March 21, 2003

Mark A. Peifer
Site Vice President
Duane Arnold Energy Center
Nuclear Management Company, LLC
3277 DAEC Road
Palo, IA 52324-0351

SUBJECT: DUANE ARNOLD ENERGY CENTER - ISSUANCE OF AMENDMENT RE: ONE-TIME EXTENSION OF CONTAINMENT INTEGRATED LEAK-RATE TEST INTERVAL (TAC NO. MB4752)

Dear Mr. Peifer:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 249 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 March 29, 2002, as supplemented by letter dated January 24, 2003.

The amendment changes the surveillance requirement of TS 5.5.12, “Primary Containment Leakage Rate Testing Program,” to allow a one-time five-year extension to the ten-year interval for performing the next Type A containment integrated leakage rate test (ILRT). The change allows ILRT testing within fifteen years from the last ILRT, which was performed in September 1993.

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

Sincerely,

***/RA by Mark L. Padovan for/***

Darl S. Hood, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures: 1. Amendment No. 249 to
License No. DPR-49
2. Safety Evaluation

cc w/encls: See next page

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Distribution w/encls:

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| PUBLIC | GGrant, RIII | GHill (2) | PDIII-1 Reading | WBeckner, TSB |
| ACRS | OGC | THarris | RPalla | SWeerakkaody |
| JPulsipher | MRubin | GBedi | BBurgess, RGN-IIIS | |

**ADAMS Accession No. ML030700243** ***Provided SE input by memo**

| OFFICE | PM:PD3-1 | LA:PD3-1 | EMEB/SC | SLPB/SC* | OGC/NLO | SC:PD3-1 |
|---|---|---|---|---|---|---|
| NAME | MPadovan for DHood | THarris | *DTerao | SWeerakkody | RHoefling | LRaghavan |
| DATE | 03/21/03 | 03/12/03 | 03/11/03 | 02/20/03 | 03/20/03 | 03/21/03 |

**OFFICIAL RECORD COPY**

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 249
License No. DPR-49

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated March 29, 2002, as supplemented by letter dated January 24, 2003, complies with the standards and requirements of the Atomic Energy Act of l954, 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 l0 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. 249 , 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 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA/***

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 21, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 249

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

| Remove | Insert |
|---|---|
| 5.0-17 | 5.0-17 |

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 249 TO FACILITY OPERATING LICENSE NO. DPR-49

NUCLEAR MANAGEMENT COMPANY, LLC

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

## 1.0 INTRODUCTION

By application dated March 29, 2002, as supplemented by letter dated January 24, 2003, Nuclear Management Company, LLC (the licensee), requested changes to the Technical Specifications (TSs) for the Duane Arnold Energy Center (DAEC, or the facility). The licensee's letter dated January 24, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 30, 2002 (67 FR 21291).

The proposed amendment would add an exception to TS 5.5.12, "Primary Containment Leakage Rate Testing Program," which, in effect, would permit a one-time extension to the Type A[1] containment integrated leakage rate test (ILRT) interval from the required 10 years to a test interval of 15 years. Specifically, TS 5.5.12 currently requires the licensee to establish a program "to implement the leakage rate testing of the primary containment as required by Title 10 of the *Code of Federal Regulation* (10 CFR) 50.54(o) and 10 CFR 50, AppendixJ, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. The proposed amendment would continue this statement by adding: "as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J": The first Type A test performed after the September 1993 Type A test shall be performed no later than September 2008."

## 2.0 REGULATORY EVALUATION

The licensee requested the proposed changes in accordance with 10 CFR Part 2, Section 2.101, "Filing of application," 10 CFR Part 50, Section 50.59, "Changes, tests, and experiments," and 10 CFR Part 50, Section 50.90, "Application for amendment of license or construction permit."

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[1] A Type A test is an overall (integrated) leakage rate test of the containment structure.

The regulation at 10 CFR Part 50, Appendix J, Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. DAEC TS 5.5.12 requires that that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide (RG) 1.163. RG 1.163, Section C, "Regulatory Position" states, "licensees intending to comply with the Option B in the amendment to Appendix J should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01 [Nuclear Energy Institute report 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995], rather than using test intervals specified in ANSI/ANS-56.8-1994. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision in NEI 94-01 for extending the test interval an additional 15 months in certain circumstances. The most recent two Type A tests at DAEC has been successful, so the current interval requirement is 10 years.

