently, the higher burnup assemblies operate at relatively lower power levels ranging up to approximately 110%. This reduced operating power level results in a reduced fission product inventory. Considering this reduced power level, an additional factor of 20% added to the equivalent peaking factor would result in a multiplier for the high burnup fuel of approximately 1.3, which is well below the peaking factor used of 1.65.

As discussed above, a comparison of the relative power levels of low-burnup fuel assemblies to high-burnup assemblies at DCPP shows a substantially lower power level for the higher-burnup assemblies. This reduced operating power level results in a reduced fission product inventory, and this reduction in power level is sufficient to assure that the actual total fission product inventory is less than the application of a generic power peaking factor of 1.65, which is adequate to conservatively determine a source term for the DCPP FHA analysis. As a result, this source term based on these conservative assumptions adequately bounds the 20% increase in I-131 predicted by NUREG/CR-5009, (Ref. 8).

Loss of RHR Cooling Accident

During core alterations and the movement of irradiated fuel assemblies the other potential accident is a loss of RHR cooling, which may provide a release of radioactive materials into the containment atmosphere via core boiling. As a result, the proposed changes could result in a direct release path to the outside environment. However, the release of radioactive materials from core boiling due to the loss of RHR cooling would be insignificant if the event does not continue for an extended period of time and does not result in core uncovery and subsequent core damage.

If core boiling does continue, the containment may become pressurized, thereby providing a driving force for the containment atmosphere to be released through the open doors and penetrations to the outside environment. However, the radiological consequences of this release of radioactive materials due to core boiling, with no consideration for core uncovery and core damage, would be less than the radiological consequences arising from a postulated FHA. This is due to the total release inventory being limited to the reactor coolant system activity (corresponding to a 1% fuel defect and TS activity limit) being less than the total gap activities in the assumed damaged rods of a FHA at the earliest time core offloading may commence (100 hours after shutdown).

The time to core boil is estimated to be greater than 5 hours if a loss of RHR cooling event occurs at the beginning of the core offload with the water level in the refueling cavity at 23 feet or greater above the top of the reactor vessel flange. TS 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – High

Water," requires actions to be taken immediately to restore the RHR cooling capability if the RHR loop requirements are not met. In addition, operators are required to close all containment penetrations providing direct access from the containment atmosphere to the outside environment within the calculated time to boil or 4 hours. If an operator takes action to restore the RHR cooling capability or provides an alternative method of core cooling, then core boiling and subsequent containment pressurization would not take place. If the time to core boiling were exceeded, isolation of all containment penetrations per the TS 3.9.5 requirements would ensure that the release of radiation outside of the containment from this event would not be a concern.

Administrative Controls

Although the new FHA analysis demonstrates that the PAL/EAL doors, equipment hatch and penetrations are not required to be closed for the first two hours post-FHA, PG&E would require that closure of any of these open pathways be initiated immediately with completion of containment closure within approximately 30 minutes of a required containment evacuation to minimize offsite exposure. In addition, the containment penetrations with direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls comparable to those provided during power operation. To ensure that these containment closure requirements are met, administrative controls will be established and maintained for each of these potentially open pathways. These administrative controls will ensure that during core alterations and/or the movement of irradiated fuel assemblies:
1) appropriate personnel are aware of the open status of the containment;
2) specific individuals are designated, trained and readily available to effect the closure of the containment; 3) equipment and tools required to support containment closure activities are easily located and available; and 4) any potential obstruction (e.g., cables, hoses, etc.) that could prevent rapid closure of the containment can be quickly removed.

Conclusions

Based on the results of the new FHA inside containment analysis provided in support of this submittal, the risk to the health and safety of the public as a result of a FHA with the equipment hatch, PAL and EAL doors, and containment penetrations with direct access from the containment atmosphere to the outside atmosphere open is minimal. In the industry, actual fuel handling accidents that have occurred in the past have resulted in minimal or no releases, which support that the assumptions and methodology utilized in the new FHA inside containment analysis are very conservative.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Pacific Gas and Electric Company (PG&E) has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92 as discussed below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change would allow the containment equipment hatch, Personnel Air Lock (PAL) doors, Emergency Air Lock (EAL) doors, and penetrations to remain open during fuel movement and core alterations. These penetrations are normally closed during this time period in order to prevent the escape of radioactive material in the event of a Fuel Handling Accident (FHA) inside containment. These penetrations are not initiators of any accident and the probability of a FHA is unaffected by the position of these penetrations.

