SAFETY EVALUATION OF THE DUANE ARNOLD ENERGY CENTER

Docket No: 50-331

U.S. ATOMIC ENERGY COMMISSION
DIRECTORATE OF LICENSING
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SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

IOWA ELECTRIC LIGHT AND POWER COMPANY

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

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ABBREVIATIONS

a-c	alternating current
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
AEC	United States Atomic Energy Commission
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transients Without Scram
B_4C	Boron carbide
BTU/hr-ft^2	Heat flux, British Thermal Units per hour per square foot
BTU/hr-ft^2-°F	Heat transfer coefficient, British Thermal Units per hour per square foot per degree Fahrenheit
BWR	Boiling Water Reactor
cal/gm	calories per gram
cc/min	Cubic centimeters per minute
cfm	Cubic feet per minute
CHF	Critical heat flux
CSS	Core Spray System
DBA	Design Basis Accident
DBE	Design Basis Earthquake
d-c	direct current
delta T	Differential Temperature

ABBREVIATIONS (cont'd)

Δ k	Reactivity change
DAEC	Duane Arnold Energy Center
DNB	Departure from nucleate boiling
ECCS	Emergency Core Cooling System
EPR	Ethylene propylene rubber
°F	Degrees Fahrenheit
FRVS	Filtration, Recirculation and Ventilation System
FSAR	Final Safety Analysis Report
ft	feet
g	acceleration of gravity, 32.2 feet per second per second
GDC	AEC General Design Criteria
GE	General Electric Company
gpm	Gallons per minute
Gd_2O_3	Gadolinium Oxide
HEPA	High Pressure Coolant Injection System
Hz	Hertz, cycles per second
ICRP	International Commission for Radiation Protection
IEEE	Institute of Electrical and Electronics Engineers
IELP	Iowa Electric Light and Power Company
in	inch
IPS	Inside pipe size
kV	Kilovolt

ABBREVIATIONS (cont'd)

kW	Kilowatt
lb	Pound
LOCA	Loss-of-coolant accident
LPCI	Low Pressure Coolant Injection System
LPZ	Low Population Zone
m	meter
MM	Modified Mercalli (earthquake intensity)
MCHFR	Minimum critical heat flux ratio
mph	Miles per hour
mrem	Millirem
m/s	Meters per second
MSL	Mean sea level
MWD/T	Megawatt - days of energy production per ton of UO_2
MWe	Megawatts electrical
MWt	Megawatts thermal
NDT	Nil ductility transition
NOAA	National Oceanic and Atmospheric Administration
NPSH	Net positive suction head
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
pH	Measure of hydrogen ion concentration on a logarithmic scale

ABBREVIATIONS (cont'd)

PMH	Probable maximum hurricane
ppb	Parts per billion
ppm	Parts per million
PSAR	Preliminary Safety Analysis Report
psi	Pounds per square inch
psig	Pounds per square inch gauge (above atmospheric)
QA	Quality Assurance
QAM	Quality Assurance Manager
QAP	Quality Assurance Program
QC	Quality Control
RCS	Reactor Coolant System
Rem	Roentgen equivalent man
RHR	Residual Heat Removal System
RWM	Rod worth minimizer
scfm	Standard cubic feet per minute
SER	Safety Evaluation Report
sec/m^3	Seconds per cubic meter
SGTS	Standby Gas Treatment System
SS	Stainless Steel
SSWS	Station Service Water System
UO_2	Uranium dioxide
µCi/cc	Microcuries per cubic centimeter
10 CFR	AEC, Title 10 Code of Federal Regulations

ABBREVIATIONS (cont'd)

Part 2	AEC Rules of Practice
Part 20	AEC Standards for Protection Against Radiation
Part 50	AEC Licensing of Production and Utilization Facilities
Part 100	AEC Reactor Site Criteria

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

This report is the Atomic Energy Commission's safety evaluation of the Iowa Electric Light and Power Company's application for a license to operate the Duane Arnold Energy Center (DAEC). The application was filed by Iowa Electric Light and Power Company (IELP, hereafter referred to as the applicant), and the Corn Belt Power Cooperative and the Central Iowa Power Cooperative (hereafter referred to as the co-applicants). The applicant and co-applicants will be co-owners of the facility. The Iowa Electric Light and Power Company is responsible for the design and construction of the facility and will be responsible for its operation. Therefore, in this Safety Evaluation, the term "applicant" refers to Iowa Electric Light and Power Company. When the intent is to refer to the other participating companies, they will be specifically identified.

The Atomic Energy Commission reported the results of its review at the Construction Permit stage in a Safety Evaluation dated February 13, 1970. Following public hearings before an Atomic and Safety Licensing Board in Cedar Rapids, Iowa, the Director of Reactor Licensing issued Provisional Construction Permit No. CPPR-70 on June 22, 1970. An Amendment of this construction permit was published on December 14, 1972 to delete certain requirements concerning Federal and State laws for protection of the environment.

The Duane Arnold Energy Center consists of a single unit boiling water reactor on a 480-acre site located on the west bank of the Cedar

see Pg [illegible]

River in Linn County, Iowa, approximately eight miles northwest of the city of Cedar Rapids. Since the Director of Regulation had granted on March 11, 1970, an exemption under the provisions of Section 50.12, 10 CFR Part 50,[2] construction work associated with facility structures began in March 1970. Authorized site related work such as land clearing had begun earlier, in March 1969.

On March 1, 1972, the applicant tendered an amended application for an operating license (OL) with six copies of the Final Safety Analysis Report[1] (FSAR) that were used by the AEC during a three week preliminary safety review. Inasmuch as more information was needed for the initial filing, the amended application for an OL was not officially docketed for the extended safety review until May 8, 1972; at that time, the FSAR and its Amendment No. 1 providing additional information were docketed and distributed.

The amended application for an OL is required by Part 50.34(b) of 10 CFR Part 50. The amended application requests a license to operate the facility at a thermal power level of 1658 megawatts (MWt) for which the corresponding ultimate electric output of the plant is expected to be about 589 megawatts-electric (MWe). The plant's thermal power level in the construction permit application was the rated thermal power level of 1593 MWt. In its Safety Evaluation for the Construction Permit review, the regulatory staff had indicated it would "perform a safety evaluation to assure that the core can be operated at a higher power level." Therefore, we have performed an evaluation of thermal, hydraulic, and nuclear characteristics of the core as supplied by the applicant for both the rated and ultimate power levels. The evaluation of engineered safety features was made at the higher power level as was our evaluation of the results of abnormal operational transients. However, before the applicant is permitted normal power operation at the higher level, a program of progressive power increase testing, documentation, and evaluation must be accomplished by the applicant. This program is described in Amendment 9 to the FSAR and will be appropriately delineated in the Technical Specifications.

During our review of the information submitted in the FSAR, we requested the applicant to provide additional information needed for our evaluation. This additional information was provided in amendments to the OL application. We also held numerous meetings with the applicant to discuss and clarify the technical information submitted. As a result, we requested a number of changes to be made in the design and planned operation of the facility; these changes are described in the applicant's Amendments (No. 1 through 11) to the FSAR and are discussed in appropriate sections of this Safety Evaluation. The FSAR and its amendments have been made available for review by members of the public at the Atomic Energy Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C. and at the Cedar Rapids Public Library, 426 Third Avenue, S.E., Cedar Rapids, Iowa, 52401. The applicant has submitted its Industrial Security Plan and certain design information on the nuclear fuel as proprietary documents. We have determined that these documents may be withheld from public disclosure under the Commission's Rules and Regulations, 10 CFR Parts 2.790(d)[3] and 9.5(a)(4).[4] Accordingly, these documents will be withheld from public disclosure in accordance with the provisions of Section 9.10 of 10 CFR Part 9.

A chronology of the review by the regulatory staff is included in Appendix A of this evaluation.

1.2 General Plant Description

The Duane Arnold Energy Center employs a nuclear steam supply system consisting of a boiling water reactor. There are sixteen jet pumps supplied by two recirculating water lines, four main steamlines, and two feedwater lines. Fuel for the reactor will be slightly enriched uranium-dioxide (UO_2) in sintered ceramic pellets. Some of these ceramic fuel pellets will contain gadolinium-oxide (Gd_2O_3) in a mixture with the uranium-dioxide. The gadolinium is a "burnable poison" for power pattern and reactivity control that permits better fuel economy and elimination of the boron curtain neutron absorbers found in older plants. The fuel pellets are enclosed in Zircaloy-2 cladding tubes which are evacuated, backfilled with helium, and sealed by welding Zircaloy end plugs in each end. A fuel channel encloses a bundle of 48 fuel rods in a 7 x 7 array; the channel is made of Zircaloy-4. Water flowing through the core serves as both a moderator of neutrons and a coolant. Movement of water and a two phase water-steam mixture through the core is accomplished by the driving force from the 16 jet pumps (8 per recirculation line) and 2 recirculation pumps and from convective forces. Steam from the boiling process in the reactor core is demoisturized and dried, then vented through the four main steamlines to the turbine-generator where its energy is converted into electricity. The steam then exhausts to a condenser located beneath the turbine where the condensate is collected and ultimately returned through a clean-up system for recycling through the reactor vessel and core. The cooling water for the turbine steam condenser is supplied in a closed system that includes two forced draft cooling towers. Makeup water to replenish evaporative losses, windage, and blowdown from the circulating condenser cooling water will be supplied from the Cedar River.