Thus, the licensee is requesting an addition to TS 5.5.12 that would permit an exception from the guidelines of RG 1.163 regarding the Type A test interval by extending the currently specified 10-year interval to a 15-year interval on a one-time basis. Specifically, the proposed TS states that the first Type A test performed after the September 1993 Type A test shall be performed no later than September 2008. The proposed change does not involve any change to a code, regulatory requirement, or acceptance criteria.

## 3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's regulatory and technical analyses in support of the proposed license amendment as described in the licensee's application dated March 29, 2002, and the additional information provided in the licensee's letter dated January 24, 2003. The detailed evaluation below will support the conclusion 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.

By Amendment No. 219, dated October 4, 1996, the NRC previously approved a revision to the DAEC TSs to incorporate the requirements of Option B for the Type A tests. The change proposed by the licensee's application dated March 29, 2002, involves only the extension of the interval between Type A containment leakage tests; Type B and C containment leakage tests will continue to be performed at the frequency currently required by DAEC's TSs.

### 3.1 Inservice Inspection for Primary Containment Integrity

The leak rate testing requirements of Option B of Appendix J, and the containment inservice inspection (ISI) requirements mandated by 10 CFR 50.55a complement each other in ensuring the leaktightness and structure integrity of the containment. Therefore, a detailed evaluation related to the inservice inspection of the containment and potential areas of weaknesses in the containment is performed in the following section.

DAEC utilizes a General Electric boiling water reactor (BWR) enclosed within a Mark I containment. The containment design includes a drywell and suppression chamber with interconnecting vent pipes and bellows, primary containment access penetrations, and other process piping and electrical penetrations. The drywell is a steel pressure vessel (0.75 to 3.0 inches thick) with a spherical lower portion and cylindrical upper portion. It is enclosed in 4- to 7-feet-thick, reinforced concrete for shielding, and provides additional resistance to deformation and buckling over areas where concrete backs up against the steel shell. Below the drywell head flange and above the foundation transition zone, the drywell is separated from the reinforced concrete by a gap of approximately 2 inches to allow for thermal expansion. Shielding over the top of the drywell is provided by removable, segmented, reinforced-concrete shield plugs.

The drywell is provided with a removable head to facilitate refueling, one combination double door personnel access lock/equipment lock, one equipment hatch, one personnel access hatch, and one control rod removal hatch. The head and hatches are all bolted in place and have double seals and provisions for leak tests.

The pressure suppression chamber is a steel pressure vessel in the shape of a torus, located below and encircling the drywell. The suppression chamber will transmit seismic loading to the reinforced concrete foundation slab of the Reactor Building. Space is provided outside the chamber for inspection. Access to the chamber is provided by two 4-foot diameter manhole entrances with double gasket (leak testable) bolted covers connected to the chamber by 4-foot diameter steel pipe inserts.

Eight 4-foot, 9-inch diameter vent pipes connect the drywell and the pressure suppression chamber. Jet deflectors are provided in the drywell at the entrance of each vent pipe to prevent possible damage to the vent pipes from jet forces or projectiles which might accompany a pipe break in the drywell. The vent pipes are provided with two-ply expansion bellows to accommodate differential motion between the drywell and suppression chamber. These bellows have test connections which allow for leak testing and for determining that the passages between the two-ply bellows are not obstructed.

The drywell vents are connected to a 3-foot, 6-inch diameter vent header in the form of a torus which is contained within the air space of the suppression chamber. Projecting downward from the header are 48 downcomer pipes, 24 inches in diameter and terminating a minimum of 3 feet below the water surface of the pool and approximately 7 feet above the bottom of the Torus.

The last ILRT for DAEC was performed in September 1993. The next scheduled ILRT begins in October 2003. With the extension of the ILRT time interval, the next overall verification will be performed no later than October 2008. In its letter dated January 24, 2003, the licensee responded to the NRC staff's request for additional information for five issues regarding the ISI of the containment and discussed potential areas of weaknesses in the containment that may not be apparent in the risk assessment. The NRC staff's evaluation of the licensee's response to these issues is discussed below.