   The new FHA analysis with an open containment demonstrates the maximum offsite doses are well within (less than 25%) the limits specified in 10 CFR 100. These offsite dose values are also well within the acceptable limits provided in NUREG-0800, Section 15.7.4. This FHA analysis results in a maximum offsite dose of 60.62 Rem to the thyroid and 0.4281 Rem to the whole body. The calculated control room dose is also well below the acceptance criteria specified in General Design Criteria (GDC) 19. The analysis results in thyroid and whole body doses to the control room operator of 11.56 Rem and 0.0072 Rem, respectively. Although the offsite and control room dose values are increased by the proposed changes, the resulting values are still well within acceptable limits and do not significantly increase the consequences of a FHA.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change does not involve the addition or modification of any plant equipment. However, the proposed change does alter the containment closure configuration and method of operation of the plant during certain operational activities. The proposed change involves a change to the technical specification (TS) that would allow the equipment hatch door, the PAL doors, the EAL doors, and containment penetrations to be open during core alterations and fuel movement inside containment. This change only affects the containment barrier configuration of the plant during certain operational activities. Even allowing these doors and penetrations to be open, all of the resulting radiological consequences remain within acceptable limits and this configuration does not create the possibility of a new or different accident than previously evaluated.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No

This proposed change creates the potential for increased dose in the control room and at the site boundary due to a FHA. However, the new analysis demonstrates that the resultant doses are well within the 10 CFR 100 limits and well below the GDC 19 limits. In the case of the offsite dose values, they remain less than 25% of the 10 CFR 100 limits, which is considered acceptable in NUREG-0800, Section 15.7.4. Based on this, even though the dose values have increased from the previously calculated values, the margin of safety is not significantly reduced.

In the new analysis, the offsite and control room doses due to a FHA with an open containment have been evaluated using conservative assumptions, such as all airborne activity caused by the FHA in the containment is released instantaneously to the outside atmosphere, which ensures the calculation bounds the expected dose. The new analysis also assumes closure of the containment within two hours. As a result, requiring immediate initiation of the closure of the containment and completion of closure within approximately 30 minutes following a containment evacuation requirement from the FHA will reduce the potential offsite doses in the event of a FHA, and provides additional margin to the calculated offsite doses.

Therefore the proposed change does not involve a significant reduction in a margin of safety.

Based on the above evaluations, PG&E concludes that the activities associated with the above-described changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Regulatory Requirements and Guidance

5.2.1 Regulations

10 CFR 100, "Reactor Site Criteria," provides criteria for evaluating the radiological aspects of a proposed site. This includes offsite dose limits at the exclusion area limit and the low population zone limit, which must be met during and following a FHA.

10 CFR 50, Appendix A, General Design Criterion (GDC) 16, "Containment Design," requires that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment, and to assure that the containment design conditions important to safety are not exceeded for as long as the postulated accident conditions require. For the FHA analysis discussed in this license amendment request (LAR), the containment barrier is not credited for the first two hours of the event. However, at the two hour limit the containment barrier is credited.

10 CFR 50, Appendix A, GDC 19, "Control Room," requires adequate radiation protection under normal and accident conditions to permit access and occupancy without personnel receiving radiation exposure in excess of 5 Rem whole body, or its equivalent to any part of the body for the duration of the accident. This regulation specifies the control room exposure limits for a FHA.

10 CFR 50, Appendix A, GDC 54, "Piping Systems Penetrating Containment," requires that piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

GDC 56, "Primary Containment Isolation," describes the isolation provisions that must be provided for lines that connect directly to the containment atmosphere and which must penetrate primary reactor containment, unless it can be demonstrated that the isolation provisions for a specific class of lines are acceptable on some other defined basis.

GDC 61, "Fuel Storage and Handling and Radioactivity Control," requires that the fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. This GDC provides the requirement to design for a FHA.