An off-gas treatment system consisting of a recombiner, condenser, moisture separator, gas reheater, prefilter, and charcoal absorber beds will provide for retention of noble gases for decay to concentration levels acceptable for release with the exhaust from the 100 meter stack

The primary reactor coolant pressure boundary includes the reactor vessel, the recirculation lines, main steamlines, feedwater lines, and branch lines to their outermost isolation valve. Enclosing this system is the primary containment structure of welded, inspected, and pressure-tested steel in a light-bulb configuration called the "drywell." Beneath and around the base of this "drywell" structure is the torus shaped "wetwell" of metal, constructed to the same standards as the drywell. The wetwell is connected to the drywell via downcomers and vents to permit the passage and condensation of any steam (vapor suppression) that may be accidentally discharged into the drywell, thereby limiting the pressure buildup below the containment maximum design pressure of 62 psig. Piping restraints have been designed and installed within the containment to limit the movement of piping during its postulated post-rupture oscillations (pipe whip). A hydrogen control system for containment atmosphere dilution (CAD) with nitrogen is provided for the normal operational containment inerting and for any post-LOCA needs. Isolation of the primary containment occurs automatically whenever there exists a potential for the uncontrolled release of radioactivity. For instance, the primary containment and the nuclear steam supply system are isolated and shut off respectively for the unusual conditions of low water level in the reactor vessel, high radiation level in main steamline, main steamline high flow or low pressure, primary containment high pressure, and many others described in Section 7 of the FSAR.

The reactor protection system (RPS) provides the means to protect against conditions that may cause fuel failures or a breaching of the nuclear system process barrier, thereby limiting uncontrolled releases of radioactivity. The RPS initiates a reactor scram following an abnormal operational transient or pressure pulse, or following a gross failure of fuel or the nuclear system process barrier. The RPS is a reliable system designed to meet the standards specified in IEEE-279.[5] Limits for RPS function are set forth in the Technical Specifications.

Normal reactivity control or rapid scram (shutdown) of the reactor is achieved by the bottom-entry cruciform-shaped control rods (neutron absorbers) that are moved vertically in the spaces between fuel assembly channels by a hydraulic mechanism; water is the hydraulic fluid, and for rapid insertion, nitrogen under pressure in an accumulator provides the driving force. Each control rod is independent of the other rods and has its own control and hydraulic system. A rod worth minimizer (RWM) is available to control positive reactivity insertion over a certain power range. To limit the effect of the reactivity insertion following a postulated control rod drop accident, the applicant will adopt and install the rod sequence control system (RSCS) or other method finally prescribed and approved by the regulatory staff for the Browns Ferry and Peach Bottom 2/3 vintage plants. A standby liquid control system is also available for use in injecting a boron solution into the reactor for emergency, long-term reactivity control.

Engineered safety features provide the capability to isolate containment, shut down the reactor, restrict radioactivity releases to acceptable minimum levels, provide for heat removal for long-term core cooling, and condense steam within the primary containment. Details on these engineered safety features are presented elsewhere in this Safety Evaluation.

The reactor building (RB) encloses the reactor and its pressure-suppression type primary containment system. The reactor building houses the refueling and reactor servicing equipment, fuel storage areas, auxiliary equipment, core standby cooling system, reactor cleanup filter demineralizer system, standby liquid control system, control rod drive system, the RPS, electrical equipment, heating and ventilation, and the standby gas treatment system (SGTS). Operation of the SGTS will produce a negative internal pressure after building isolation such that the RB atmosphere is filtered and discharged via the SGTS and plant stack at a rate equal to one building volume per day. Other structures such as the turbine building, the control building, the administration building, pump house, the intake structure and pumping facility, the cooling towers and 100 meter stack are described in varying detail in this evaluation but are also amply covered in appropriate sections of the FSAR and its amendments.

1.3 Comparison with Similar Facilities

Many features of the design of Duane Arnold Energy Center are similar to those we have evaluated and approved previously for other nuclear power plants now under construction or in operation. To the extent feasible and appropriate, we have made use of our previous evaluations during our review of those DAEC features which are substantially the same as those earlier considered. Where this has been done, the appropriate sections of this evaluation will include the identification of the other facilities involved. Our Safety Evaluations for these other facilities are published and are available for public inspection at the AEC's Public Document Room at 1717 H Street, N.W., Washington, D.C.

1.4 Identification of Agents and Contractors

General Electric Company is furnishing the nuclear steam supply system for the Duane Arnold Energy Center, including the first fuel loadings and the turbine-generator for the station. For those items of the plant within its scope of work, General Electric has acted as procurement agent.

Bechtel Corporation is the architect-engineer firm and the facility constructor. In this capacity, Bechtel has designed and provided the balance-of-plant systems.

The Chicago Bridge and Iron Company has supplied the on-site fabricated reactor vessel and the containment vessels. The firm of John A. Blume and Associates was retained by the applicant for consulting work on dynamic analysis of structures. Other firms associated with this facility included: Commonwealth Associates, Nuclear Services Corporation, Dames and Moore, TRC of New England, Biotest Laboratories, NUS Corporation, and Pickard, Lowe, and Associates.

1.5 Summary of Principal Review Matters

This Safety Evaluation summarizes the results of the technical evaluation of the Duane Arnold Energy Center performed by the Commission's Regulatory Staff. Our evaluation included a technical review of the information submitted by the applicant, the principal portions of which are summarized below:

a. We reviewed the population density and land use characteristics of the site environs and the physical characteristics of the site, including seismology, meteorology, geology, and hydrology to determine that these characteristics have been adequately described and was given appropriate consideration in the plant design, and that the site characteristics are in accordance with the Commission's siting criteria (10 CFR Part 100)[6] taking into consideration the design of the facilities, including the engineered safety features provided.

b. We reviewed the design, fabrication, construction, testing, and expected performance of the plant structures, systems, and components important to safety to determine that they are in accord with the Commission's General Design Criteria[7] (GDC), other appropriate codes and standards, and the Commission's Quality Assurance Criteria,[8] and that any departures from these criteria, codes, or standards have been identified and justified.

c. We evaluated the response of the facilities to various anticipated operating transients and to a broad spectrum of postulated accidents, to determine that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed those of all the other accidents considered. This review included evaluation of the applicant's analysis of core thermal and hydraulic performance at the ultimate thermal power level of 1658 MWt. We performed conservative analyses of the design basis accidents to determine, in the very unlikely event of their occurrence, that the calculated offsite doses that might result do not exceed the Commission's guidelines for site acceptability given in 10 CFR Part 100.[6]

d. We evaluated the applicant's plans for the conduct of plant operations, the organizational structure, the technical qualifications of operating and technical support personnel, the measures taken for industrial security, and the planning for actions to be taken in the unlikely event of an accident that might affect the general public. Our evaluation in this area was designed to determine that the applicant is technically qualified to operate the facilities and has established effective organizations and plans for continuing safe operation of the facilities.

e. We evaluated the design of the systems provided for control of the radiological effluents from the facilities to determine if these systems can control the release of radioactive wastes from the station within the limits of the Commission's regulations and if the applicant will operate the facilities in such a manner as to reduce radioactive releases to levels that are as low as practicable.

f. We evaluated the financial qualifications of the applicant and the other two participating companies, to determine that the financial position of the applicant and co-applicants are adequate to operate the Duane Arnold Energy Center in accordance with activities permitted by the operating license.

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location

The Duane Arnold Energy Center is situated on a 500 acre tract of land located in Linn County, Iowa on the west bank of the Cedar River. The nuclear facility is located approximately 2-1/2 miles NNE of the village of Palo, Iowa and 8 miles northwest of Cedar Rapids, Iowa.

2.1.2 Site Description

The minimum exclusion distance, as defined by the applicant, is 400 meters from the plant stack to the nearest property line. The nearest boundary of Cedar Rapids, which is the nearest boundary of a densely populated geographic center containing more than 25,000 persons, is about eight miles from the plant site, and, therefore, the population center distance is considered to be eight miles. Based upon this population center distance, the low population zone (LPZ) distance is 6 miles (9650 meters). Figure 1 shows the exclusion area for this site.

The Duane Arnold site is located on a relatively flat plain, at approximately 750 feet mean sea level, which extends from the site towards the village of Palo to the southwest. Across the Cedar River from the site, the land rises from an elevation of 750 feet to an elevation of about 900 feet within a horizontal distance of 2,000 feet. These slopes are heavily wooded. To the northwest, the terrain rises to an elevation of 850 feet. Adjacent and to the east of the site is a heavily wooded low area. The general topographical features in this section of the Cedar River drainage area consist of broad valleys with relatively narrow flood plains.

2.1.3 Population and Population Distribution

The closest cities with population exceeding 25,000 are Cedar Rapids, Iowa with a population of 110,600 approximately 8 miles southeast, Waterloo at 40 miles northwest with a population of 75,500 and Iowa City approximately 35 miles to the southeast with a population of 46,800. The area within 5 miles of the site has a population of about 2735. There are three farm houses within one mile of the plant. The closest farm dwelling is 2900 feet from the plant. There are three schools located within the LPZ. These schools are in Palo (175 students), Shellsburg (370 students), and Toddville (150 students) and are located 2.7 miles, 5 miles and 2.7 miles respectively from the site. Figures 2 and 3 show the 1970 and predicted year 2010 cumulative population data relevant to the Duane Arnold site.

2.1.4 Uses of Adjacent Lands and Waters

At the present time, the land surrounding the Duane Arnold Energy Center is predominantly agricultural. The major crop harvested is corn with secondary crops of oats and soybeans. Farm animals raised include cattle, hogs and poultry.

The Cedar River in the area of the site is used for sport fishing, but there is no commercial fishing in the vicinity of the site. The closest area suitable for power boating, water skiing and swimming is in the vicinity of the Seminole Valley Park, 6 miles downstream. Directly east of the site and adjacent to the eastern bank of the Cedar River is a 177 acre conservation area which is undeveloped and is used for hiking, wilderness camping, nature study and hunting.