### 3.1.1 DAEC's ISI Program

The licensee states that ISI program is established in accordance with the requirements of the 1992 Edition with the 1992 Addenda of Subsections IWE of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The first 10-year containment inspection is divided into three periods as follows:

| | |
|---|---|
| First Period: | May 22, 1998 - May 21, 2001 |
| Second Period: | May 22, 2001 - May 21, 2005 |
| Third Period: | May 22, 2005 - May 21, 2008 |

The inspections performed in accordance with the ASME Code, Section XI, Subsection IWE include the interior liner and exterior concrete surfaces. In general, the areas and items subject to inspection include the Class MC pressure retaining containment surface areas, including structural attachments and penetrations, seals, gaskets, moisture barriers, pressure retaining bolting, and Class MC supports. Exceptions taken to the ASME Code requirements have been documented and approved by the NRC staff as relief requests. Inaccessible areas are evaluated for degradation when conditions in accessible areas indicate the presence of or result in degradation not meeting the established acceptance standards.

The ASME Code, Section XI, Subsection IWE containment inspections provide assurance that degradation of the containment structure is identified and corrected before a containment leakage path is introduced.

3.1.2 Implementing Subsubarticle IWE-1240

Subsubarticle IWE-1240 of Subsection IWE of the ASME Code requires the identification of the surface areas requiring augmented examinations. Paragraph IWE-1241 provides the selection criteria for those areas requiring augmented examinations. The surface areas likely to experience accelerated degradation and aging require augmented examinations. The licensee states that DAEC's augmented examination areas are identified as follows:

- interior and exterior containment surface areas that are subject to acceleration corrosion with no or minimal corrosion allowance or areas where the absence or repeated loss of protective coatings has resulted in substantial corrosion and pitting. Typical locations of such areas are those exposed to standing water, repeated wetting and drying, persistent leakage, and those with geometries that permit water accumulation, condensation, and microbiological attack. Such areas may include penetration sleeves, surfaces wetted during refueling, concrete to steel shell or liner interfaces, embedment zones, leak chase channels, drain areas, or sump liners.

- interior and exterior containment surface areas that are subject to excessive wear from abrasion or erosion that causes a loss of protective coatings, deformation, or material loss. Typical locations of such areas are those subject to substantial traffic, sliding pads or supports, pins or clevises, shear lugs, seismic restraints, surfaces exposed to water jets from testing operations or safety relief valve discharge, and area that experience wear from frequent vibrations.

Torus Augmented Examination:

DAEC's torus was initially coated in 1973 with an inorganic zinc protective coating, CarboZinc 11, with a 4-foot wide band epoxy coating, Phenoline 368, at the waterline. The licensee has a proactive inspection program for the interior of the torus. The licensee inspected and performed coating repairs to the torus in 1977, 1980, 1981, and 1983. In 1985, all internal surfaces of the torus shell and external surfaces of the vent system were recoated with CarbZinc 11. Additional inspections and repairs to the coating were performed in 1987, 1988, 1990, 1992, and 1993. During the 1993 inspection, the licensee performed a quantitative inspection on a one square foot evaluation area representative of the worst case corrosion observed during the qualitative inspections. A grid was established and the coordinates of each pit in the evaluation area were recorded so that the rate of corrosion (pit depth) could be monitored and trended in the future. The licensee makes repairs to the coating as necessary.

The licensee states that it will continue to perform a general visual examination on 100 percent of the torus exterior and interior surfaces (above the water line) each period. In addition, a visual (VT-1) examination on 100% of the surfaces (below the water line) will be performed twice per interval. Since both sides are accessible for visual examination, no volumetric examination is required.

This examination monitors the coating on the interior surface of the torus. Areas which are detected to have a “corrosion cell” (small area of uncoated metal, typically 1/4" to 1/2") will be repaired. Performing a general visual of 100% of the exterior and interior surfaces (above the water line), VT-1 of 100% of the inferior surfaces (below the water line) twice per interval, provides assurance that the structural integrity of the torus is acceptable. Repairs to the coating are performed when necessary.

During refueling outage (RFO) 16, a visual examination (VT-3) was conducted on the exterior surface of Torus Bay 15. This examination revealed a degraded condition in the coating (1-inch x 2-inch area) which required an engineering evaluation. The licensee’s engineering evaluation accepted the degraded condition, however the cause of the degraded condition was determined to be leakage of water and grease from motor operated valve (MOV) MO-2001, which is located above. This MOV is a residual heat removal (RHR) loop A drywell spray outboard isolation valve. A repair was performed on the coating. This 1" x 2" area is currently included in the Augmented Category E-C, Item E4.11.