5.2.2 Design Bases (Updated Final Safety Analysis Report (UFSAR))

UFSAR Section 15.4.5

The DCPP design basis FHA is defined as the dropping of a spent fuel assembly in the fuel handling building or inside containment. Both analyses assume the rupture of the cladding of all the fuel rods in the dropped assembly. UFSAR Section 15.4.5.2.1, which will be revised per the new FHA analysis, discusses the radiological consequences of the postulated FHA inside containment.

5.2.3 Approved Methodologies

U.S. NRC Regulatory Guide (RG) 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," provides NRC guidance, which describes a method acceptable to the NRC staff for licensee evaluation of the potential radiological consequences of a FHA.

U.S. NRC Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued in July 2000, provides new guidance on acceptable applications of alternative source terms. In Appendix B of this regulatory guide, guidance is provided on evaluating the radiological consequences of a FHA and acceptable overall decontamination factors of 200 if the water depth above the damaged fuel is greater than 23 feet. This decontamination factor is a relaxation over previous guidance and is used in the new FHA analysis in support of the proposed TS changes.

NUREG-0800, "U.S. NRC Standard Review Plan," Section 15.7.4, provides guidance to the NRC staff for the review and evaluation of system design features and plant procedures provided for the mitigation of the radiological consequences of postulated FHAs. Although DCPP is not subject to this NUREG, its guidance is used as a point of comparison in this submittal.

International Commission on Radiological Protection (ICRP) Publication 30, "Limits for Intakes of Radionuclides by Workers", dated 1979, provides thyroid dose conversion factors. These conversion factors replace those previously provided in Regulatory Guide 1.109, which is consistent with current NRC expectations.

NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors" provides guidance that an FHA offsite thyroid dose from I-131 could be increased by a factor of 1.2 for high-burnup fuel. DCPP has reviewed this guidance and found that the methodology and conservative assumptions concerning gap release fractions and radial peaking factors for individual fission product inventories used at DCPP and reflected in the FHA inside containment analysis adequately account for this guidance.

5.2.4 Analysis

The method of analysis used for evaluating the potential radiological consequences of the postulated FHA inside containment is consistent with RG 1.25, ICRP-30, RG 1.183, GDC 61, and the guidance in NUREG-0800, Section 15.7.4. The calculated doses are within the Standard Review Plan criteria of 6 Rem to the whole body and 75 Rem to the thyroid. The analysis presented in Section 15.4.5.2.1 of UFSAR, as will be revised per the new FHA analysis, demonstrates the adequacy of the plant design features and the plant procedures for the mitigation of the radiological consequences of postulated FHAs

5.2.5 Conclusion

The technical analysis provided in this submittal demonstrates that the consequent doses in the control room, at the exclusion area boundary, and the low population zone boundary are well within the limits of 10 CFR 100 and GDC 19, even without crediting the primary containment for confinement for the first two hours of the event. In addition, the proposed TS changes do not modify the containment barrier functions and those functions remain in compliance with GDC 16, 54, and 56.

6.0 ENVIRONMENTAL CONSIDERATION

The proposed changes in this LAR will change the requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, PG&E has evaluated the proposed changes and has determined that the changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

7.0 REFERENCES

1. 10 CFR Part 100, Paragraph 11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."
2. DCL-89-005, PG&E's response to Generic Letter 88-17, titled " Loss of Decay Heat Removal," dated January 6, 1989,
3. Industry/TSTF Standard Technical Specification Change Traveler TSTF-312, "Administratively Control Containment Penetrations, " Rev. 1.
4. FSAR Section 15.5.22, "Environmental Consequences of a Fuel Handling Accident."
5. NUREG-0800, Standard Review Plan, Section 15.7.4, Rev. 1, July 1981.
6. 10 CFR 50, Appendix A, General Design Criteria 19, Control Room.
7. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," dated March 23, 1972.
8. NUREG/CR-5009, "Assessment of the Use of Extended Burn-up Fuel in Light Water Reactors."
9. U.S. NRC Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued in July 2000

Precedents

On October 2, 2000, Amendment 169 to the license for Waterford Steam Electric Station, Unit 3, was approved to allow the containment equipment door, personnel airlocks, emergency airlocks, and other penetrations to remain open, but capable of being closed, during core alterations or movement of irradiated fuel in containment.