The only major user of potable water within 50 miles downstream from the Duane Arnold site is the city of Cedar Rapids (about 15 miles downstream) which obtains its water by wells located adjacent to the Cedar River. Major industrial water use within 50 miles downstream is for power plant condenser cooling and process water for other industrial facilities.

2.1.5 Conclusions

Based on the 10 CFR Part 100[6] definitions of the population center distance, and the exclusion area and low population zone distances, on our analysis of the onsite meteorological data from which dilution factors were calculated for various time periods (Section 2.3 of this report), and on the calculated potential radiological dose consequences of design basis accidents (Section 15.0 of this report), we conclude that the exclusion area radius and the low population zone distance are acceptable.

2.2 Nearby Industrial, Transportation and Military Facilities

There are no missile sites within a 10 mile radius of the site. The nearest commercial airport is the Cedar Rapids Municipal Airport located 15 miles south southeast of the plant site. A small landing strip not shown on current aeronautical charts exists approximately 4 miles southeast of the plant. The maximum size aircraft using the turf runway is 3000 lbs. These light airplanes are used for weekend and summertime evening pleasure flying. There are no other airports within 10 miles of the plant.

The staff has reviewed the question of airport proximity to nuclear power plants in various other licensing cases. On the basis of these studies, we conclude that the Duane Arnold site is sufficiently far away from an airport of significant size that the probability of a crash at the site is essentially that associated with general overflights and that the Duane Arnold Energy Center need not be designed or operated with special provisions to protect the facility against the effects of an aircraft crash.

There are no oil or gas lines, mineral mines or petroleum wells within 5 miles of the plant site. There is a rock quarry located approximately 3 miles southwest of the reactor site. The applicant has provided the results of a study by its consultant, Blume and Associates, which indicates that routine blasting at the quarry will have no effect on the safe operation of the Duane Arnold facility.

The applicant will use the seismic instrumentation installed in the nuclear facility to validate the consultant's calculations of ground shock wave velocities and associated accelerations at the plant site due to routine quarry blasting operations. We conclude that the consultant's study demonstrates that any effect on plant safety due to quarry operation is unlikely to occur.

2.3 Meteorology

2.3.1 Regional Meteorology

The climate of eastern Iowa is that of a continental interior, uninfluenced by any proximity of large bodies of water. Such a climate is characterized by cold, dry winters and warm, humid summers. Continental polar air, generally of Canadian origin, is the predominant type of air mass over Iowa in the winter. Summer air masses over this area are predominantly maritime tropical, with origins over the Gulf of Mexico. The other two seasons - spring and autumn - are relatively short; being characterized in the former case by increasing and in the latter by decreasing temperatures and precipitation. High air pollution potential (atmospheric stagnation) exists only rarely in this area, occurring on the average less than one day in a year. Atmospheric diffusion conditions are generally close to the average for all sites in the United States.

2.3.2 Local Meteorology

The plant site is on the west bank of the Cedar River, eight miles northwest of Cedar Rapids, Iowa. From south-southeast through west-southwest of the site, the terrain is flat or gently rolling. The terrain rises and becomes more hilly in the other directions from the site and is heavily wooded across the Cedar River toward the east. During the period 1953-62, eleven tornadoes have been reported within the one degree latitude-longitude square containing the site, giving a mean annual tornado frequency of 1.1. The computed recurrence interval for a tornado within the 500 acre plant site area is 1171 years. The predominant wind flow over the site is from the south, with a secondary flow from the northwest.

2.3.3 Onsite Meteorological Measurements Program

An onsite meteorological measurements program was initiated in January 1971. The program consisted of the installation of and measurements from a 165-ft tower which is located about 1700-ft south-southeast of the reactor building. The tower has wind, temperature and dew point measurement instruments at the 35-ft and 165-ft levels. The applicant has submitted a one year period of data record (1/71-1/72) in joint frequency distribution form, similar to that suggested in Safety Guide 23,[9] to provide a basis for the staff's evaluation of atmospheric diffusion conditions. For building and vent releases, the joint frequency distribution of wind speed and direction measured at the 35-ft level and vertical temperature difference (Δt) between the 35-ft and 165-ft levels was used. For releases from the plant's 100 meter (328-ft) stack, the joint frequency distribution of wind direction and speed measured at the 165-ft level and vertical temperature difference (Δt) between the 35-ft and 165-ft levels was used. The joint frequency distribution data recovery during the one year period of record was 92 percent.

2.3.4 Short Term (Accident) Diffusion Estimates

A ground-level release with a building wake factor, cA, of 911 meters^2 was assumed in the evaluation of short term (0-2 hr. at the site boundary and 0-8 hr. at the LPZ) accidental releases from the buildings and vents. The relative concentration (χ/Q) for 0-2 hours which is exceeded 5% of the time was calculated to be 2.2×10^{-3} sec/m^3 at the minimum site boundary distance of 540 meters. This relative concentration is equivalent to dispersion conditions produced by Pasquill type F stability with a wind speed of 0.3 meters/second. The relative concentration (χ/Q) for 0-8 hours which is exceeded 5% of the time was calculated to be 8.9×10^{-5} sec/m^3 at the LPZ distance of 9654 meters. The assumed 8-24 hour relative concentration was 3.6×10^{-5} sec/m^3.

In the evaluation of accidental releases from the 100 meter (328-ft) stack, an elevated point source was assumed. The 0-2 hour relative concentration (χ/Q) at or beyond the site boundary which is exceeded 5% of the time was calculated to be 1.7×10^{-5} sec/m^3. The 0-8 hour χ/Q at the LPZ which is exceeded 5% of the time was calculated to be 8.1×10^{-6} sec/m^3. The estimated relative concentration for the 8-24 hour period was 2.5×10^{-6} sec/m^3, for the 1-4 day period was 8.6×10^{-7} sec/m^3 and for the 4-30 day period was 2.4×10^{-7} sec/m^3.

The applicant's relative concentration estimates are generally less conservative by a factor of two to four than those calculated by the staff. These differences can be attributed to the use of different meteorological parameters in determining atmospheric dispersion conditions.

2.3.5 Long Term (Routine) Diffusion Estimates

Computations of annual average offsite relative concentrations for stack releases considering plume rise as a function of wind speed showed a maximum value of 1.1×10^{-7} sec/m^3 north of the stack at a distance of 1 mile. The highest offsite annual average relative concentration of 7.2×10^{-6} sec/m^3 for vent releases occurred at the site boundary west-southwest of the reactor building.

The applicant's relative concentration estimate in the case of the stack release was less conservative by a factor of two than that calculated by the staff. For vent releases, the applicant's relative concentration estimate was twice as conservative as that calculated by the staff. In this case, the applicant did not apply a correction factor for the wake effect of the building when making the calculation.

2.3.6 Conclusions

The opinion of the staff is that the onsite meteorological data presented in the FSAR and verified by the applicant indicate that atmospheric dispersion conditions at the plant site are much less favorable than would normally be expected in this part of the country. Therefore, the staff concludes that the relative concentration estimates are very conservative.

2.4 Hydrologic Engineering

2.4.1 Hydrologic Description

The Duane Arnold Energy Center is located on the west bank of Cedar River in Linn County, Iowa, about 8 miles northwest (and about 15 river miles upstream) of Cedar Rapids. The Cedar River is the largest tributary of the Iowa River and the confluence of the two rivers is about 100 river miles downstream of the plant. The Cedar River has a total drainage area (watershed) of about 7819 square miles, of which 6250 square miles is upstream of the plant. Basin topography is characteristic of the central Iowa farm country. The Cedar River floodplain is of variable geometry, ranging from fairly narrow and moderately steep valley slopes to floodplains three to four miles wide with relatively flat slopes.

There are 12 low-head dams within the basin used primarily for hydro-electric generation or for thermal cooling of non-nuclear power plants. In addition, there are four natural and five manmade lakes in the upstream subbasins that are used primarily for recreation.

Makeup water for the DAEC mechanical draft cooling towers is obtained from the Cedar River. A Category I seismic-design weir[1]

(1) See paragraph 3.7.2 for information on Category I seismic design structures.

has been constructed across the river to direct river flow to the plant intake structure. The applicant states that the minimum water intake required for the emergency cooling water system is 13 cubic feet per second (cfs).

2.4.2 Floods

2.4.2.1 Flood History

The maximum flood of record at the Cedar Rapids, Iowa streamflow gage (about 15 river miles downstream of the plant) occurred on March 21, 1961 and produced a peak river flow of about 73,000 cfs. The applicant has estimated that this flood reached a peak water surface elevation at the plant of 746.5 feet mean sea level (MSL).

2.4.2.2 Flood Design Considerations

Finished plant grade is 757.0 feet MSL. Based on the applicant's estimate of the probable maximum flood (PMF) level, with coincident wind waves, protection had been provided to elevation 769.0 ft using stop logs at the accesses to safety-related buildings. Since the staff's conservative estimate of wind waves differed from the applicant's (see paragraph 2.4.3), the staff requested the applicant to provide flood protection to elevation 770.5 feet MSL on the northerly side of safety-related buildings; to 773.7 on the southerly side of safety-related buildings; and to 769 feet MSL on all other sides of safety-related buildings. The applicant has agreed to provide the flood protection requested by the staff.

The applicant states that severe rainfall capable of producing a local PMF will exceed the capacity of the site drainage system, but will have no adverse effect on safety-related buildings. Further, the applicant reports all safety-related buildings can support water accumulations up to the top of the roof parapets without failure. Roof penetrations extend higher than the parapets on all safety-related buildings.