During the last refueling outage (RFO 17), inspection of welds performed in accordance with the DAEC IWE Program identified a portion of weld, located between the torus shell and ring girder, that was incomplete for approximately 26-1/4". The weld was repaired with underwater welding (see the licensee’s letter dated May 1, 2001, “NDE-R042 Request for Authorization to Use Code Case N-516-1").

3.1.3 IWE Table-2500-1, Examination Categories E-D and E-G for Seals and Gaskets, and Examination and Testing of Bolts

By letter dated October 19, 1999, the NRC staff previously authorized Relief Requests MC-R002 and MC-R003 for examination Categories E-D and E-G. In relief requests MC-R002 and MC-R003, an alternative requirement eliminated the need to perform visual examination of seals, gaskets, and bolts; instead, these items are tested in accordance with Option B of 10 CFR Part 50, Appendix J. The Type B test frequencies discussed in MC-R002 and MC-R003 are in accordance with the DAEC Performance-Based Containment Testing Program Manual. The initial test frequency for performing a leak test on Type B components such as seal, gaskets, and bolts (other than airlocks) is each refueling cycle, not to exceed 30 months, until acceptable performance is demonstrated. Following the completion of two consecutive periodic tests with the results within allowable limits, the testing interval may be extended to 120 months. If a test result is greater than the allowable limit, then Type B testing shall be performed each refueling cycle until acceptable performance is demonstrated.

Regardless of the above schedule, any repairs or disassembly of a component with a seal, gasket, or bolted connection requires a post-maintenance Appendix J Type B test.

Thus, the licensee does not rely solely on the Type A testing for seals, gaskets, or bolted connections at DAEC.

3.1.4 Integrity of Stainless Steel Bellows

In the past, the NRC staff has found that two-ply stainless steel bellows are susceptible to trans-granular stress corrosion cracking, and the leakage through them is not detectable by Type B testing (see NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing"). The licensee states that in response to Information Notice 92-20, it evaluated and modified the test method used to measure leakage for DAEC's two-ply bellows to include provisions to detect potential damage to the bellows prior to determining leakage rate. The drywell-torus vent bellows are tested at a 120-month interval in accordance with the DAEC Performance Based Containment Testing Program and Regulatory Guide 1.163. The licensee states that there have been no ILRT failures of the drywell-torus vent bellows in the last ten refueling outages.

3.1.5 Inspection of Embedded Side of the Containment Steel Shell

The NRC staff is concerned that inspections of some reinforced concrete and steel containment structures at other nuclear power plants have identified degradation (e.g., corrosion) on the embedded side of the containment steel shell of the primary containment that cannot be inspected. The major areas of the Mark I containment that cannot be inspected are the vertical portion of the drywell shell and that part of the shell that is sandwiched between the drywell floor and basemat. A summary of work performed in the inaccessible region of the containment follows:

The licensee states that the inspections of the containment are performed during the time between ILRTs. The extension of time between ILRTs will not affect the inspections. The performance-based ILRT program guidance (NEI 94-01 and RG 1.163) requires a minimum of three visual examinations of accessible interior and exterior surfaces of the containment system to allow for early revealing of evidence of structural deterioration. The discrepancies identified in the liner, penetrations, and concrete are documented and dispositioned in accordance with the appropriate Code and design requirements. The details are as discussed above. The licensee states that approximately 85 percent of the containment surface area is accessible (i.e., approximate 15 percent of the area is inaccessible).

The licensee states that the potential leakage due to age-related degradation in areas that cannot be inspected is factored into DAEC's risk assessment which supports the requested ILRT interval extension from 10 to 15 years.

3.1.6 Maintaining Positive Pressure in The Containment

The licensee stated that during power operation the primary containment atmosphere is inerted with nitrogen to ensure that no external sources of oxygen are introduced into containment. The Containment Atmosphere Control System provides a supply of makeup nitrogen to maintain primary containment oxygen concentration within TS limits. That system automatically maintains pressure between approximately 0.5 and 1 psig during power operation. Primary Containment pressure is continuously recorded in the Control Room on both paperless and pen recorder and DAEC's Operations staff monitor it via daily and shiftly surveillance. Additionally, primary containment high and low pressure is annunciated in the Control Room to alert DAEC's Operators to off normal conditions. The NRC staff finds that this continuous monitoring of slight positive pressure will ensure that areas of containment degradation are detected before they could result in large leakage rates.

3.1.7 Conclusion

On the basis of the discussion above, the NRC staff finds that implementation of the licensee's containment ISI program, including the areas covered by augmented inspections, will provide reasonable assurance that the containment structural and pressure integrity will be maintained during the extended ILRT period.