On April 16, 1999, amendment 203 to the Arkansas Nuclear One (ANO), Unit No. 2 plant license was approved to permit the equipment hatch and personnel air locks to remain open during fuel handling activities.

On August 10, 2000, Entergy Operations submitted a LAR for their Arkansas Nuclear One (ANO), Unit No. 2 plant, which would allow them to remove the closure requirements for containment penetrations during refueling operations. Instead the containment penetrations would be required to be capable of being closed.

MARKED-UP TECHNICAL SPECIFICATION

Remove Page	Insert Page
Page 1.1-3	Page 1.1-3
Page 3.9-3	Page 3.9-3
	Page 3.9-4

1.1 Definitions (continued)

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of NRC Regulatory Guide 1.109, Rev. 1, October, 1977,

INSERT 1

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;
2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

(continued)

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

INSERT 2

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts;

b. One door in each air lock closed; and

INSERT 3

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation valve.

INSERT 4

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS. AND	Immediately
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	Verify each required containment purge and exhaust ventilation isolation valves actuates to the isolation position on an actual or simulated actuation signal.	24 months

TECHNICAL SPECIFICATIONS INSERTS

Insert 1

or those listed in International Commission on Radiological Protection Publication 30, "Limits for Intakes of Radionuclides by Workers", 1979.

Insert 2

capable of being closed and held in place by four bolts;

Insert 3

One door in each air lock capable of being closed; and

Insert 4

-------------------------------------- NOTE --
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

MARKED-UP TECHNICAL SPECIFICATION BASES
(FOR INFORMATION ONLY)

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B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

~~During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed by automatic means. Since any potential for containment pressurization yields very low levels, the 10CFR50, Appendix J leakage criteria and tests are not required. (Ref. 1)~~

In MODES 1, 2, 3, and 4, t~~T~~he containment serves as a pressure boundary to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10CFR100. Additionally, in all operating modes the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions. However during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to maintain the pressure boundary can be less stringent. An analysis has been performed that shows by meeting the LCO, during CORE ALTERATION and movement of irradiated fuel assemblies in containment, the potential release as a result of a fuel handling accident (FHA) will remain well within the requirements of 10 CFR 100 limits.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. ~~During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.~~ The LCO requires that during CORE ALTERATIONS or the movement of irradiated fuel assemblies the equipment hatch must be capable of being closed and held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment Personnel A~~a~~ir L~~l~~ocks (PAL) and Emergency Air Lock (EAL), which are also part of the containment pressure boundary, provide a means for personnel and emergency access during MODES

1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each of these air locks has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when the PAL and EAL are not required to be closed, the door interlock mechanisms may be disabled, allowing both doors of each of the ~~an~~ air locks to remain open for extended periods when frequent containment entry is necessary.

~~During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed for normal entry and exit.~~

(continued)

BASES

BACKGROUND (continued)

Per the FHA inside containment analysis, there are no closure restrictions required to limit any release to well within the requirements of 10 CFR 100 limits for offsite dose as the result of a fuel handling accident during refueling. The LCO requirements for containment penetration closure are not provided to meet regulatory requirements, but rather to reduce the potential volume of the ~~ensure that a~~ release of fission product radioactivity within containment ~~will be restricted from escaping~~ to the environment. ~~The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.~~

The Containment Purge and Exhaust System includes two subsystems. The normal subsystem includes a 48 inch purge penetration and a 48 inch exhaust penetration in which the flow path is limited to being open 200 hour or less per calendar year. The second subsystem, a pressure equalization system provides a single 12 inch supply and exhaust penetration. The three valves in the 12 inch pressure equalization penetration can be opened intermittently. Each of these system are qualified to closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5.

In MODE 6, large air exchanges are necessary to conduct refueling operations. The normal 48 inch purge system is used for this purpose, and all four valves are closed by the ESFAS in accordance with LCO 3.3.6, "Containment Purge and Exhaust Isolation Instrumentation."

