February 22, 1995

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Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
IES Utilities Inc.
Post Office Box 351
Cedar Rapids, Iowa 52406

SUBJECT: AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-49 - DUANE ARNOLD ENERGY CENTER (TAC NO. M90155)

Dear Mr. Liu:

The Commission has issued the enclosed Amendment No. 207 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated August 15, 1994, as supplemented on December 21, 1994, and January 20, 1995.

The amendment revises the TS by increasing the allowable main steam isolation valve (MSIV) leakage and deleting the TS requirements applicable to the MSIV leakage control system (LCS). MSIV leakage will be directed to the main steam drain lines and the main condenser instead of the LCS.

In the near future, the staff intends to perform a confirmatory audit to examine the soil-structure interaction analysis of the turbine building including the validation of the version of the computer code (CLASSI) that was used by EQE (licensee's contractor) in the analysis.

A copy of the Safety Evaluation is enclosed. Notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by Glenn B. Kelly
Glenn B. Kelly, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Encls: 1. Amendment No. 207 to License No. DPR-49
2. Safety Evaluation

*See Previous Concurrence

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 22, 1995

Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
IES Utilities Inc.
Post Office Box 351
Cedar Rapids, Iowa 52406

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Sincerely,

Glenn B. Kelly

Glenn B. Kelly, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures: 1. Amendment No. 207 to License No. DPR-49
2. Safety Evaluation

cc w/encls:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

IES UTILITIES INC.

CENTRAL IOWA POWER COOPERATIVE

CORN BELT POWER COOPERATIVE

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 207
License No. DPR-49

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by IES Utilities Inc., et al., dated August 15, 1994, as supplemented on December 21, 1994, and January 20, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 207 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Leif J. Norrholm, Project Director
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of issuance: February 22, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 207

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove	Insert
iii	iii
3.7-3	3.7-3
3.7-4	3.7-4
----	3.7-4a
3.7-12	3.7-12
3.7-29	3.7-29
3.7-30	3.7-30

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

1) Test Pressure

All Type B tests shall be performed by local pneumatic pressurization of the containment penetrations, either individually or in groups, at a pressure not less than Pa.

2) Acceptance Criteria

The combined leakage rate of all penetrations subject to Type B and C tests shall be less than 0.60 La.

c. Type C Tests

1) Type C tests shall be performed on containment isolation valves. Each valve to be tested shall be closed by normal operation and without any preliminary exercising or adjustments.

2) Acceptance criteria - The combined leakage rate for all penetrations subject to Type B and C tests shall be less than 0.60 La.

3) The leakage from any one main steam isolation valve shall not exceed 100 scf/hr at a test pressure of 24 psig.* The combined maximum pathway leakage rate for all four main steam lines shall not exceed 200 scf/hr at a test pressure of 24 psig.

4) The leakage rate from any containment isolation valve whose seating surface remains water covered post-LOCA, and which is hydrostatically Type C tested, shall be included in the Type C test total.

* If a main steam isolation valve exceeds 100 scf/hr, it will be restored to ≤ 11.5 scf/hr.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

d. Periodic Retest Schedule

1) Type A Test

After the preoperational leakage rate tests, a set of three Type A tests shall be performed, at approximately equal intervals during each 10-year service period. (These intervals may be extended up to eight months if necessary to coincide with refueling outages.) The third test of each set shall be conducted when the plant is shut down for the 10-year plant in-service inspections.

The performance of Type A tests shall be limited to periods when the plant facility is nonoperational and secured in the shutdown condition under administrative control and in accordance with the plant safety procedures.

2) Type B Tests

a) Penetrations and seals of this type (except air locks) shall be leak tested at greater than or equal to 43 psig (P_a) during each reactor shutdown for major refueling or other convenient interval but in no case at intervals greater than two years.

b) The personnel airlock shall be pressurized to greater than or equal to 43 psig (P_a) and leak tested at least once every six (6) months. This test interval may be extended to the next refueling outage (up to a maximum interval between P_a tests of 24 months) provided there have been no airlock openings since the last successful test at P_a.

c) Within three (3) days after securing the airlock when containment integrity is required, the airlock gaskets shall be leak tested at a pressure of P_a.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3) Type C Tests

Type C tests shall be performed during each reactor shutdown for major refueling or other convenient interval but in no case at intervals greater than two years.

4) Additional Periodic Tests

Additional purge system isolation valve leakage integrity testing shall be performed at least once every three months in order to detect excessive leakage of the purge isolation valve resilient seats. The purge system isolation valves will be tested in three groups, by penetration: drywell purge exhaust group (CV-4302 and CV-4303), torus purge exhaust group (CV-4300 and CV-4301), and drywell/torus purge supply group (CV-4307, CV-4308 and CV-4306).

LIMITING CONDITIONS FOR OPERATION

F. Deleted

SURVEILLANCE REQUIREMENTS

F. Deleted

72 hours is allowed to restore the vacuum breaker to OPERABLE status. The 72-hour Completion Time takes into account the redundant capability afforded by the remaining breakers, reasonable time for the repairs, and the low probability of an event occurring during this period requiring the vacuum breakers to function.

An open vacuum breaker allows communication between the drywell and suppression chamber airspace, and, as a result, there is the potential for suppression chamber overpressurization due to this bypass leakage if a LOCA were to occur. Therefore, the open vacuum breaker must be closed. The 2-hour Completion Time is based on the time required to complete the alternate method of verifying that the vacuum breakers are closed, and the low probability of a DBA occurring during this period.

3.7.F and 4.7.F Bases: Deleted

3.7.G and 4.7.G BASES

Suppression Pool Level and Temperature

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1040 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum allowable pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 43 psig which is below the design pressure of 56 psig. The maximum volume of 61,500 ft^3 (equivalent to an indicated level of 60%) ensures the clearing loads from SRV discharges are not excessive and do not result in excessive pool swell loads during a Design Bases LOCA. The minimum volume of 58,900 (equivalent to an indicated level of 40%) ft^3 results in a submergence of approximately 3 feet. Based on Humboldt Bay, Bodega Bay, and Marviken test facility data as utilized in General Electric Company document number NEDE-21885-P and data presented in Nutech document, IES Utilities Inc. document number 7884-M325-002, the following technical assessment results were arrived at:

1. Condensation effectiveness of the suppression pool can be maintained for both short and long term phases of the Design Basis

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-49

IES UTILITIES INC.
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By letter dated August 15, 1994, and as supplemented by letters dated December 21, 1994, and January 20, 1995, IES Utilities Inc. (IES or the licensee), proposed a license amendment to change the Technical Specifications (TS) for the Duane Arnold Energy Center (DAEC or the facility), Facility Operating License No. DPR-49. The proposed changes would increase the allowable leakage rate specified in TS 4.7.A.2.c.3 from the current 11.5 standard cubic feet per hour (scfh) for any one main steam isolation valve (MSIV) when tested at 24.0 psig to 100 scfh for any one MSIV with a total maximum allowable pathway leakage of 200 scfh through all four main steam lines when tested at 24.0 psig. The proposed changes would also delete TS 3.7.E and 4.7.E for the MSIV Leakage Control System and the associated Bases section.

Specifically, the licensee requested that:

1. Allowable leakage rate specified in TS 4.7.A.2.c.3 be modified from the current 11.5 scfh for any one MSIV when tested at 24.0 psig to 100 scfh for any one MSIV with a total maximum pathway leakage of 200 scfh through all four main steam lines when tested at 24.0 psig;

2. TS 3.7.E and 4.7.E and their Bases be deleted to permit the disabling of the MSIV Leakage Control System (LCS) and to delete its requirements from the Technical Specifications. The licensee proposes these changes as an alternative to Regulatory Guide (RG) 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," by utilizing the main steam lines and condenser as an alternate method for MSIV leakage treatment.

3. The Index be administratively revised to reflect the above requested changes.

The proposed changes are a result of extensive work performed by the Boiling Water Reactor Owners Group (BWROG) in support of the resolution of Generic Issue C-8, "MSIV Leakage and Leakage Failure." In addition to the licensee's submittals, Generic Electric (GE) report NEDC-31858P, Revision 2, "BWROG Report for Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems," dated September 1993 (Reference 8), also provided technical justification for the proposed changes. Although the BWROG report has not been approved by the staff, the staff relied upon portions of the earthquake experience data, piping data, and main condenser data in the report in preparing this safety evaluation.