2.4.3 Probable Maximum Flood (PMF) On Streams and Rivers

The depth-area-duration relationship of the probable maximum precipitation (PMP) was developed by standard transposition and maximization techniques as suggested by the Hydrometeorological Branch of NOAA. Subsequent to these analyses, the selected PMF was compared with estimates contained in a report entitled, "Probable Maximum Precipitation for the Minnesota River Basin",[10] January 1969 by the Hydrometeorological Branch of the U.S. Weather Bureau (now NOAA), and indicated the applicant's estimates of the PMP is conservative.

Precipitation losses were estimated at 1.5 inches of initial loss and 0.1 inches per hour for infiltration. The applicant developed unit hydrographs (runoff models of subbasins) using the river flow records of significant floods in the Cedar River Basin and standard techniques similar to those prescribed by the Corps of Engineers in EM 1110-2-1405,[11] "Flood Hydrology Analyses and Computations". The

PMF is estimated to produce a peak discharge of 316,000 cfs and a peak stage at the plant of 764.1 feet MSL.

The staff has independently reviewed the development of the PMF and compared it with other detailed PMF studies in the general region. The peak discharge appears to be a conservative estimate of the area PMF. The stage estimates made by the applicant used standard water surface elevation modeling techniques.

Upstream dams are low-head facilities which would be "drowned-out" by the PMF. The applicant states, and the staff concurs, that these dams present no threat of flooding to the plant.

Wind wave effects assumed to occur coincidentally with the PMF were initially based on a wind speed of 30 miles an hour (mph). At the staff's request the applicant estimated the wind wave effect of a coincident 45 mile per hour over-water wind speed. This wind speed conforms with criteria established by the staff and is applied by the staff uniformly to all such situations. However, the staff does not agree with the wave height and runup estimates provided by the applicant. The staff's conservative independent analyses using standard techniques result in the requirements stated in paragraph 2.4.2.2. As noted previously, the applicant has agreed to provide flood protection to the elevations determined by the staff.

2.4.4 Potential Dam Failures (Seismically Induced)

The applicant stated, and the staff agrees, that failure of upstream dams, because of their low-head design (the nearest upstream dam has a gross head of 11 feet), would produce water surface elevations at the plant substantially less than the PMF.

2.4.5 Probable Maximum Surge and Seiche Flooding

Not applicable to the DAEC site.

2.4.6 Probable Maximum Tsunami Flooding

Not applicable to the DAEC site.

2.4.7 Ice Flooding

There is some evidence of ice affecting river flow. At the staff's request, the applicant investigated the potential for ice flooding. The applicant states, and the staff concurs, that ice jams occur primarily during periods of low flow and, therefore, it may be concluded that flooding from downstream jams would produce water surface elevations less than the PMF. Upstream blockage could occur, producing a decrease in river flow. However, such blockages are generally short-lived, generally being overtopped and/or broken up in a short time. In addition, such jams are not water tight and the applicant states that the flow at the site from upstream jams can be expected to be greater than the requirement for makeup water.

2.4.8 Cooling Water Canals and Reservoirs

The plant intake structure is located on the bank of the Cedar River and is not dependent on canals or reservoirs for water supply.

2.4.9 Channel Diversions

The channel and flood plain configuration in the vicinity of the plant is not believed to be conducive to channel diversions.

However, should such an event occur, which is most likely when occurring coincident with a flood approaching the severity of the PMF or with a severe earthquake, some water would be trapped in the remaining channel segment. This trapped water supply, augmented by ground water return flow, is considered to provide sufficient water to maintain the plant in a safe shutdown condition until auxiliary supplies become available.

2.4.10 Flooding Protection Requirements

At the staff's request, the applicant has agreed to provide flood protection measures to the elevations discussed in Section 2.4.2.2. The proposed protection consists of stop-logs to be placed at the exterior accesses to all safety-related structures. At the staff's request, these measures will be augmented with plastic sheeting to be held in place with sand bags to reduce inleakage. However, these measures are not known to be 100% effective and, therefore, the staff has required a technical specification to require shutdown of the plant when flood water elevations exceed plant grade. The technical specification is more fully discussed in paragraph 2.4.14.

2.4.11 Low Water Considerations

Both ice jams (discussed in paragraph 2.4.7) and droughts may produce low flow at the plant. Of the two, the applicant considers droughts to produce the most critical condition because of their relatively long periods of low flow, while ice jams are more transitory in nature. The applicant has stated that the historical minimum daily flow was 236 cfs, on July 4, 1934; the minimum instantaneous low flow was 178 cfs on September 25, 1935. The applicant provided a flow duration analysis which indicates the Cedar River flow exceeds 6,600 cfs about 10% of the time and 620 cfs about 90% of the time. The applicant has provided a frequency analysis which projects a 1,000-year minimum daily flow of 120 cfs. While the accuracy of extrapolation of rare events to such a degree is questionable, it should be noted that this estimate, as well as historical low flows, are about a decade greater than the plant safe shutdown requirement of 13 cfs. We conclude the plant will have an adequate safety-related supply of water available from the Cedar River.

2.4.12 Environmental Acceptance of Effluents

The staff performed an independent analysis of the effects of accidental releases of liquids containing radionuclides, utilizing estimates of the maximum (rather than average) permeability of the near-station surficial soils. These permeabilities, combined with other conservative estimates of surface and subsurface flow, indicate a minimum effluent dilution of 1:1500. It should be noted, however, that this estimate is based on present ground and surface water usage and the applicant must be alert to future regional water supply uses that could be affected by accidental releases (see also paragraph 15.4).

2.4.13 Groundwater

The applicant states that two aquifers underlie most of the plant site. The upper unconfined aquifer is composed of fine-to-medium sands and is separated from the lower aquifer by a 10-60 foot aquiclude of relatively impervious clayey material. The lower artesian-type aquifer is under pressure and any groundwater transfer would be from the lower into the upper aquifer. The applicant has collected data on surrounding wells, and states that the groundwater gradient is fairly steep with flows generally southeasterly under the site towards the Cedar River. Presently, there are no wells down-gradient between the plant and the river. Flooding in the Cedar River may cause some flow reversal of groundwater, but this effect should be transitory in nature. The nearest significant downstream groundwater user is Cedar Rapids which draws water from wells located near the Cedar River. A large portion of the water drawn from these wells is recharged from the river.

2.4.14 Technical Specifications for Emergency Operation Requirements

For reasons previously discussed, the staff will require a technical specification requiring shutdown and cooldown of the plant when severe river flood water elevations exceed plant grade. The applicant is expected to propose a proviso allowing them to request relief from this technical specification from the AEC if there is an extreme local requirement to keep the DAEC on line. The staff has no objections to the anticipated proviso, providing it is clear that in the absence of the granting of such relief, for whatever reason, the plant will be shutdown when the river water level exceeds plant grade.

2.4.15 Conclusions

The staff concludes that the site and the design of the safety-related facilities for the Duane Arnold Energy Center will provide protection from less severe floods, but that the less frequent river produced flooding (up to and including the probable maximum flood) will require implementation of flood emergency procedures. In addition, the site and design of safety-related features will provide protection against locally severe rainstorms. However, because of the nature of protection from the more severe river produced flood levels, a technical specification is required for plant shutdown and maintenance thereof, as discussed in paragraph 2.4.14. The staff concludes that potential dam failures will not produce water levels in excess of the PMF, that an adequate water supply will exist even in the unlikely event of a river channel diversion, and that there is little likelihood of contaminating any existing public surface of ground water supply.

2.5 Geology and Seismology

2.5.1 Basic Geologic and Seismic Information

The site is located in the Interior Lowlands Tectonic Province of the Central Stable Region of North America. There are no known earthquake epicenters within 75 miles of the site. The major and even the moderate earthquake regions are sufficiently far from the site to have only minor seismic influence. There are no known geologic structures that could be expected to localize seismicity near the site. No known faults exist within the basement rock or overlying strata in the vicinity of the site. Two minor faults nearby have been postulated based on well log data. The closest of these is 10 miles north of the site and the other is 17 miles southeast. Evidence shows that these faults have been inactive at least since before Pleistocene time (more than 500,000 years ago) and possibly since Paleozoic time (two hundred million years ago).

The critical structures and equipment, which are designated Category I, are designed to respond elastically with no loss of function to the ground accelerations postulated for the Operating Basis Earthquakes (OBE). These same Category I structures and equipment are also designed so that the plant can be shutdown safely even if subjected to a postulated Design Basis Earthquake (DBE) with ground accelerations that are double the OBE values.

The applicant used one set of seismic design foundation level accelerations for structures founded on rock and another set for structures founded on soil. For the reactor containment building, which will be supported on bedrock or a lean concrete fill, the Operating Basis Earthquake (OBE) horizontal acceleration of 0.06 g was used with vertical accelerations taken as 80 percent of the horizontal accelerations. All other Category I structures are supported at grade or on 30 to 50 feet of competent soil or fill. For these structures a horizontal acceleration of 0.09 g was used for the OBE with vertical accelerations taken as two thirds of the horizontal.

We and our consultants reviewed the geology and seismology of the site at the construction permit stage of our review. No new developments have occurred since that time to change our previous conclusion on the acceptability of these characteristics for this site. The report on site seismicity for the Duane Arnold Energy Center prepared during the construction permit review by our consultant, the Seismology Division of the U.S. Coast and Geodetic Survey, is attached as Appendix C.

2.5.2 Stability of Subsurface Materials

The field investigation performed by the applicant to study the bedrock conditions in the plant area revealed varying degrees of solution activity in the limestones and dolomites underlying the site. The solution activity ranged from the formation of very small cavities to one about 12 feet in diameter. Borings were made under all Category I structures. All bore holes and the cavities revealed by them have been cleaned and filled with grout under pressure using procedures which we have reviewed. During the construction permit (CP) stage of review, the applicant performed an analysis which showed that any undetected cavities will not affect the support of the structures. Also at the CP stage of review, we and our consultant, N. M. Newmark Consulting Engineering Services, reviewed the results of the exploratory drilling, remedial treatment, and stress analysis programs and concluded that the Category I structures will be adequately supported. There have been no new developments that would change this conclusion.