On the basis of its review of the information provided by the licensee, the NRC staff finds that (1) the structural integrity of the containment vessel is adequately verified through periodic inservice inspections as required by Subsection IWE of the ASME Code, Section XI; (2) the integrity of the penetrations, containment isolation valves and bellows is periodically verified through Type B and Type C tests as required by 10 CFR Part 50, Appendix J and the DAEC's TSs, and (3) the potential leakages from areas that cannot be inspected are factored into and addressed in the licensee's risk assessment. In addition, the system pressure tests for the containment pressure boundary (i.e., Appendix J tests, as applicable) are required to be performed following repair and replacement activities in accordance with Subarticle IWE-5000 of the ASME Code, Section XI. Significant degradation of the primary containment pressure boundary is required to be reported under 10 CFR 50.72 or 10 CFR 50.73.

The NRC staff concludes, on the basis of the considerations discussed above, that the licensee has adequate procedures to examine and monitor potential age-related and environmental degradations of the pressure retaining components of the DAEC primary containment. Thus, granting the one-time extension of performing the ILRT, as proposed by the licensee, is acceptable.

3.2 Risk Impact Assessment

In its application dated March 29, 2002, the licensee provided a risk impact assessment of extending the Type A test interval to 15 years. The licensee provided additional analysis and information in a letter dated January 24, 2003. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," and RG 1.174, "An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," dated September 1995, provided the technical basis to support rulemaking to revise leakage rate testing requirements contained in

Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in EPRI Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The EPRI study estimated that relaxing the test frequency from 3 in 10 years to 1 in 10 years will increase the average time that a leak detectable only by a Type A test goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10 percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage for the PWR and BWR representative plants confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3 in 10 years to 1 in 20 years leads to an "imperceptible" increase in risk on the order of 0.2 percent and a fraction of one person-rem per year.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem/year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the Option B rulemaking in 1995, the NRC staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than $10^{-6}$/year and increases in large early release frequency (LERF) less than $10^{-7}$/year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original 3 in 10 year interval. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the change in the conditional containment failure probability for the proposed change to demonstrate that the defense-in-depth philosophy is met.

The licensee provided an analysis which estimated all of these risk metrics and whose methodology is consistent with previously approved submittals. The following conclusions can be drawn from the analysis associated with extending the Type A test frequency:

1. A slight increase in risk is predicted when compared to that estimated from current requirements. Given the change from a 3 in 10 year test interval to a 1 in 15 year test interval, the increase in the total integrated plant risk, in person-rem/year, is estimated to be about 0.24 percent. This increase is comparable to that estimated in NUREG-1493, in which it was concluded that a reduction in the frequency of tests from 3 in 10 years to 1 in 20 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. The increase in LERF resulting from a change in the Type A test interval from the original 3 in 10 years to 1 in 15 years is estimated to be $3.7 \times 10^{-8}$/year. However, there

is some likelihood that the flaws in the containment estimated as part of the Class 3b frequency would be detected as part of the IWE/IWL visual examination of the containment surfaces (as identified in the ASME Code, Section XI, Subsections IWE/IWL). The most recent visual examination of the Duane Arnold containment was performed in 2001. The next scheduled IWE/IWL containment visual examination is 2003. Visual examinations are expected to be effective in detecting large flaws in the visible regions of the containment, and would reduce the impact of the extended test interval on LERF. The licensee performed additional risk analysis to consider the potential impact of corrosion in inaccessible areas of the containment shell on the proposed change. The risk analysis considered the likelihood of an age-adjusted flaw that would lead to a breach of the containment. The risk analysis also considered the likelihood that the flaw was not visually detected but could be detected by a Type A test. When possible corrosion of the containment surfaces is considered, the increase in LERF resulting from a change in the Type A test interval from the original 3 in 10 years to 1 in 15 years is estimated to be $3.8 \times 10^{-8}$/year. The NRC staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The licensee estimates the change in the conditional containment failure probability to be an increase of 0.4 percentage points for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. The NRC staff finds that the defense-in-depth philosophy is maintained based on the change in the conditional containment failure probability for the proposed amendment.

On the basis of the above, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines, while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

## 4.0 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.

## 5.0 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 or changes a surveillance requirement. The NRC 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 (67 FR 21291). 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.

## 6.0 CONCLUSION

The NRC 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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