Because the original design basis at DAEC of certain main steam piping and components is not Seismic Category I, the licensee has performed evaluations and seismic verification walkdowns to demonstrate that the main steam system piping and components that comprise the alternate leakage treatment system are seismically rugged and are able to perform the safety function of MSIV leakage treatment following a safe shutdown earthquake (SSE). It should be noted that there are no provisions in the DAEC Final Safety Analysis Report (FSAR) and the staff's safety evaluation associated with the facility operating license that would permit the use of experience data as a means of seismic qualification for piping systems and components. However, requiring the non-seismically analyzed portions of the main steam system piping and components to meet Seismic Category I requirements would not be practical because modifications required to upgrade the system to Seismic Category I requirements cannot be justified from the cost-benefit standpoint.

The staff determined, during its review, that additional information would be required from the licensee in order to demonstrate that the proposed alternate leakage treatment system has adequate seismic capability, that the radiological consequences of a postulated loss of coolant accident (LOCA) will be within the dose reference values set forth in 10 CFR Part 100, and that the control room operator dose limits specified in GDC-19 of Appendix A to 10 CFR Part 50 will not be exceeded. This additional information was provided by submittals dated December 21, 1994, and January 20, 1995 (References 10 and 13).

2.0 EVALUATION

The main steam lines (MSLs) contain dual quick-closing MSIVs. These valves function to isolate the reactor coolant system in the event of a break in a steam line outside the primary containment, a design basis LOCA, or other events requiring containment isolation. Although the MSIVs are designed to provide a leak-tight barrier, it is recognized that some leakage through the valves will occur. Operating experience at various boiling water reactor (BWR) plants has indicated that degradation occasionally has occurred in the leak-tightness of MSIVs, and the specified low leakage has not always been maintained.

Because of recurring problems with excessive leakage of MSIVs, Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," recommended the installation of a supplemental leakage control system (LCS) to ensure that the isolation function of the MSIVs complies with the specified limits. To meet this requirement, the licensee installed a safety-related MSIV LCS that is designed to eliminate the release of fission products. This is accomplished by developing a negative pressure in the sections of the MSLs between the inboard and outboard MSIVs. This negative pressure is developed by a series of blowers that discharge the leakage to the standby gas treatment system (SGTS).

Due to design limitations, the LCS would be unavailable if the MSIV leak rate were greatly in excess of the allowable value in the Technical Specifications. Hence, Generic Issue C-8 was initiated in 1983 to assess (1) the causes of MSIV failures, (2) the effectiveness of the LCS and alternative leakage paths, and (3) the need for regulatory action to limit public risk. The resolution of C-8 (see NUREG-1372, Regulatory Analysis for the Resolution of Generic Issue C-8, "Main Steam Isolation Valve Leakage and LCS Failure" dated June 1990) concluded that no backfit requirements to reduce public risk were warranted and that no action should be taken. However, one of the alternative resolutions of C-8 showed that several non-Seismic Category I paths gave off lower doses than the LCS and could handle larger MSIV leak rates.

In a parallel effort, the BWROG formed an MSIV Leakage Committee in 1982 to identify and resolve the causes of high MSIV leakage rates. The BWROG then formed a follow-on MSIV Leakage Closure Committee to address alternate actions to resolve on-going, but less severe, MSIV leakage problems and to address the limited capability of the LCS. The results of these committee activities were submitted to the NRC in several General Electric (GE) proprietary reports, the latest of which is NEDC-31858P, Revision 2 (September 1993), titled, "Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems." This report concludes that the proposed increase of the MSIV leakage limit will reduce radiation exposures to maintenance personnel, reduce outage durations, and extend the effective service life of the MSIVs. The report also concludes that the proposed elimination of the LCS will similarly reduce exposures to maintenance personnel, reduce outage durations, and that the LCS can be replaced with an alternate method for MSIV leakage treatment using the MSLs and condenser. The licensee referred to this report as a basis to delete the TS requirements for the MSIV LCS and requested a substantially higher (100 scfh per MSIV and a total of 200 scfh for all four MSLs) MSIV leak rate limit.

The proposed alternate treatment method recommended in the BWROG report, and as proposed by the licensee, takes advantage of the large volume in the main steam lines and main condenser to provide hold-up and plate-out of fission products that may leak from closed MSIVs. This method uses the main steam drain lines to direct leakage to the main condenser. In this approach, the main steam piping, the bypass/drain piping, and the main condenser are used to mitigate the consequences of an accident, which could result in potential offsite exposures comparable to 10 CFR Part 100. The licensee also is required to demonstrate that the main steam system piping and components that comprise the alternate leakage treatment system are seismically rugged and are able to perform the safety function of MSIV leakage treatment following a safe shutdown earthquake (SSE). The BWROG report and the licensee's submittals provide the technical justification for the seismic capability of the alternate treatment path and also provide the dose calculations to demonstrate the acceptability of the system.

This evaluation has been performed in several parts. Section 2.1 provides the radiological assessment; Section 2.2 provides the seismic assessment; and Section 2.3 provides the drain path functional design assessment.

2.1 RADIOLOGICAL ASSESSMENT

In order to demonstrate the adequacy of the DAEC engineered safety features designed to mitigate the radiological consequences of design basis accidents (DBAs) with a maximum MSIV leak rate of 200 scfh total from four main steam lines and without the MSIV LCS, the licensee assessed the offsite and control room radiological consequences that could result from the occurrence of a postulated LOCA and presented the results of the offsite dose calculations in their submittal.

In 1975, the staff assessed the offsite radiological consequences of a LOCA using a 45 scfh MSIV total leak rate from four main steam lines through the MSIV LCS using revised atmospheric relative concentration values (χ/Qs). In that assessment, the staff considered containment leakage and main steam isolation valve leakage as the sources and radioactivity transport paths to the environment following a postulated LOCA.

In this evaluation, the staff recalculated the radiological consequences resulting from only the main steam isolation valve leakage pathway due to an increase in MSIV leak rate and deletion of the main steam LCS. The staff used the radiological consequences calculated from the previous analysis for the containment leakage pathway. The procedures used in the staff's recalculation of offsite and control room radiological consequences were based on the current TID-14844 source term, which are consistent with the guidelines provided in the applicable Standard Review Plan (SRP, NUREG-0800) and Regulatory Guides, except the following two deviations. The staff has provided a credit for radioactive iodine removal in the main steam lines and main condenser by hold-up for decay and deposition, and has accepted deletion of the TS requirements for the MSIV Leakage Control System.

The Duane Arnold Energy Center (DAEC) is designed and constructed with a Mark I BWR containment, and therefore, the staff has provided a suppression pool decontamination factor of 5 in accordance with SRP Section 6.5.5 (issued subsequent to the issuance of the DAEC operating license) in its radiological consequence assessments. The staff used in its assessment, the offsite χ/Q values documented by the licensee in their submittal letter dated January 20, 1995 (Reference 13). The staff's recalculated offsite doses resulting from a postulated LOCA and the parameters and assumptions used in the staff's recalculation are given respectively in Tables 1 and 2 of this evaluation. The staff's recalculated control room operator doses are given in Table 3.

In response to the MSIV leakage concerns, the BWR Owners Group (BWROG) in 1986 commissioned studies to determine the causes of high leak rates and the means to eliminate them. The results of these studies were submitted to the NRC in the General Electric proprietary reports, NEDO-31643P (November 1988), NEDO-31858P Revision 0 (February 1991), NEDO-31858P Revision 1 (October 1991), and NEDO-31858P Revision 2 (September 1993), all titled, "Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems."

The current assumption used by the staff for operating plants in calculating radiological consequences of potential DBAs is based upon a conservative assumption that the leakage limit allowed by the TS is released directly into the environment. No credit currently is taken for the integrity and leaktightness of the main steam piping and condenser to provide holdup and plateout of fission products. The proposal developed by the BWROG and adopted by the licensee would allow higher leakage limits (200 scfh total from four steam lines) and delete the TS requirements for the main steam LCS.

2.1.1 <u>Iodine Release Pathways</u>

Following a LOCA, three potential release pathways exist for main steam leakage through the MSIVs:

(1) Main steam drain lines to the condenser, with delayed release to the environment through the low pressure turbine seals.

(2) Turbine bypass lines to the condenser, with delayed release to the environment through the low pressure turbine seals.

(3) Main steam lines through the turbine stop and control valves, and through high pressure turbine seals to the environment, bypassing the condenser.

The consequences of leakage from pathways 1 and 2 will be essentially the same because the condenser can be used to process MSIV leakage. The condenser iodine removal efficiency will vary depending on the inlet location of the bypass or drainline piping, but in either case, iodine will be removed. For pathway 3, MSIV leakage through the closed turbine stop and control valves will not be processed via the condenser. For this case, the high-pressure turbine (having a large internal surface area associated with the turbine blades) will remove iodine.