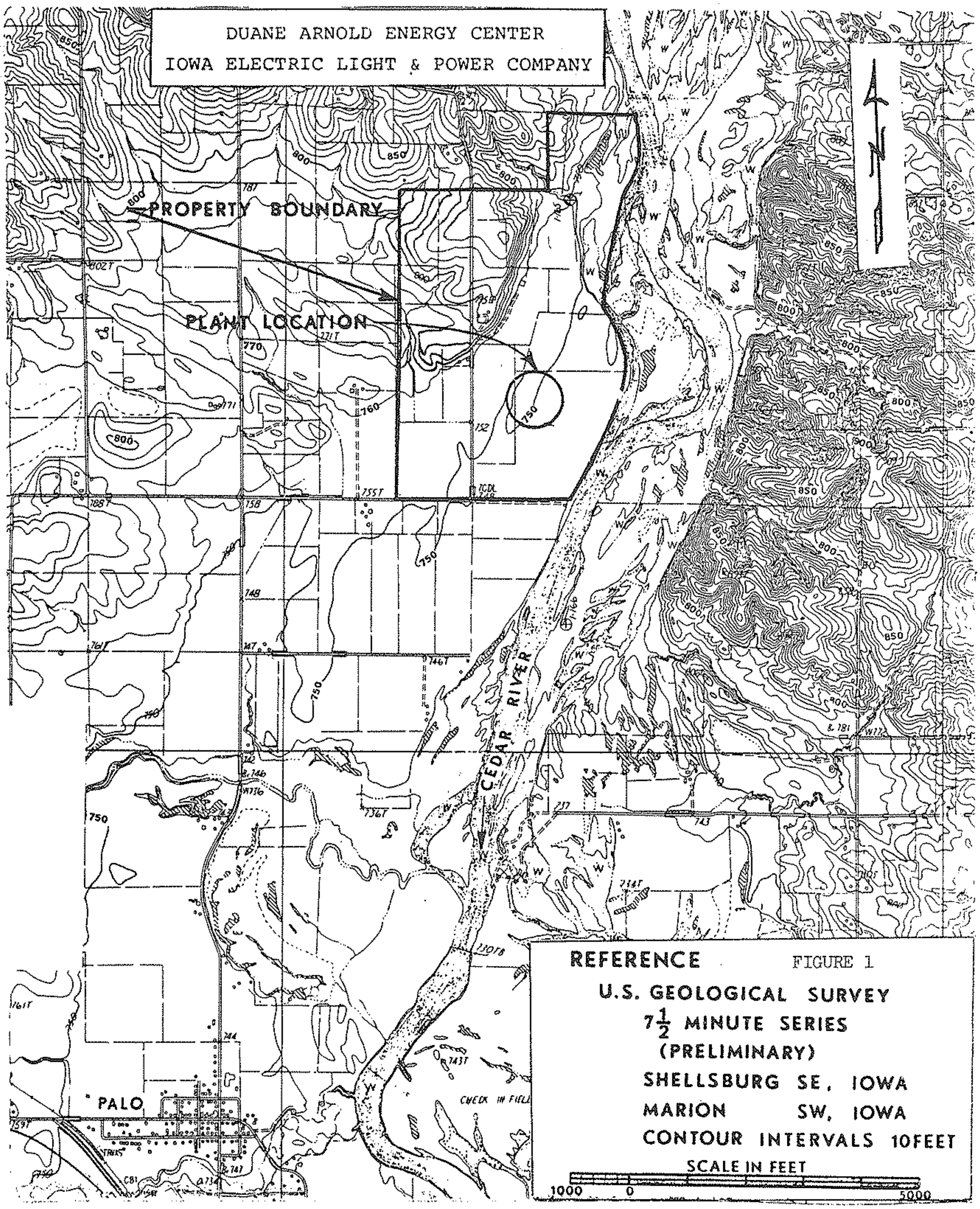
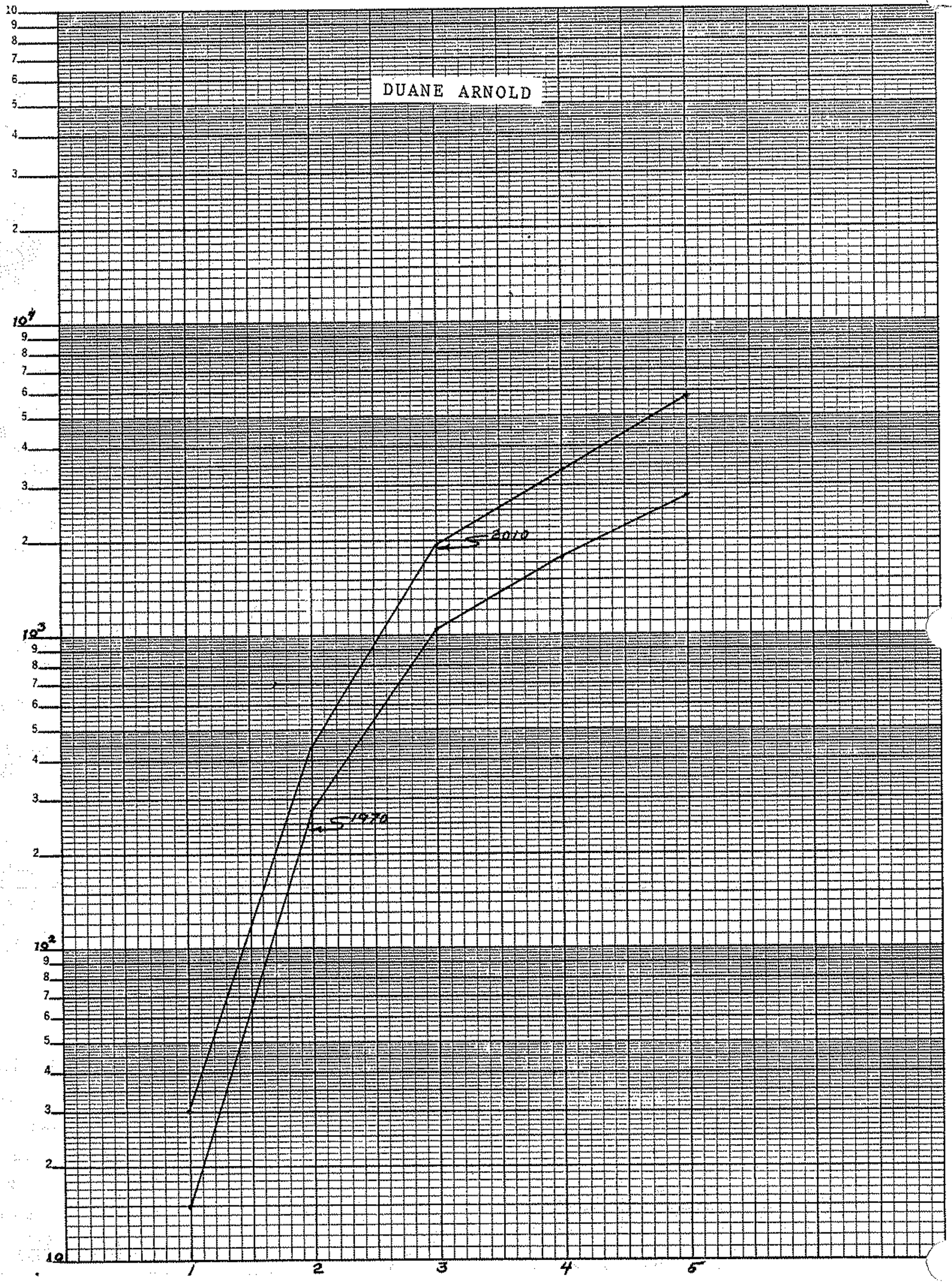
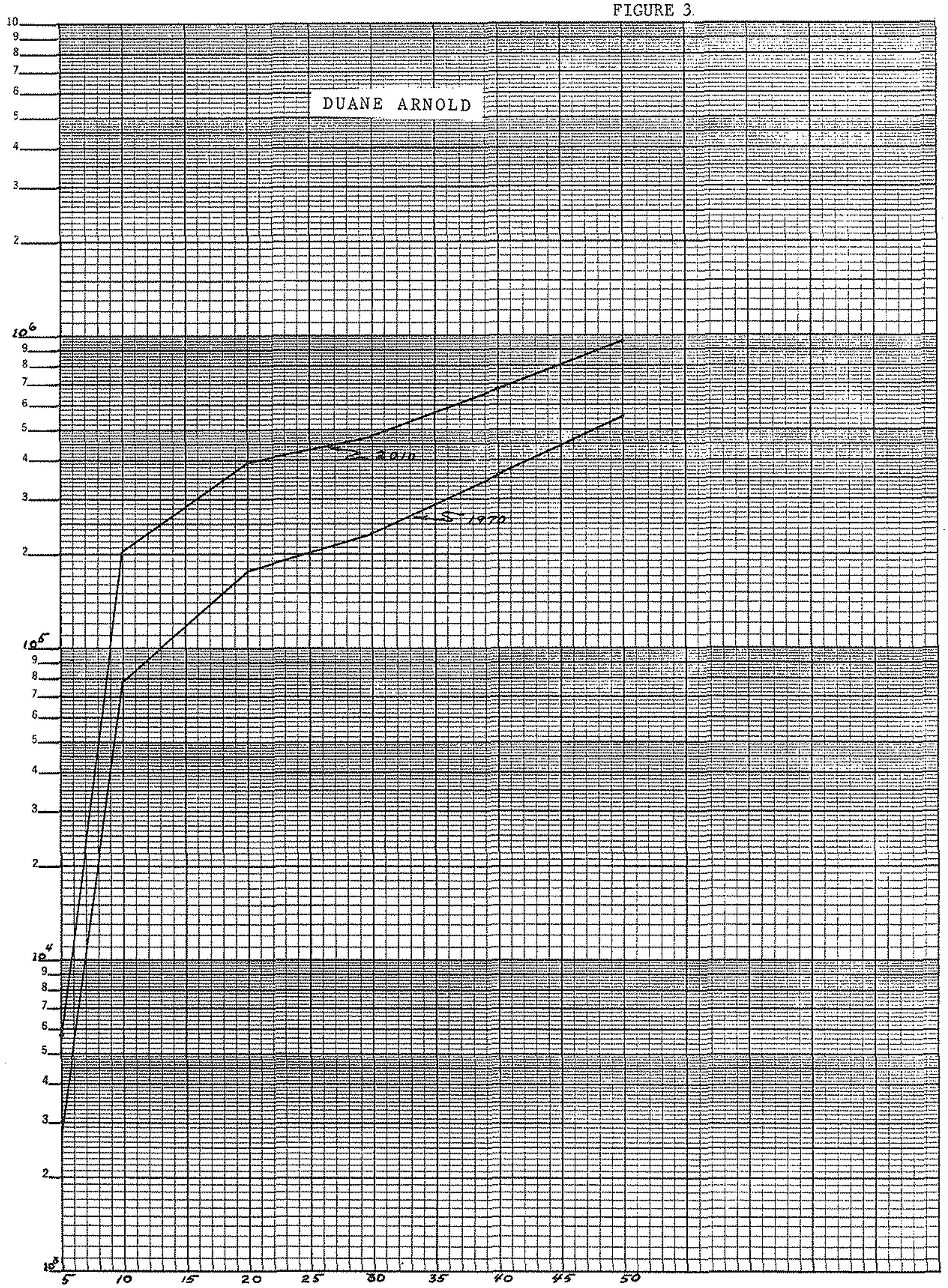

DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
PROPERTY BOUNDARY
PLANT LOCATION
CEDAR RIVER
PALO
REFERENCE
U.S. GEOLOGICAL SURVEY
7 1/2 MINUTE SERIES
(PRELIMINARY)
SHELLSBURG SE, IOWA
MARION SW, IOWA
CONTOUR INTERVALS 10FEET
SCALE IN FEET
1000 0 5000
FIGURE 1

FIGURE 2



DISTANCE, (MILES)

FIGURE 3



DISTANCE, (MILES)

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 Conformance with AEC General Design Criteria

At the construction permit stage of the review we evaluated and found satisfactory the applicant's conformance with the then available July 11, 1967 version of the AEC General Design Criteria (GDC). Subsequently, the GDC were revised. The applicant presents in Appendix F and in Amendment No. 5 of the FSAR an evaluation of the design basis of the Duane Arnold Nuclear Facility measured against the GDC of 10 CFR Part 50[7] effective May 21, 1971 and subsequently amended July 7, 1971. This version of the GDC is currently being used for all our evaluations of applications. We have reviewed the applicant's assessment of his conformance to the GDC and the stated DAEC design criteria and we are satisfied that the applicant has met the intent of the July 7, 1971 version of the General Design Criteria.

3.2 Classification of Structures, Components, and Systems

3.2.1 Seismic Classification

The applicant has identified iin Section 12 of the FSAR those Category I plant features, i.e., structures, systems and components important to safety that are designed to withstand the effects of the Safe Shutdown Earthquake and remain functional. These plant features are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shutdown the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100. Category I corresponds to Seismic Class I and the Safe Shutdown Earthquake corresponds to the Design Basis Earthquake (see paragraph 3.7.2 for further discussion of definitions).

All other structures, systems and components that may be required for operation of the facility but not classified as Category I (Seismic Class I) are Seismic Category II (Seismic Class II). Included in Seismic Category II are those portions of Category I systems which are not required to perform a safety function.

We have reviewed the seismic classification of the structures, systems and components set forth in Table 12.3-2 of the FSAR and have concluded that those items classified as Category I for this facility are acceptable.

3.2.2 System Quality Group Classification

The AEC Quality Group Classification System in Safety Guide 26[12] has been applied to those water and steam containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety where reliance is placed on these systems: (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and maintenance in the safe shutdown condition, and (3) to contain radioactive material. The applicant has identified in Appendix A of the FSAR those fluid systems or portions of fluid systems important to safety and the industry codes, standards and supplementary criteria applicable to each pressure-containing component in the systems.

For those fluid systems set forth in Table A.2-3 of the FSAR, we and the applicant are in general agreement on the application of the Quality Group Classification System. Piping and Instrumentation Diagrams identify the boundary limits of each classification group within the fluid systems.

We find that the system quality group classifications as specified by the applicant are acceptable.

3.3 Wind and Tornado Criteria

The design wind velocity for the Seismic Category I structures is 105 mph at 50 feet above ground based on a recurrence interval of 100 years. The reactor building, the control room, diesel generator building, and the intake structure were designed to protect the equipment and components which require tornado protection. The design tornado for these structures is a 300 mph rotational velocity at the periphery and a translational velocity of 60 mph. The simultaneous atmospheric pressure drop is 3 psi for a duration of 3 seconds. Some of the compartments for the Seismic Class I structures were designed for the differential pressure from venting.

The technique described in ASCE Paper No. 3269[13] was utilized to determine the loads resulting from these wind and tornado effects. The load factor associated with the wind is 1.25. For the tornado loads a load factor of 1.0 was used.

We believe that the above load factors are consistent with those used for previously approved plants and the methods of converting wind and tornado velocities into forces on the structures are in accordance with the state-of-the-art. The wind and tornado criteria are acceptable.

3.4 Water Level (Flood) Design Criteria

The finished plant grade is at elevation 757.0 feet. The facility was designed during the construction permit period of review to resist flood waters to an elevation of 767.0 feet, an elevation which was arrived at considering the maximum probable flood as well as the effects of the wind. Further review of the wave action and runup caused by winds have resulted in a new requirement accepted by the applicant for additional flood protection. The details are discussed in paragraphs 2.4.2.2, 2.4.14, and 2.4.15 of this evaluation report.

The bouyant forces created by normal ground water and during flooding were both considered in the design of Category I seismic design structures with load factors of 1.2.

We find the water level design criteria are acceptable.

3.5 Missile Protection Criteria

The consideration of tornado generated missiles included a spectrum of possible items that could be dislodged during tornadic winds and become missiles. The applicant's consideration of missiles included a 4" x 12" x 12' wooden plank traveling end-on at 300 mph and an automobile weighing two tons with a contact area of 20 sq. ft. traveling not more than 25 feet off the ground at 50 mph.

In addition, the reactor building walls, floor slabs, and the control room were designed to withstand the loads imposed by missiles generated by the failure of the turbine-generator.

We find that the missile protection criteria proposed by the applicant are adequate on the basis that they have been used on previous plants and represent the present state of knowledge in providing an acceptable means of damage assessment.

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

Nonlinear time-response dynamic analyses of the main steam and recirculation lines were performed by the applicant to verify the design adequacy of existing piping restraints including the effects of a gap between the piping and its restraint. Discrete mass-spring mathematical models were used. Both longitudinal and circumferential breaks were considered. The criteria used to determine the break locations and orientations is to locate them such that they cause maximum loads at the restraint. The stresses at the break locations do meet the limit of 2 S_m which is consistent with the staff position regarding break locations. Forcing functions representing the time variation of blowdown loads were provided. Two types of piping restraints were designed and installed. The allowable stress for the design of restraints is less than one-half the ultimate uniform strain and thereby meets the currently acceptable criteria.

We find this approach for protection against pipe whip to be acceptable for the Duane Arnold plant.

3.7 Seismic Design

3.7.1 Seismic Input

The seismic design response spectra curves were presented in the applicant's PSAR and approved by the AEC prior to the issuance of the construction permit for the Duane Arnold Nuclear Plant. The modified earthquake time histories used for component equipment design are adjusted in amplitude and frequency to envelope the response spectra specified for the site. We conclude that the seismic input criteria proposed by the applicant provides an acceptable basis for seismic design.

3.7.2 Seismic System and Subsystem Analyses

Modal response spectrum multi-degree-of-freedom and normal mode-time history methods were used for all major Category I[1]

(1) The use of "Category I" seismic design and "Class I" seismic design as descriptive phrases for structures, systems or components means the same in both instances. The most recent and recommended wording is "Category I" seismic design; this wording conforms with that which appears in Safety Guide 29. The structures, systems, or components designed to remain functional if a Safe Shutdown Earthquake (equivalent to a Design Basis Earthquake) occurs are Category I.

structures, systems and components. Governing response parameters were combined by the square root of the sum of the squares method to obtain the modal maximums when the modal response spectrum method was used. The absolute sum of responses was used for in-phase closely-spaced frequencies. Floor spectra inputs used for design and test verification of structures, systems and components were generated from the normal mode-time history method. A vertical seismic system dynamic analysis was employed for all structures, systems and components. In order to obtain the most conservative resultant value for combining horizontal and vertical responses, the applicant was required to either select two horizontal and one vertical component responses which are then combined by the square root of the sum of the squares method or, alternately, determine the absolute sum of the responses due to one horizontal component and one vertical component. The representative highest stressed regions of the structures, equipment and components must then be checked, using one of the above combination of vertical and horizontal responses to verify adequacy of the seismic design.

We conclude that the seismic system dynamic analysis methods and procedures proposed and used by the applicant and considering ultimate fulfillment of the above requirement provide an acceptable basis for the seismic design.