The pressure equalization system is disassembled and used in MODE 6 for other outage functions.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side if they are not opened under administrative controls. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. The fuel transfer tube is open but closure is provided by an equivalent isolation of a water loop seal. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, ventilation barrier for the other containment penetrations during fuel movements (Ref. 1).

APPLICABLE SAFETY ANALYSIS

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accident inside the containment ~~s, analyzed in Reference 2,~~ is based on ~~consists of~~ dropping a single irradiated fuel assembly of which all 264 fuel rods rupture. In addition the analysis assumes free and rapid communication of air from the containment to the outside environment; the accident occurs 100 hours after reactor shutdown; almost instantaneous release of the entire containment volume to the outside atmosphere; thyroid dose conversion factors based on ICRP 30 (Ref. 4); a radial peaking factor of 1.65 based on

105% full power operation; and the other guidance from RG 1.25. (Ref 5)

The requirements of LCO 3.9.7, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4,

(continued)

APPLICABLE SAFETY ANALYSIS (continued)

Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values.

Containment penetrations satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed or capable of being closed~~except for the OPERABLE containment purge and exhaust penetrations~~. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure times specified in the FSAR can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.

This LCO allows the equipment hatch to be open during CORE ALTERATIONS or the movement of irradiated fuel assemblies in containment provided it is capable of being closed and the following administrative controls are established and maintained: 1) appropriate personnel are aware of the open status of the equipment hatch; 2) specific individuals are designated, trained and readily available to effect the closure of the equipment hatch; 3) the tools and equipment required to support the closure of the equipment hatch and the location of these tools and equipment relative to the equipment hatch are controlled; and 4) any potential obstruction (e.g., cables, hoses, etc.) that could prevent rapid closure of the equipment hatch can be quickly removed to support immediate initiation of closure and completion of closure within approximately 30 minutes.

The LCO allows both of the personnel air lock (PAL) doors and both of the emergency air lock (EAL) doors to be open during CORE ALTERATIONS or the movement of irradiated fuel assemblies, provided one of the PAL doors and one of the EAL doors is capable of being closed. This is acceptable if administrative controls are established and maintained to ensure that: 1) appropriate personnel are aware of the open status of the PAL and/or EAL doors; 2) specific individuals are designated, trained and readily available to effect the closure of the PAL and EAL doors;~~,~~ 3) the tools and equipment required to support the closure of the PAL and EAL doors and the location of these tools and equipment relative to the PAL and EAL doors are controlled;~~,~~ and 4) any potential obstruction (e.g., cables, hoses, etc.) that could prevent rapid closure of the PAL and EAL doors can be quickly removed to support immediate initiation of closure and completion of closure within approximately 30 minutes.

The LCO is also modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. The required administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment;~~,~~ and 2) specific individuals are designated, trained and readily available to effect the closure of the penetrations;~~,~~ 3) the tools and equipment required to support the closure of the penetrations and the location of these tools and equipment relative to the penetrations are controlled;~~,~~ and 4) any potential obstruction (e.g., cables, hoses, etc.) that could prevent rapid closure of the penetrations can be quickly removed to support immediate initiation of closure and completion of closure within approximately 30 minutes.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates by inspection or administrative means that each of the containment penetrations is closed or capable of being ~~required to be in its~~ closed ~~position is in that position~~. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment that exceeds acceptable limits.

SR 3.9.4.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 24 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.6, the Containment Purge and Exhaust Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a CFT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 24 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

REFERENCES

1. Design Criteria Memorandum T-16, Containment Functions.
2. FSAR , Section 15.4.5.
3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.
4. ICRP Publication 30, 1979.
5. RG 1.25

REVISED TECHNICAL SPECIFICATION PAGES

1.1 Definitions (continued)

DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of NRC Regulatory Guide 1.109, Rev. 1, October, 1977, or those listed in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers", 1979.
$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

(continued)

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch capable of being closed and held in place by four bolts;

b. One door in each air lock capable of being closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

   1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

   2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation valve.

------------------------------------ NOTE ------------------------------------

Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS. <u>AND</u> A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	Verify each required containment purge and exhaust ventilation isolation valves actuates to the isolation position on an actual or simulated actuation signal.	24 months