The staff believes that as long as either the turbine bypass or drainline leakage pathway is available, MSIV leakage through the closed turbine stop and control valves (pathway 3) will be negligible. Essentially, all of the releases will be through the main condenser, because there will be no differential pressure in the MSL downstream of the MSIVs following the closure of the valves. Further, MSIV leakage through pathway 3, if any, will have been subjected to the same iodine removal processes in the MSLs (up to turbine stop valves) as the other pathways. The leakage will be further subjected to iodine removal by deposition in the high-pressure turbine internal surfaces. Removal by the main condenser is not applicable in pathway 3.

The licensee has selected to utilize pathway 1 (using the main steam piping and its drain piping, and the condenser) as the primary pathway to mitigate the radiological consequences of an accident that could result in potential offsite exposures comparable to the dose reference values specified in 10 CFR Part 100. The staff has accepted the licensee's proposed pathway. In the calculation of the contribution to the LOCA dose, the staff assumed that one of the inboard isolation MSIVs failed to close, thus allowing contaminated steam to travel to the outboard valve. This outboard valve and the outboard valves from the other three steamlines were assumed to have a total leak rate of 200 scfh.

2.1.2 Iodine Transport Model

The transport of gaseous iodine in elemental and particulate forms has been studied for many years, and several groups proposed different models to describe the observed phenomena (References 1 through 4). In this evaluation, the staff used the model specifically developed by an NRC contractor (Reference 5) for iodine removal in BWR main steam lines and the main condenser following a LOCA.

The staff model treats the MSIV leakage pathway as a sequence of small segments for which instantaneous and homogeneous mixing is assumed; the mixing computed for each segment is passed along as input to the next segment. The number of segments depends upon the parameters of the line and flow rate, and can be as many as 100,000 for a long, large-diameter pipe and a low flow. Each line segment is divided into five compartments that represent the concentrations of the three airborne iodine species, the surface that contains iodine available for resuspension, and surface iodine that has reacted and is fixed on the surface.

The staff's model considers three iodine species: elemental, particulate, and organic. A fourth species, hypoiodous acid, was considered for the purpose of the staff's model to be a form of elemental iodine. All iodine in the segment undergoes radioactive decay. The resulting concentration from each segment of the deposition compartment serves as the input to the next segment.

The GE model, as well as the one developed and used by the staff, is based on time-dependent temperature adsorption phenomena with instantaneous and perfect mixing in a given volume. Both models use the same MSIV leakage pathways. They differ however, in the treatment of buildup of iodine in the main steamlines and condenser. GE assumed steady state iodine in equilibrium in a large volume, while the staff model assumed transient buildup of iodine in a finite number of small volumes. The staff does not consider these differences to be significant since the staff finds that the resulting iodine deposition and removal rates in the main steam lines and condenser are in good agreement.

The staff's transport model also assumed iodine transport through the condenser as a dilution flow rather than the plug flow as in the steam lines. The staff assumed that the iodine input into the condenser mixes instantaneously with a volume of air in the condenser and that the diluted air exhausts at the same time and same rate as the input air (MSIV leakage) flows into the condenser.

The staff developed the equations for iodine deposition velocities, resuspension rates, and surface fixation rates as a function of temperature using published data found in the literature. The equations and data are contained in the contractor's report (Reference 5). The equation for the deposition velocity of elemental iodine is based on the least-squares fit to the available data. Deposition velocity equations for HOI and organic iodine are based on the values at 30 °C; due to the lack of data at elevated temperatures, their temperature dependence is assumed to be similar to elemental iodine. Resuspension and fixation equations as a function of temperature are based on measurements available in the literature at ambient temperature. The staff assumed that resuspension and fixation rates will increase with increasing temperature.

The technical references, and the GE and staff models indicate that particulate and elemental iodine would be expected to deposit on surfaces with rates of deposition varying with temperature, pressure, gas composition, surface material, and particulate size. Therefore, the staff believes that an appropriate credit for the removal of iodine in the MSLs and main condensers should be provided in the radiological consequence assessment following a design-basis accident.

Sections III(c) and VI of Appendix A to 10 CFR Part 100 require that structures, systems, and components necessary to ensure the capability to mitigate the radiological consequences of accidents that could result in exposures comparable to the dose guidelines of Part 100 be designed to remain functional during and after a safe-shutdown earthquake (SSE). Thus, the MSL, portions of its associated piping, and the main condenser are required to remain functional if credit is taken for deposition of iodine and if the SSE occurs.

For the purpose of providing a credit for iodine holdup and plateout, the staff's model requires that the main steam piping (including its associated piping to the condenser) and the condenser remain structurally intact following an SSE, so they can act as a hold-up volume for fission products. By the term "structurally intact," the staff assumes the steam line will retain sufficient structural integrity to transport the relatively low flow rate (≤ 2 ft^3/min) of MSIV bypass leakage throughout the steam lines and condenser. The staff considers, in its radiological consequence assessment, that the condenser is open to the atmosphere via leakage through the low pressure turbine seals. Thus, it is only necessary to ensure that gross structural failure of the condenser will not occur during an SSE. The seismic capability of the alternate pathways is discussed in Section 2.2 of this safety evaluation (SE).

2.1.3 Control Room Habitability

The DAEC control room is located in the control building, which is adjacent to, but physically separated from the reactor and turbine buildings. When a predetermined level of airborne radioactivity is detected at the normal outside air intake, the control room makeup air is diverted to the emergency makeup air filtration system. The system is designed to maintain the control room at a positive pressure related to adjacent areas. The pressurization is accomplished by introducing 1000 cfm of outside air. The filtration unit is an engineered safety feature system and is redundant. Both trains contain, among other things, a charcoal adsorber and HEPA filters.

The staff has evaluated the control room operator doses following a postulated LOCA in accordance with SRP Section 6.4 and found the calculated doses were within the guidelines of SRP Section 6.4. In the evaluation, the staff considered the fission product releases from the low pressure turbine seal due to the MSIV leakage (up to 200 scfh total) through the MSIV drain lines and the main condensers. The staff assumed a ground level release of airborne fission products from the turbine building as a fission product diffusion source and the control room emergency air intake as a single point receptor. The staff used control room χ/Q values provided by the licensee in their submittal letter dated January 20, 1995 (Reference 13).

The staff's recalculated control room operator doses following a postulated LOCA are listed in Table 3. The staff finds that the recalculated whole-body and equivalent organ doses (thyroid) are within the guidelines of SRP Section 6.4.

2.1.4 Conclusion

Several technical references (Reference 1 - 5), including an NRC contractor's report (Reference 6), indicate that particulate and elemental iodine would be expected to deposit on surfaces with rates of deposition varying with temperature, pressure, gas composition, surface material, and particulate size. The staff, therefore, concludes that an appropriate credit for the removal of iodine in the main steam lines and main condensers should be provided in the radiological consequence assessment following a DBA.

The staff has reviewed the licensee's analysis and has performed an independent reassessment of the radiological consequences resulting from the MSIV leakage transport pathway described in this SE. The calculated thyroid and whole-body dose are listed in the revised Table 1. Based on the above evaluation and the calculated radiological consequences shown in Table 1, the staff concludes that the MSIV leak rate limit of 200 scfh total from four main steam lines and the proposed deletion of the TS requirements for the MSIV Leakage Control System are acceptable.

The staff further concludes that the existing distances to the exclusion area and to the low population zone boundaries of the DAEC site, in conjunction with the remaining engineered safety features provided in the DAEC, are still sufficient to provide reasonable assurance that the radiological consequences of a postulated LOCA will be within the dose reference values set forth in 10 CFR Part 100 and the control room operator dose limits specified in GDC-19 of Appendix A to 10 CFR Part 50.

2.2 SEISMIC ASSESSMENT

The BWROG retained Earthquake Engineering, Inc. (EQE), as a consultant to conduct a study of the earthquake experience data on the performance of facility piping and condensers. The study summarized the data on the performance of main steam system piping and condensers in non-nuclear applications that experienced strong motion earthquakes. In addition, it compared these piping systems and condensers with the piping systems and condensers typically used in GE BWRs in the United States. The result of the comparison appears to support the BWROG's contention that main steam piping and condensers employed in GE BWRs would maintain pressure boundary integrity during a design basis earthquake (SSE). According to EQE, based on past earthquake experience, welded steel piping and condensers designed and constructed to normal industrial practices (e.g., ANSI B31.1 and Heat Exchange Institute (HEI) Standard, respectively) have been found to be seismically rugged and not susceptible to a primary collapse mode of failure, as a result of the seismic vibratory motions experienced at sites examined in the earthquake database. The report notes that a relatively small number of seismically induced piping failures have occurred due to excessive relative support movements or seismic interactions.