3.7.3 Criteria for Seismic Instrumentation Program

The type, number, location and utilization of strong motion accelerographs to record seismic events and to provide data on the frequency, amplitude and phase relationship of the seismic response of the containment structure corresponds to the recommendations of Safety Guide 12.[14]

Seismic instrumentation will be installed on Category I structures, systems and components in order to provide data for the verification of the seismic responses determined analytically during the design analysis for such Category I items.

We conclude the seismic instrumentation provided and its utilization are acceptable.

3.8 Design of Category I Structures

The staff's review and evaluation of the Category I (seismic design) structures included the structural foundations, the reactor building, the control building, intake structure, a portion of the pumphouse, a portion of the turbine building and the offgas stack.

The Category I structures were built from a composite of structural steel and reinforced concrete members. In general, the structures were designed as continuous systems such as the reactor building. The various structural components that were integrated into the continuous structures consist of slabs, walls, beams, and columns.

The analyses were based on elastic analysis procedures with the design being executed using the working stress design method. The design method for reinforced concrete followed that of ACI 318-63[15] for ultimate strength design with the use of specific loading combinations applicable to nuclear power plant design conditions. For the structural steel the AISC Specifications[16] were utilized.

The loading combinations used for the design of the structures included normal dead and live loads, wind and tornado loads, the flood loads, the missile loads and the earthquake loads.

The applicant has specified and utilized numerous loading combinations for the normal loading conditions as well as for the severe loading conditions that include the accident, the tornado and/or the design basis earthquake.

For the reinforced concrete structures additional specific requirements were set forth for ductile moment resisting frames as well as structural elements resisting mainly earthquake loads. The applicant's design for the reinforced concrete structures allowed the reinforcing steel under the worst design conditions to reach $0.90f_y$ with the concrete stress not exceeding $o.85f'_c$.

For the structural steel under normal operation the allowable stress was used as the limit with a 25% increase allowed under the operating basis earthquake and a 33% increase allowed under the wind loads. The elements that carry mainly earthquake forces used only the allowable stress as a limiting value. Under the more severe loads the steel stresses were allowed to reach $0.9f_y$.

As a result of our review and evaluation of the applicant's criteria and the procedures related to design and construction, we find that the Category I seismic design structures have been adequately designed.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

Preoperational vibration testing for prototype reactor internals will be performed in accordance with Safety Guide 20.[17] However, the surface inspection will be conducted after cold flow testing with all core support structures in place. For a group of similar components, those exhibiting the greatest response will be inspected. Measurements of the response will be made in both cold and hot flow testing to verify that they are compatible in magnitude and frequency content. We find the above program of preoperational vibration testing to be acceptable.

For assuring the design adequacy of reactor internals subjected to dynamic effects that may result from a postulated loss-of-coolant accident, responses were computed by the applicant using postulated blowdown flow modes anda dynamic model of the reactor internals. The results indicate that the response level of internals are low for the following two reasons: (1) a BWR is a two-phase steam-water system that operates at or close to saturation conditions so that the loading is not a shock type load and (2) the normal frequencies of internal components are separated by more than a factor of 10 from the loading frequencies. Thus, no severe amplification of vibration will occur. We find the above analytical results to be acceptable.

A piping vibration test program will be conducted by the applicant to verify the design adequacy of Category I seismic design piping and piping restraints to withstand hydrodynamic transients. Portable IRD accelerometers will be used for measuring the vibration amplitudes on the piping. The allowable displacements are defined by the applicant as those which produce stresses less than one half of the endurance limit. We find this approach to be acceptable.

The applicant has conducted either testing or analysis for each item of equipment to assure proper functioning of the Category I mechanical equipment during a seismic event. Additional information describing the tests and/or analysis for each type of equipment as reviewed by the staff in meetings with the applicant will be documented in Amendment 12. We find this commitment to be acceptable based on similarity of the mechanical equipment to that used in previously reviewed plants.

3.9.2 ASME Code Class 2 and 3 Components

Category I seismic design systems, components and equipment have been constructed, as applicable, to Sections III, VIII and IX of the ASME Boiler and Pressure Vessel Code;[18] USAS B 31.1.0[19] - 1967, B 16.25 and B 16.5; and appropriate standards of the Tubular Exchanger Manufacturer's Association (TEMA), Manufacturer's

Standardization Society of the Valve and Fitting Industry (MSS), American Society for Testing and Materials (ASTM) and American Welding Society (AWS). We find the above standards an acceptable basis for construction.

3.10 Seismic Qualification Testing of Category I Instrumentation and Electrical Equipment

The reactor protection system, engineered safety feature circuits, and the emergency power system are designed to meet Category I seismic design criteria. The seismic requirements were verified by seismic qualifications testing and were incorporated into equipment specifications to ensure that the equipment will function properly during the postulated safe shutdown earthquake. We find the qualification testing conducted by the applicant to be acceptable.

4.0 REACTOR

4.1 General

The nuclear steam supply system includes a General Electric Company (GE) boiling water reactor (BWR) which generates steam for direct use in the steam-driven turbine generator. The design of the Duane Arnold Energy Center (DAEC) reactor is similar to the Vermont Yankee, the Brown's Ferry, and other reactors which have been evaluated by the regulatory staff at both the construction permit and operating license stages. The DAEC reactor core, containing nuclear fuel elements and control rods, is supported in a domed, cylindrical shroud inside the reactor vessel. Steam separators are mounted on the shroud dome. Two external, motor-driven recirculating pumps inject high-velocity water into 16 jet pumps which are located in the annulus between the shroud and the reactor vessel. The high velocity water from the jet nozzles entrains and imparts energy to additional water from the annular region. The combined liquid flow enters the bottom of the reactor core. This fluid becomes a steam-water mixture as it passes through and cools the reactor core. The steam emerges from the steam separators and dryers and enters four 20-inch diameter pipes leading to the turbine-generator.

Reactor power is controlled either by movement of control rods or by changing the speed of the two external recirculation pumps. Reactor power operation is terminated (reactor shutdown) by inserting control rods into the core. A standby liquid control system is provided as a backup system for reactor shutdown and operates by pumping a sodium pentaborate solution into the reactor.

4.2 Mechanical Design

4.2.1 Fuel Design

The reactor employs Zircaloy-clad fuel rods which contain slightly enriched uranium dioxide pellets. Some pellets in some of the fuel rods also contain gadolinium oxide which is used to control the neutron flux distribution. Groups of 49 fuel rods in a square array within a square Zircaloy channel box form fuel assemblies. Three types of fuel assemblies with varying distributions of U-235 enrichments and gadolinia concentrations are used. The Type I assemblies which have a low average enrichment and no gadolinia are removed from the core at the end of the first fuel cycle. The Type II assemblies each contain two rods which have gadolinia-urania pellets over their full length. The Type III assemblies each contain two rods which have gadolinia-urania pellets over their full length and two rods which have gadolinia-urania pellets over part of their length.

The design of the fuel is the same as the design of the fuel for the Browns Ferry and Peach Bottom Units 2/3 reactors which were previously reviewed and found acceptable. The fuel design is similar to the design of the fuel in currently operating reactors, but differs in that the clad thickness is greater, a hydrogen-getter material is used inside the rods, and urania-gadolinia fuel pellets are used.

The increase in clad thickness and use of a hydrogen-getter are design changes made to improve the performance of the fuel during normal operation by further reducing the potential for cladding failures and the consequent radioactive off-gas release rate. Since the increase in clad thickness has an insignificant effect on the fuel rod thermal properties, the effect on post-accident temperature transients is negligible.

Also, the thicker clad will be less subject to embrittlement and ballooning.

A detailed description of the hydrogen-getter will be submitted by the applicant in Amendment 12. Based on our discussion with the applicant, we believe that the hydrogen-getter would not react with the cladding or reduce its integrity in any other way either during normal operation or a post-accident transient. The use of the hydrogen-getter should have no effect on the safety of the reactor.

Urania-gadolinia pellets are used in other operating reactors. The differences in fuel damage limits due to the reduction in thermal conductivity and melting point of urania-gadolinia as compared with urania was evaluated during the operating license review of the Quad-Cities reactors.

The design of the fuel has been evaluated on the same basis and meets the same criterion for design as previously reviewed and accepted for other boiling water reactors. This criterion is that no fuel cladding damage should occur during normal operation or in the event of anticipated transient conditions. Fuel damage can result from overheating, excessive expansion or collapse, or corrosion of the clad. Overheating will not occur if the mode of heat transfer remains in the nucleate boiling regime. Although heat transfer effectiveness would decrease if departure from nucleate boiling occurred, the resultant increase in clad temperature would be only 500°F and would not necessarily result in failure of the clad. Therefore, a conservative damage limit is defined as the critical heat flux (CHF) at which the departure from nucleate boiling occurs. Evaluation of the CHF is discussed in the section on thermal hydraulic design.

Excessive expansion is defined as greater than 1% strain. Tests indicate that at this strain less than 5% of the cladding would be expected to fail. Expansion of the clad is caused by expansion of the fuel pellets and is a function of both fuel burnup and temperature. Therefore, a second fuel damage limit is defined as the value of linear heat generation rate, as a function of burnup, that will produce a clad strain of 1%. For rods with urania pellets, this limit is calculated to be 28, 26.5, and 24 kW/ft at burnups of zero, 20,000, and 40,000 MWd/T, respectively. For urania-gadolinia pellets, the limits are approximately 3 kW/ft less than the urania pellets limits given above.

Collapse of the cladding can occur due to the effect of densification of the fuel pellets and the creep of the clad. This phenomenon has been observed in some reactors and its causes and effects are described in the staff's "Technical Report on Densification of Light Water Reactor Fuels", which was issued November 14, 1972. Based on a preliminary evaluation, fuel cladding collapse is not expected to occur in the DAEC core. However, we have requested that the applicant evaluate whether fuel densification and clad collapse could occur. The results of this evaluation were submitted on January 9, 1973 and is currently undergoing review and evaluation for all nuclear plants by the staff. A supplement to this Safety Evaluation will be written regarding this matter on completion of the above cited review by the staff.