The primary components to be relied upon for the proposed alternate leakage treatment system are the main turbine condenser, a primary drain path, and an alternate drain path. The proposed primary drain path at DAEC employs a main steam line (MSL) drain downstream of the MSIVs, through Path 3"-EBD-3, to the condenser. There are two normally closed motor-operated valves (MOVs), MO1043 and MO1044, in series in this line between the MSLs and the main condenser. Both MOVs are powered by essential power to ensure that they can be opened to accommodate the required drain path following the design basis LOCA and to support the radiological analysis.

An alternate drain path will be available to convey MSIV leakage to the isolated condenser, if either of the MOVs fails to open. The alternate drain path consists of the bypass lines, through Path 2"-EBD-2, around the MOVs in the primary drain path. This alternate drain path contains a "fail open" valve, CV-1064, which is normally closed, and a restricting orifice. Consequently, if either of the MOVs on the primary drain path fails to open as required, the alternate drain path would be available to convey MSIV leakage to the main condenser. Radiological dose calculations have been performed for this alternate drain path, as well as for the primary drain path (See Section 2.1 of this SE).

The condenser forms the ultimate boundary of the main steam drain paths. Boundaries upstream of the condenser were established by utilizing existing valves, i.e., valves V03-0004, V03-0005, MO1054, MO1055, MO1169, MO1170, as well as Main Steam Stop & Control Valves and Bypass Control Valves. They are used to define the extent of the seismic verification walkdowns.

2.2.1 Earthquake Experience Database

This section of the SE assesses the information provided by the licensee on earthquake data to ensure that the vibratory ground motion, experienced at each of the facilities with equipment being used as a surrogate for that at DAEC, did indeed exceed the DAEC SSE. IES Utilities has indicated (Reference 8, Reference 10) that only about 25 of the ground motion estimates in the data base are from actual instrument recordings at or near the cited facility site. For the other facility-earthquake pairs in the database, the ground motion estimates were extrapolated from instruments located at some distance from the facilities or are made by speculation based on nearby damage or other arguments.

The licensee submittal indicated that the DAEC condenser design is typical of those at the El Centro Steam Plant, which experienced the Imperial Valley 1979 earthquake; the Valley Steam Plant, which experienced the San Fernando 1971 earthquake; and the Moss Landing Steam Plant, which experienced the Loma Prieta 1989 earthquake. The submittal also indicated that the DAEC main drain and associated piping, and interconnected systems are similar to the piping found in commercial piping systems at the PALCO Co-generation Plant, which experienced the Cape Mendocino earthquakes of 1992; the Southern California Edison Cool Water Station, which experienced the Landers and Big Bear 1992 earthquakes; the El Centro Steam Plant, which experienced the Imperial Valley 1979 earthquake; the Valley Steam Plant, which experienced the San Fernando 1971 earthquake; and the Bulk Mail Facility, Lutheran Towers, and California Federal Bank Facility, all of which experienced the Whittier 1987 earthquake. In order to evaluate the appropriateness of the ground motion estimates made for each of the above facilities, the staff transmitted a request for additional information (RAI) (Reference 9) to the licensee on the database ground motions provided. The RAI requested specific information on how the ground motion values used in the experience database were obtained including the following:

a. the name, date, time, and magnitude of the earthquake;

b. the name, location, foundation material, and distance to the earthquake epicenter of the facility being used for the database;

c. the distance to the seismic instruments from the facility on which the ground motion at the facility is based, and the foundation material for the instruments;

d. the level of ground motion recorded by the instruments;

e. the level of ground motion estimated for the facility, and

f. a description of the method used to estimate the ground motion at the facility

The licensee's reply (Reference 10) to the RAI provided most of the information requested, which allowed the staff to make independent analyses of the suitability of the ground motion for all of the facilities listed above except for Lutheran Towers, the California Federal Bank Facility, and the Bulk Mail Facility. Insufficient information was provided about these three sites to allow the staff to make its independent assessment regarding estimates involving these sites.

The EQE, Inc. ground motion estimate at the El Centro Steam Plant from the Imperial Valley 1979 earthquake was based on a recording made at a U. S. Geological Survey (USGS) strong ground motion station about 1 kilometer from the facility. Because of the density of seismic recordings in that area and the distribution of the ground motion, the staff concluded that the estimate for the site is significantly larger than the DAEC SSE and is therefore appropriate for use in establishing the seismic capacity of the DAEC equipment similar to those in the El Centro Steam Plant.

The EQE, Inc. ground motion estimate at the Valley Steam Plant from the San Fernando 1971 earthquake was based on an extrapolation of data from a relatively distant location. In 1988, the USGS performed studies to estimate the ground motion at selected sites from the San Fernando 1971 earthquake in support of the NRC's resolution of the USI A-46 program (Reference 11). Figure 1 is a plot of the DAEC SSE response spectrum, and the EQE, Inc. and USGS estimates of the Valley Steam Plant response spectrum from San Fernando 1971. The accelerations in the USGS spectrum, while lower than the EQE, Inc. spectrum, are significantly higher than the corresponding values from the DAEC spectrum. As was concluded in the resolution of A-46 ground motion issue, the staff considers the USGS estimate to be the characterization of the ground motion at the Valley Steam Plant from the San Fernando 1971 earthquake and is therefore, appropriate for use in establishing the seismic capacity of the DAEC equipment similar to that in the Valley Steam Plant.

The EQE, Inc. ground motion estimate at the Moss Landing Steam Plant from the Loma Prieta 1989 earthquake is based on a study performed by Pacific Gas and Electric Company (PG&E) (Reference 12), the owner of the Moss Landing Steam Plant. Since the staff concluded that the analysis performed by PG&E was technically sound and comprehensive, it determined that PG&E's estimate of the ground motion is appropriate for use in establishing the seismic capacity of the DAEC equipment similar to that in the Moss Landing Steam Plant.

The ground motion estimate at the PALCO Co-generation Plant from the Cape Mendocino magnitude 7 earthquake of 1992 is based on a recording at a California Department of Mines and Geology station in Rio Dell at some distance from the facility. The NRC staff, using two ground motion estimation formulas which are based on a statistical analysis of a large set of empirical data, made its own estimate of the ground motion at the PALCO Co-generation Plant from the Cape Mendocino 1992 earthquake. Figure 2 contains a plot of the DAEC SSE response spectrum, the EQE, Inc. response spectrum, and the two NRC estimates of the PALCO Co-generation Plant response spectrum from the Cape Mendocino 1992 earthquake. The NRC spectra, while lower than the EQE, Inc. spectrum, are both higher than the DAEC SSE spectrum. The staff considers the lower bound envelope of the NRC estimates to be the appropriate characterization of the ground motion at the PALCO Co-generation Plant from the Cape Mendocino 1992 earthquake, and are therefore, acceptable for use in establishing the seismic capacity of the DAEC equipment similar to that in the PALCO Co-generation Plant.

IES Utilities stated (Reference 10) that the Southern California Edison Cool Water Station ground motion from the Landers and Big Bear 1992 earthquakes was recorded on site. Therefore, the staff considers this ground motion to be appropriate for use in establishing the seismic capacity of the DAEC equipment similar to the equipment in the Southern California Edison Cool Water Station.

As indicated earlier, IES Utilities did not provide sufficient information about the Lutheran Towers, the California Federal Bank Facility, and the Bulk Mail Facility data for the staff to make an assessment of the ground motion estimates. The staff was also unable to obtain relevant information relating to these three sites. However, since none of the seismic capacity estimates of equipment in the alternate leakage treatment system at DAEC relies solely on the estimates for these sites for its seismic capacity determination, the staff has concluded that the lack of the necessary information is not particularly significant to its assessment of the overall earthquake experience data provided by the licensee.

Based on the independent analysis of the earthquake experience database, the staff has concluded that DAEC SSE demand is well-below the seismic ground motion that was experienced at the facilities discussed above. Consequently, the use of the database is acceptable for establishing the seismic adequacy of equipment in the alternate leakage treatment system at DAEC.

2.2.2 Turbine Building Soil-Structure Interaction and Seismic Reanalysis

In connection with its request for a Technical Specification (TS) amendment to eliminate the Main Steam Isolation Valve (MSIV) Leakage Control system, the licensee provided an evaluation of the seismic adequacy of the turbine building (TB). The licensee stated that although the building is classified in the DAEC FSAR as nonseismic, the criteria for Seismic Category I structures were employed for its structural design using the original seismic analysis performed for the building. The licensee concluded, therefore, that the building was capable of withstanding the design basis seismic SSE loading. The staff concurs with the licensee's determination.

The licensee also performed a seismic reanalysis of the turbine building for the SSE, considering the effects of the soil-structure interaction (SSI). The reanalysis was performed to support the determination of the seismic ruggedness of the building and to generate in-structure response spectra for the evaluation of the alternate leakage treatment system piping and supports located in the turbine building. Following a meeting with the licensee on October 17, 1994, the NRC staff requested additional information regarding the SSI analysis of the turbine building (Reference 9). By letter dated December 21, 1994 (Reference 10), the licensee provided the requested additional information.

As described in Reference 10, the DAEC turbine building (TB) is a three-story rectangular structure which is 260 feet (ft) long, 140 ft wide, and 107 ft high. The foundation basemat is founded on about 15 ft of undisturbed glacial till over bedrock. The top of the basemat is at elevation (El) 734 ft. The sides of the TB were backfilled with compacted soil up to grade level at El 757 ft. The basement of the structure consists of a reinforced concrete basemat with reinforced concrete walls. The remainder of the structure consists of braced steel frame walls and roof with reinforced concrete floor slabs. The reinforced concrete turbine pedestal is centrally located and extends from the basemat to the operating floor at El 780 ft.

The original seismic analysis of the TB was performed by John A. Blume & Associates in the early 1970s using a very simplified SSI model shown in schematic form in Fig 1-3 of Reference 10. According to the information provided by the licensee's consultant, EQE, the original analysis considered the effects of the single overlying glacial till layer by a horizontal translational soil-spring and a rotational soil-spring that are capable of transmitting horizontal seismic excitation only. Effects of vertical excitation were not considered in the original analysis.

EQE performed a seismic reanalysis of the TB and generated new in-structure response spectra to obtain a more realistic estimate of the seismic response of the structure. For this reanalysis, EQE performed the SSI analysis using the Continuum Linear Analysis for Soil-Structure Interaction (CLASSI) code based on an elastic half-space model. This code accounts for the effects of radiation damping in the foundation soil (which reduces the structural response) using the current state-of-the-art knowledge of wave propagation in soil medium, whereas the original analysis performed in the 1970s could not account for such effects. Recently, the NRC staff has reviewed, and accepted, the validation of other versions of this code in connection with seismic analysis of several nuclear power plants. In its reanalysis, using the CLASSI code, EQE utilized the same structural (stick) model and soil properties that were used in the original analysis. The structure was assumed to be surface founded, and the ground motion with a peak ground acceleration (PGA) of 0.12 g was specified at the surface of the half-space. The licensee confirmed during the teleconference call on January 6, 1995, that the surface of the half-space was assumed to be at El 734 ft, which coincides with the elevation of the top of the foundation basemat, and that the ground motion was input at El 734 ft, and not at the plant grade at El 757 ft. The specification of ground motion at the level of the foundation basemat is acceptable to the staff, since this involves no deconvolution of ground motion from the grade surface at El 757 ft.

According to the updated final safety analysis report (UFSAR) (Rev. 5), the horizontal PGA associated with the design basis earthquake (DBE) for structures founded on 30 to 50 ft of compacted fill or natural glacial soils is 0.18 g; whereas the FSAR specifies a PGA of 0.12 g for structures founded on rock and for those structures founded on about 10 ft of compacted fill and/or natural glacial soils overlying bedrock. For the TB which has a glacial soil layer of 15 ft above bedrock, the original seismic analysis used a PGA of 0.12 g, and EQE also used the same PGA value in the CLASSI analysis.

This is acceptable to the staff.

In its reanalysis, using an improved analytical technique as described earlier, EQE used the dynamic structural and soil properties based on the original seismic calculations performed by John A. Blume & Associates. EQE back-calculated a shear wave velocity of 860 ft/sec (fps) and a shear modulus of 2762 kips/square foot (ksf) from the stiffness values used for the soil springs in the original analysis. During the teleconference call on January 6, 1995, the staff learned from the licensee's consultant that the above shear modulus was used as a constant value corresponding to the maximum shear strain level. The licensee has further stated in Reference 10 that it did not consider any variations in the design basis soil properties in its reanalysis, since its objective was to generate the best estimate in-structure floor response spectra (FRS) instead of "conservative design" FRS. In view of the fact that the TB is not a Seismic Category I structure, the staff considers the above procedure of obtaining the soil properties reasonable for the limited purpose of performing the seismic reanalysis in connection with the licensee's request for the MSIV Leakage Control System Technical Specification Amendment Request.

In the seismic reanalysis, the licensee used three (two horizontal and one vertical) free-field acceleration time histories. The response spectra calculated from these acceleration time histories matched the 84th percentile non-exceedance NUREG/CR-0098 ground response spectrum for soil sites at 5 percent damping. The design ground motion spectrum was anchored to the design basis PGA of 0.12 g horizontal and 0.10 g vertical. The staff accepts this seismic input motion (which was applied at the basemat level as confirmed by the licensee during the teleconference call on January 6, 1995).

Using the above procedure, the licensee calculated the best estimate FRS for the TB using a structural damping ratio of 7 percent. Reference 10 gives the resulting FRS at various elevations obtained by the seismic reanalysis, along with the FRS from the original analysis. The comparison of the FRS from the CLASSI analysis with those from the original analysis given in Reference 10 indicates that the original analysis results were very conservative, probably because of the approximate analytical technique used in the 1970s, which did not account for the effects of radiation damping in the soil medium as stated earlier. The FRS from the current analysis show relatively large spectral amplifications (by a factor of about 6) of peak structural responses at El 834 ft (which is the elevation of the roof of the steel superstructure), and reasonable amplifications on the order of about 4 in the north-south direction and about 2 in the east-west direction at the top of the massive concrete pedestal at El 780 ft. However, the spectral amplifications of peak responses at El 757 ft (basement floor at plant grade level) and at El 780 ft (operating floor level at the top of first floor) are only slightly greater than a factor of 1.0 (about 1.2). The shape of the FRS at these locations resemble that of the ground response spectra that is applied at the basemat level.

A review of the structural data and the type of the structural responses of the various parts of the structure at different elevations described earlier indicates that the massive and extremely stiff concrete basement and first floor structure appear to behave like a rigid box that responds to ground motion as a single unit with the concrete basemat. This may explain the fact that the shape and amplifications of the FRS at the top of the basement and at the operating floor level are quite similar to the ground response spectrum, whereas the steel superstructure above El 780 ft and the pedestal show reasonable amplifications of peak responses as described earlier, because of their low stiffnesses compared to those of the basemat and the basement and the first floor structure. It is worth mentioning here that the effect of the soil-confinement provided to the turbine building by 23 ft of embedment below grade level has been ignored in the SSI model used in the analysis; this soil-confinement would act to reduce the structural response. The staff considers this approximation to be conservative and reasonable.

Based on the above staff evaluation of the licensee's submittal (Reference 10), and on the information provided by the licensee and its consultant, EQE, the staff concluded that the results of the seismic reanalysis performed by the licensee for the TB in connection with its request for MSIV Leakage Control System Technical Specification Amendment Request were acceptable. However, the staff's acceptance of the turbine building SSI analysis is limited to its application for this amendment request, and does not apply to other licensing issues at DAEC.

2.2.3 <u>Seismic Verification Walkdowns</u>

The alternate leakage treatment system consists of the main steam piping (beyond the outboard main steam isolation valves), the steam drain lines, the condenser, and interconnected piping. The system, in general, is not analyzed against Seismic Category I criteria, as this analysis was not required in the original licensing basis of DAEC.

In order to confirm the functional capability of the alternate leakage treatment system, the licensee performed seismic verification walkdowns for DAEC, in accordance with DAEC Walkdown Procedure, No. 42083-P-002, Revision 0, dated March 25, 1992. The purpose of the walkdowns was to ensure that the MSLs, the main steam drain lines, the condenser, and interconnected piping and equipment that were not seismically analyzed, fall within the bounds of the design characteristics of the seismic experience database as discussed in Section 6.7 of the BWROG Report (Reference 8). Specifically, the walkdowns were performed to (1) verify that DAEC plant features have attributes similar to those in the earthquake experience database that have demonstrated good seismic performance (e.g., piping flexibility, unique layout configurations, supports, and support configurations), (2) verify general conformance of pipe support spans to the requirements of ANSI B31.1, and (3) examine the alternate leakage treatment system to identify potential seismic vulnerabilities considering those structural details and causal factors that resulted in component damage at database plants.

The potential vulnerabilities which were typically classified as "outliers" fall within one of the following five types:

(1) potential deficiency in anchorage or support capacity;

(2) potential valve malfunction and collapse of the masonry walls which support the piping;

(3) potential damaging interaction between piping and nearby components;

(4) differential displacement of piping supports or attachments; and

(5) valves with extended motor operators beyond screening guidelines.

The licensee's August 15, 1994, submittal (Reference 7), presents a complete list of the "outliers" that were identified during the walkdowns and actions taken by DAEC for their resolution. These "outliers" have been evaluated by DAEC to demonstrate acceptability as-is or to identify the necessity to implement plant modifications to resolve the concerns. As a result of the walkdowns and the subsequent evaluations, DAEC summarized the following actions for resolution of identified "outliers":

(1) Add supports to a pipe section that exceeds pipe span screening criteria (main steam to 2nd stage reheater);

(2) Tighten anchor bolts, or relocate support and replace bolts, for pipe support with loose anchor bolts (main steam to 2nd stage reheater);

(3) Modify and reinstall pipe support EDB 3-H-44 (steam line drains in turbine building);

(4) Verify and modify (as needed) clearances of as-installed support, if support is not capable of accommodating differential building movement (main steam line branches);

(5) Reset overloaded spring support (main steam bypass to turbine steam seal);

(6) Remove damaged pipe support U-Bolt (main steam bypass to turbine steam seal);

(7) Modify fire protection piping by adding new supports (main steam bypass to turbine steam seal);

(8) Install the missing U-bolt for pipe support on adjacent air line (main steam instrumentation lines).

The licensee has committed to complete the above modifications or repairs prior to implementing the proposed TS change, to ensure that the damage reported for the database components would not occur to the DAEC main steam piping and condenser and the associated supports.

2.2.4 Additional Earthquake Performance Data

During a meeting on December 10, 1993, held at the NRC headquarters with Georgia Power Company (GPC), concerning a similar request for eliminating the

MSIV leakage control system at the Hatch Plant, EQE, acting as a consultant for GPC, presented the survey results for EQE data and open literature for 18 strong-motion earthquakes that covered 29 sites and 96 power plants. The 18 earthquakes range in Richter magnitude from 5.4 to 8.1. The EQE estimates of the average peak-ground accelerations (PGAs) from these earthquakes were in the range of 0.1 g to 0.85 g. The survey found no precedent for failure of the main steam piping pressure boundary and condenser shell. However, the survey did find damage to piping insulation, valve operators, piping supports, as well as condenser tubes. The EQE database covers facilities with underlying foundations varying from soft soils to rock. Also included were a substantial number of diverse structures and designs that house a wide variety of pipe runs, cable trays, conduits, tubing, and related components. The database also contained numerous records of equipment installations, from vintage 1930 to new.

The staff found the earthquake data provided in the BWROG report to be insufficient in covering pipes of smaller sizes, in the range of one-inch to 10 inches in diameter. During the December 10, 1993, meeting, the staff requested that GPC submit additional earthquake data to cover these smaller pipes. The supplemental and updated earthquake performance data, which included 24 earthquakes and about 126 sites, were subsequently provided in the GPC submittal of January 6, 1994. This same additional database was referenced in the DAEC's August 15, 1994, submittal. The measured or estimated horizontal ground accelerations for these updated database sites range from 0.15 g to 1.0 g, with the majority of the sites having estimates of peak-ground accelerations of 0.3 g or higher. The duration of strong motion (on the order of 0.10 g or greater) were estimated to range from five seconds to more than 50 seconds. The staff's evaluation of the earthquake data base ground motion estimates, provided by the licensee, is discussed in Section 2.2.1 of this SE. No other ground motion estimates have been evaluated by the staff for this review. The staff determined that the supplemental data on small piping expand the original piping database provided in the BWROG report, and envelop the DAEC alternate leakage treatment piping.

2.2.5 Alternate Leakage Treatment System

As indicated in the August 15, 1994, submittal (Reference 7), portions of the main steam piping, including the main steam lines from the outboard MSIVs to the turbine stop valve, the main steam bypass lines to the bypass valves, and various main steam branch piping, were seismically analyzed in accordance with ANSI B31.1, as part of the original plant design. In fact, this main steam piping and all branch lines 3.0 inches in diameter and larger have been seismically analyzed up to the seismic anchors downstream of the isolation valves for the branches. The licensee stated that the design methods for these analyzed lines were consistent with Seismic Category I qualification methods for DAEC and that design capacities are expected to be adequate to ensure good seismic performance under the DAEC safe shutdown earthquake (SSE).

The main steam drain to the condenser and interconnected piping, on the other hand, were designed by rule or by approximate methods, without consideration of seismic loads. They are composed of welded steel piping and standard component supports. The licensee stated that these pipes are bounded in diameter and diameter-to-thickness ratio by those installed in the earthquake experience database plants, as evidenced by findings in the BWROG report, the supplemental and updated earthquake performance data discussed above, and the additional information provided in the December 21, 1994, submittal. According to the licensee, the piping systems in the database represent the full range of design configurations and supporting details that could be encountered in commercial power plant piping design practice. The spectrum of housed piping systems include a wide variety of support conditions, geometrical configurations, size distributions, and several other piping system variables. As a result, the licensee stated that piping (i.e., similar to the alternative leakage treatment system piping) designed to commercial codes and standards has exhibited good seismic performance except when limited and identifiable critical design characteristics are present.

The licensee's contention that the DAEC piping was representative of the seismic experience database piping was also based on the comparison of DAEC design attributes and installation codes (e.g., designs to ANSI B31.1 code, recommended pipe spans, piping flexibility, unique piping layout, pipe diameters, support types and configurations) with those of database systems, as well as the results of the seismic verification walkdowns. Based on the above, and the fact that all necessary modifications will be implemented for the identified "outliers," the staff concurred with the licensee that piping position retention and pressure boundary integrity will be maintained by the deadweight supports under normal and SSE loadings.

In the August 15, 1994, and December 21, 1994, submittals, the licensee stated that the condenser shell and pertinent internal structural members are seismically adequate, based on the earthquake experience database. According to the licensee, the DAEC condensers (one high pressure and one low pressure condenser) are bounded in size, weight, and condensing area, by several large condensers in the earthquake experience database, in particular Moss Landing and Ormond Beach. Furthermore, in response to the staff's request, the licensee submitted detailed drawings for the condenser anchorage and evaluation results for the condenser anchorage during an SSE. The drawings indicate that horizontal loads in the direction parallel to the turbine shaft are restrained by two flex plates and the horizontal loads in the direction perpendicular to the turbine shaft are restrained by a shear key at the center of the condenser. The vertical loads are restrained by sixteen 2.25-inch diameter bolts (four bolts in a group in each of the four corners of the condenser) that are embedded about 27 inches into the concrete. The evaluation results indicate that the capacities of the flex plates, shear key, and anchor bolts are greater than the seismic demand. Based on this information, the staff determined that the licensee has provided sufficient information to conclude the adequacy of the anchorage and that overall operability of the DAEC condenser would be maintained under a postulated SSE event.

The staff's assessment of the seismic adequacy of piping supports included a review of the licensee's methodology for anchorage evaluation, as well as the entire support structure. In the August 15, 1994, submittal, the licensee stated that pipe support anchorage was evaluated using the Generic Implementation Procedure (GIP) criteria established by the Seismic Qualification Utility Group (SQUG) and accepted by the staff for the evaluation of the adequacy of equipment anchorage at older operating plants. The licensee further stated that the seismic demand was conservatively determined using a factor of 1.25 times the peak of the appropriate in-structure floor response spectra and compared against the anchorage capacity specified in Appendix C of the GIP. As a result of the demand versus capacity evaluations, the licensee concluded that pipe support anchorages were adequate to resist the seismic demand.

In response to the staff's request of December 1, 1994 (Reference 9), the licensee provided the results of its seismic margin evaluations of 22 supports on the primary and alternate drain paths and 15 additional supports from two typical interconnected piping systems. The methodology utilized to demonstrate the seismic margin is called Conservative Deterministic Failure Margin (CDFM), as described in the EPRI report, EPRI NP-6041-SL, Revision 1, dated August 1991. Since this methodology has not been approved by the NRR staff for licensing review involving Seismic Category I systems, the staff acceptance of the DAEC support evaluation will be based on the available safety margins demonstrated for the supports. According to the licensee, a minimum margin of 2.0 was available to demonstrate the functional operability of the alternate leakage treatment system. The calculated safety margins, in turn, are also related to the use of the new in-structure response spectra of the turbine building, generated in a recent SSI analysis of the building by the licensee. Therefore, the staff determined that the support designs on the alternate leakage treatment pipes are adequate since the SSI analysis of the turbine building has also been determined to be acceptable. The staff conclusion is also applicable to the licensee's evaluation of the identified "outliers," since such evaluation was also based on the use of the newly developed in-structure response spectra for the turbine building, instead of the in-structure response spectra developed during the original construction of the plant.

As indicated in the December 21, 1994, submittal (Reference 10), a modification will be performed by the licensee involving addition of motor operators to valves V03-0004 and V03-0005. The new designations for these valves will become M01362A and B, respectively. These two valves and MOVs M01169, M01170, M01043, and M01044 will be supplied with Class 1E power. The licensee further stated that all valves (i.e., fluid-operated and motor-operated valves) that may be required to change position during or following the earthquake, and instrument racks containing instruments and gages that are in the seismic verification boundary have been evaluated for seismic adequacy.

The licensee further stated, that all valves within the seismic verification boundary that are required to reposition to establish the boundary or treatment path, including V03-0004 and V03-0005, will be included in the ASME Section XI IST program. The highly reliable power source in combination with the required testing for the valves, as discussed above, is believed to provide a high degree of confidence that the subject valves will remain functional. This is acceptable to the staff.

In addition, according to the licensee, the primary and alternate MSIV leakage treatment paths from the main steam lines to the condenser, including interconnected lines up to the boundary valves, will also be included in the ASME Section XI inservice inspection (ISI) program. This is acceptable to the staff.

Based on the above, the staff determined that the DAEC non-seismically analyzed main steam system piping and condenser that will be used for the alternate leakage treatment system compared well with the earthquake experience database, and that the seismic verification walkdowns of the system and subsequent evaluations have addressed characteristics associated with the limited component damage incidents that were observed at the database facilities. The staff also determined that the licensee has taken proper measures to ensure resolution for all of the identified "outliers" and has analytically demonstrated adequate margins of safety for alternate leakage treatment system piping supports. In addition, the staff also determined that the licensee has taken proper measures to ensure the capability of the alternative leakage treatment system valves to perform its function under the design basis loadings.

2.2.6 Conclusion

Based on the above evaluation, the staff concludes that upon completion of the plant modifications necessary for the identified "outliers," and incorporation of alternate leakage treatment piping and valves in the DAEC ISI and IST programs, respectively, there is reasonable assurance that the DAEC main steam lines, main steam drain lines, condenser, and associated interconnected piping and supports will be seismically adequate for the proposed alternate MSIV leakage treatment system. The staff's conclusion is based on (1) the DAEC main condenser is generally enveloped by the condensers in the earthquake experience database, (2) portions of the main steam system piping were seismically analyzed as part of the initial design of the plant, (3) the seismic verification walkdowns indicated that the remaining non-seismically analyzed alternate leakage treatment pipes are represented by those in the earthquake experience database that demonstrated good seismic performance, and (4) the alternate leakage treatment pipe supports were analytically evaluated and found to be adequate to withstand the seismic loads. The staff, therefore, concludes that the licensee's proposed alternate leakage treatment system is seismically adequate to withstand the DAEC safe shutdown earthquake and maintain its functionality, and hence, meets the requirements of GDC-2 of Appendix A to 10 CFR Part 50.

It should be noted that the staff's consideration of the experience-based methodology as presented by the BWROG and IES Utilities Inc., is restricted to its application for ensuring the pressure boundary integrity and functionality of the main steam drain path associated with the MSIV leakage treatment system. The staff's consideration of the methodology for this application is not an endorsement for the use of the experience-based methodology for other applications at Duane Arnold Energy Center.

2.3 DRAIN PATH FUNCTIONAL DESIGN ASSESSMENT

The proposed MSIV leakage alternate (alternate to the existing MSIV-LCS and Regulatory Guide 1.96) drain pathway is considered the primary success (credited) pathway for "treating" MSIV leakage following a LOCA and employs a MSL drain line downstream of the MSIVs. There are two motor-operated valves (MOVs) in series in this line between the MSLs and the main condenser. Both valves must be open to establish the primary drain path. The first (upstream) MOV, MO1043, is normally closed and will fail "as-is" on a loss of power. The second (downstream) MOV, MO1044, is normally closed (with a small bypass orifice around it to allow drainage during normal operation) and will fail "as-is" on a loss of power. Both MOVs are required to be opened following the design basis accident LOCA to establish the primary drain path to support the radiological analysis. Both valves are powered from Class IE sources.

To address the single failure criterion, the licensee also evaluated an alternate drain path to convey the MSIV leakage to the main condenser. The alternate (or secondary) path consists of the bypass lines around the MOVs in the primary drain path. This alternate path contains a "fail open" valve, which essentially bypasses MO1043, and the restricting orifice, which bypasses MO1044. If either valve in the primary flow path failed to open as required, the alternate drain path would be available to convey the MSIV leakage to the main condenser. The licensee committed to update the procedures, as necessary, to address the alternate leakage treatment methods. The licensee also verified in its December 21, 1994, submittal that all valves required to reposition to establish the boundary and treatment paths will be included in the ASME Section XI inservice testing (IST) program.

The licensee further proposed new requirements in the revised TS 4.7.A.2.c.3 related to restoration of acceptable leak rates, if any of the proposed limits are exceeded. The new requirements basically require that if any single MSIV leakage rate exceeds 100 scfh, it will be repaired and retested to meet a leak rate limit of 11.5 scfh per valve (the current criterion for leakage), and that the maximum total leak rate will be restored to less than or equal to 200 scfh. The staff concludes that this new requirement will restore the leakage rates to values that are consistent with the revised radiological analysis and is, therefore, acceptable.

Conclusion

Based on its evaluation as described above, the staff concludes that the design of the alternate treatment method meets the requirements of GDC-41 with respect to performing its safety function with and without offsite power and assuming a single active failure.

2.4 Conclusion

Based on the above reviews, the staff concludes that the alternate leakage path design is acceptable and that the proposed changes to the technical specifications to increase MSIV leak rates limits and to eliminate the LCS are acceptable and should be approved. The staff, therefore, concludes the proposed design is acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Iowa State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATIONS

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (59 FR 47169). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: February 22, 1995

Attachments:
1. References
2. Table 1
3. Table 2
4. Table 3
5. Figure 1
6. Figure 2

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Table 1
Radiological Consequences of Loss-of-Coolant Accident
(rem)

	EAB		LPZ	
	Thyroid	Whole Body	Thyroid	Whole Body
Containment Leakage	13	2	40	3
MSIV leakage	1.1	2.3	10.5	1
Total	14.1	4.3	50.5	4

ATTACHMENT 2

Table 2
Assumptions Used to Evaluate the Loss-of-Coolant Accident

Parameter	Value
Power level	1658 Mwt
Fraction of core inventory released	
Noble gases	100%
Iodine	50%
Iodine initial plate-out fraction	50%
Iodine chemical species	
Elemental	91%
Particulate	5%
Organic	4%
Suppression pool decontamination factor	
Noble gas	1
Organic iodine	1
Elemental iodine	5
Particulate	5
Iodine dose conversion factors	ICRP-30
MSIV leak rate	200 SCFH
Standby gas treatment system	
Filter efficiency	95%
Flow rate	4000 ft^3/min
Primary containment free volume	4.49E+5 ft^3
Dose conversion factors and breathing rates	ICRP-30
Computer Code	Revised TACT-5

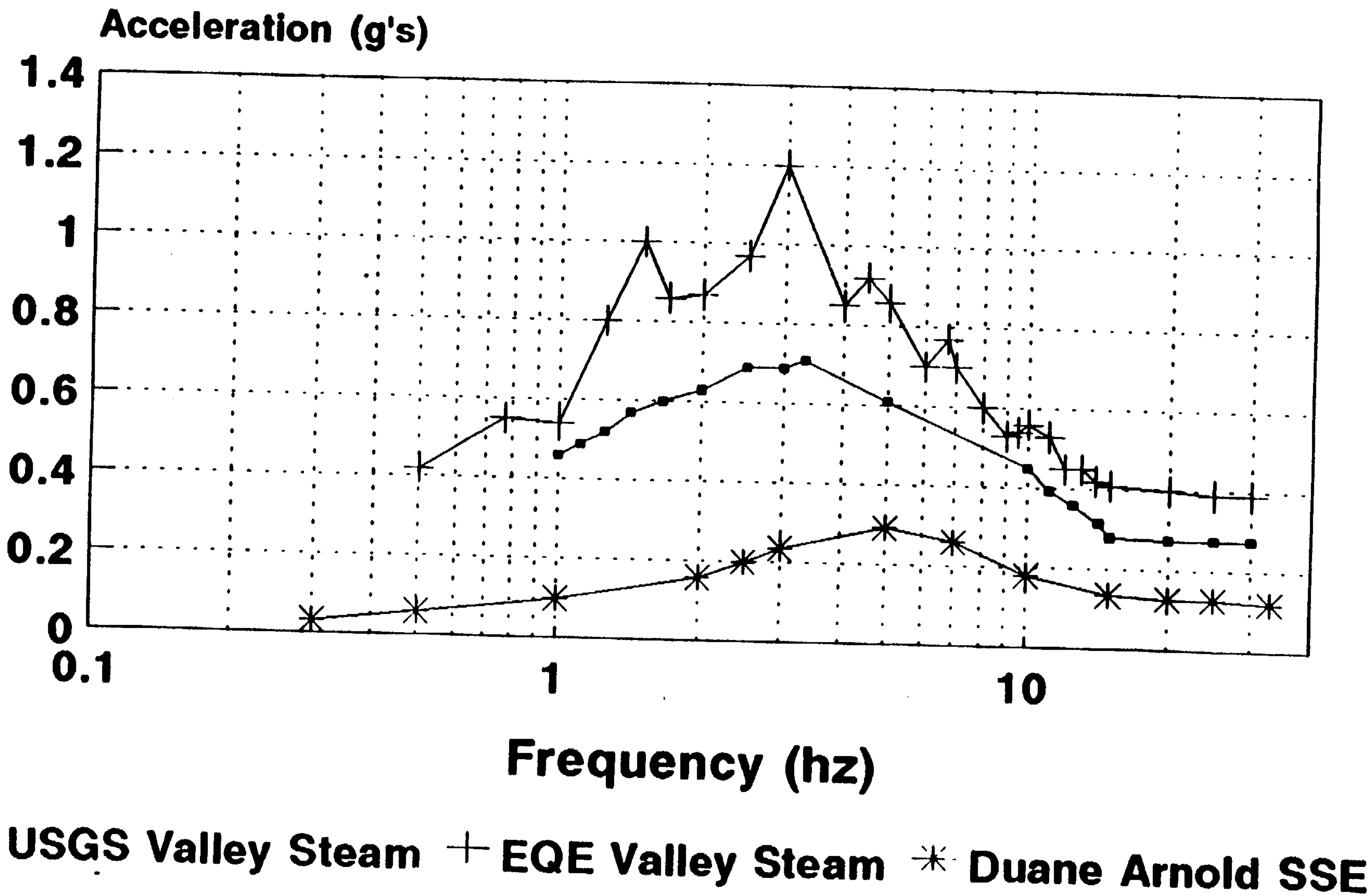
TABLE 3
ASSUMPTIONS AND ESTIMATES OF THE RADIOLOGICAL CONSEQUENCES TO CONTROL ROOM OPERATORS FOLLOWING A LOCA

Control room free volume	1.55E+5 ft^3	
Filtered Intake	1000 CFM	
Unfiltered Intake	67.5 CFM	
Filter Efficiency	95%	
Accident Duration	30 days	
Breathing rate of operators in control room for the course of the accident	3.47×10^{-4} m^3/sec	
Meteorology (wind speeds for all sectors		
0 - 8 hours	5.00×10^{-4} sec/m^3	
8 - 24 hours	2.90×10^{-4} sec/m^3	
24 - 96 hours	1.07×10^{-4} sec/m^3	
96 - 720 hours	2.55×10^{-5} sec/m^3	
Iodine protection factor	7	
Iodine Dose Conversion Factors*	ICRP-30	
Control Room Operator Occupational Factors		
0 - 8 hours	1	
8 - 24 hours	1	
24 - 96 hours	0.6	
96 - 720 hours	0.4	
Doses to control room operators	Thyroid dose* (rem)	Whole body dose** (rem)
	30	<1

*unweighted dose equivalent
**unweighted dose equivalent (red bone marrow) due to immersion in an infinite cloud

ATTACHMENT 4

Duane Arnold MSIV
San Fernando 71 Valley Steam

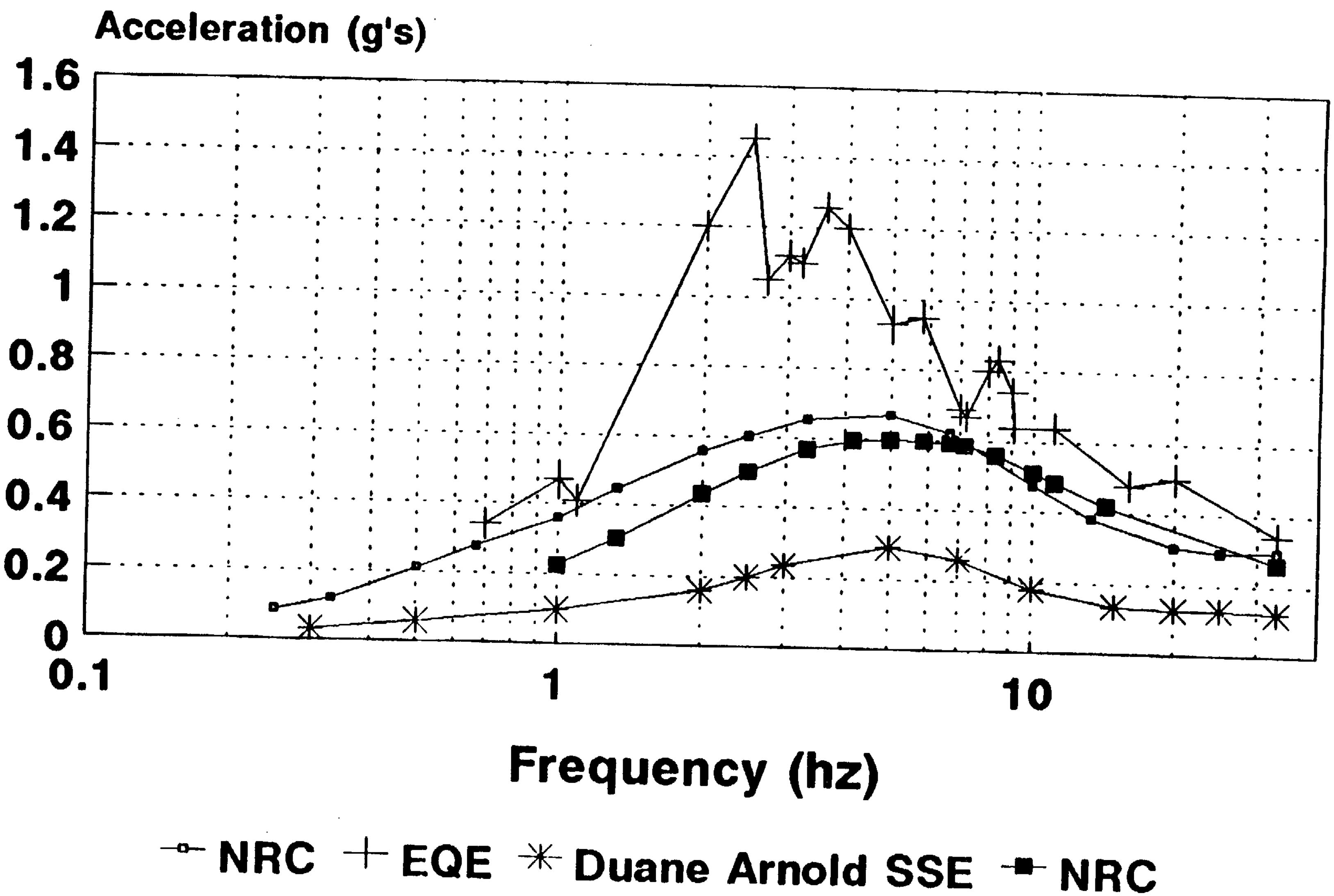


5% damping

Figure 1

ATTACHMENT 5

Duane Arnold MSIV
Cape Mendocino 92 PALCO Co-Gen



5% damping

Figure 2

ATTACHMENT 6