Corrosion of the cladding due to local formation of hydrides on the inner clad surfaces has occurred in several reactors and caused clad failures and higher than desired off-gas activity. Water vapor present in the rods after their assembly was presumed to be the cause. The fuel rod manufacturing process has been modified and a hydrogen-getter has been added in the rod as means of assuring that moisture is not present or will not contribute to internal hydriding. Although fuel clad failures may still occur, any increase in cladding failure will be detected by an increase in coolant or off-gas activity. Before the release becomes excessive, i.e., exceeds Technical Specification limiting conditions of operation, appropriate lower operational limits will require a restriction on plant operations.

4.2.2 Reactor Vessel Internals (Mechanical Design)

For normal design loads of mechanical, hydraulic, and thermal origin, including anticipated plant transients and the operational basis earthquake, the reactor internals were designed to the stress limit criteria of Article 4, of the ASME Boiler and Pressure Vessel Code (BPVC) Section III.[18]

Under design basis accident conditions, which include the combined loads from a recirculation line break or a steam line break plus the Design Basis Earthquake, the reactor internal components were designed to the criteria submitted in Appendix C of the FSAR. These criteria are consistent with comparable ASME BPVC emergency and faulted operating condition category limits and the criteria which have been accepted for all recently licensed plants. We find the applicant's criteria acceptable. The dynamic analyses of the Duane Arnold Energy Center reactor internals are discussed in paragraph 3.9.1, "Dynamic System Analysis and Testing", of this evaluation report.

4.2.3 Reactivity Control Systems

Reactor power can be controlled either by movement of control rods or variation in reactor coolant recirculation system flow rate. The fuel rods will contain full length and partial length gadolinium oxide, a burnable poison, to supplement the moveable control rods in controlling the core reactivity throughout the core life. A standby liquid control system is also provided as a backup reactor shutdown system.

Control rods (89 in number) are used to bring the reactor through the full range of power (from shutdown to full power operation), to shape the reactor power distribution, and to compensate for changes in reactivity resulting from fuel burnup. Each control rod drive has separate control and rapid insertion (scram) devices.

The drives have a common supply pump (and one parallelled spare pump) as the hydraulic pressure source for normal operation and a common discharge volume for scram operation. On the basis of our review of the drive system design and the supporting evidence accumulated from operation of similar systems in other General Electric reactors, we conclude that the installed system will meet the functional performance requirements in a safe manner.

The current plan for operation at power levels below 10% of rated power will require limiting of selected control rod reactivity worths to less than 1% $\Delta k/k$. This is accomplished by a computer program and monitoring system known as the Rod Worth Minimizer (RWM). The RWM restricts the selection and movement of rods to the properly sequenced control rod patterns such that the total worth of any in-sequence rod that can be moved will be no more than 1% $\Delta k/k$. Calculations of the consequences of a control-rod-drop accident are discussed in Section 15.0. Use of the RWM is presently under study in conjunction with our review of General Electric Topical Reports NEDO-10527[20] and its Supplement, titled "Rod Drop Accident Analysis for Large BWR's". The background and status of our review on these reports are contained in paragraph 15.2.2 of this Safety Evaluation. Also indicated elsewhere (paragraph 7.6) is the applicant's commitment to install a staff-approved system for control of rod reactivity worth and thus the consequences of a postulated control rod drop accident.

A control-rod-ejection accident, to be distinguished from the rod drop accident, is precluded by a control rod housing support structure located below the reactor pressure vessel, similar to that installed on the other large General Electric Reactors. This structure limits the distance that a ruptured control rod drive housing could be displaced. The applicant concluded, and we agree, that the control rod displacement would be so small in this event that any resulting nuclear transient could not be sufficient to cause fuel rod failure.

Reactor power can also be controlled through changes in the primary coolant recirculation flow rate. The recirculation flow control system can automatically adjust reactor power level to station load demand whenever the reactor is operating between approximately 70% and 100% rated power. The recirculation flow control system is designed to allow either manual or automatic control of reactor power. This method of reactor power control has been satisfactorily demonstrated in the Dresden Units 2 and 3, Monticello, and Millstone I facilities.

The standby liquid control system is available to pump a sodium pentaborate into the reactor vessel. This system is designed to bring the reactor to a cold shutdown condition from the full power steady-state operating condition at any time in core life, independent of the control rod system capabilities. The injection rate of the system is adequate to compensate for the effects of xenon decay.

Except for the long-term evaluation of the expected favorable performance of gadolinia as a burnable poison in the nuclear fuel and the final staff evaluation and approval of the General Electric Company's plan for the control of rod reactivity worth, we conclude that the applicant's systems for reactivity control is satisfactory. Time is available for staff and applicant conference, a final documentation, and installation, which will be required prior to fuel loading, of the device(s) needed for rod reactivity control.

4.3 Core Thermal and Hydraulic Design

The thermal and hydraulic characteristics of the Duane Arnold Energy Center are similar to those for the Brown's Ferry and Hatch I nuclear facilities. For the Duane Arnold facility, our evaluation was made on the same basis as the reviews for these other plants.

The core thermal and hydraulic design bases are formulated to limit the local power density and coolant flow within the core to values such that the fuel damage limits, as described in paragraph 4.2.1, are not exceeded during normal operation or operational transients. One damage limit is the critical heat flux. The present critical heat flux limits are calculated using the correlation reported in the GE topical report APED-5286,[21] "Design Basis for Critical Heat Flux Conditions in Boiling Water Reactors", issued in 1966. This correlation is based on experimental data taken over the range of conditions representative to BWRs. The minimum critical heat flux ratio (MCHFR) is defined as the ratio of the critical heat flux correlation value at the corresponding fluid conditions to the actual maximum calculated heat flux occurring at a given point in the fuel assembly at any time during operation, including reactor anticipated transients. A MCHFR >1.0 conservatively assures that cooling of the fuel is maintained through nucleate boiling heat transfer.

The current design basis for normal operation is that the MCHFR calculated for any point is greater than 1.9 during normal operation and greater than 1.0 during anticipated transients. These limits provide considerable margin between expected conditions and those required to cause fuel clad damage since the critical heat flux correlation presented in APED-5286[21] is conservatively based on a limit line drawn below all of the available experimental data points. The maximum linear heat generation rate reached during normal rated power operation is not expected to exceed 18.5 kW/ft, corresponding to a MCHFR of 1.9. Analysis of anticipated operational transients shows that the lowest MCHFR, a value of 1.3, occurs following a loss of all offsite power.

A second fuel damage limit is the linear heat generation rate (LHGR) which produces a clad strain of 1%. The LHGR producing a strain of 1% is more than 24 kW/ft during normal operation. The maximum LHGR that may be attained by fuel rods during steady state operation is 18.5 kW/ft. Although higher peak powers occur during anticipated operational transients, fuel temperatures and the resulting expansion are not sufficient to produce the 1% clad strain.

We have reviewed the methods used to calculate the thermal and hydraulic limits, the experimental basis for the calculations, and the applicant's analyses of normal operation and anticipated transients for this plant and previously reviewed reactors, and conclude that the design provides adequate margin to protect the core against fuel damage. This evaluation considered reactor operation under normal plant conditions at the ultimate power level of 1658 MWt. Substantiation of the applicant's analysis and prediction of performance before increase in power level from rated 1593 (MWt) to the ultimate of (1658 MWt) must be accomplished as described in Amendment 9 (response to question 1.2) and elsewhere in the safety evaluation (paragraph 1.1).

5.0 REACTOR COOLANT SYSTEM

5.1 General

The principal equipment or system items to be discussed in this section are the reactor pressure vessel, the reactor recirculation system, the main steam and feedwater lines, and the pressure relief system. These items form the major components of the reactor coolant pressure boundary (RCPB). The pressure boundary also contains portions of the cooling system, residual heat removal system and reactor water cleanup system. Portions of these systems as well as other piping that extend from the reactor vessel out to the second outermost isolation valve are considered within the RCPB.

All of the components of the reactor coolant pressure boundary were designed and built to the appropriate codes in effect at the time of order. As a result of our review of the relevant portions of the application, we have determined that the codes and code editions used by the applicant comply with the provisions of 10 CFR Part 50, Section 50.55a, "Codes and Standards."

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Design of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary is designated as a Category I seismic design system. The component codes and code cases used in its construction, as referenced in the Final Safety Analysis Report, are acceptable and in conformance with §50.55a of 10 CFR 50 and AEC Safety

Guide 26.[12] Such conformance is an acceptable basis for meeting the requirements of AEC General Design Criterion #1.

The components of the reactor coolant pressure boundary have been designed to remain within the stress limits set by the appropriate Codes when subjected to the loads calculated to result from the Design Basis Accident, the Design Basis Earthquake (the same as Safe Shutdown Earthquake), the combination of these postulated events plus the normal loads of mechanical, hydraulic, and thermal origin and anticipated transients.

Active components, i.e., pumps and valves which are required to operate reliably in order to perform a safety function such as safe shutdown of the reactor or mitigation of the consequences of a pipe break, are designed to deformation limits that are consistent with operational requirements. Under these restrictive deformation criteria, calculated primary stresses will be in the elastic range. We find the above stress and deformation criteria used by the applicant to be acceptable.

5.2.2 Pressure Relief System

The objectives of the pressure relief system are (a) to limit any overpressure of the reactor coolant pressure boundary (RCPB) that might occur from abnormal operational transients, and (b) to provide a method for rapid depressurization of the primary coolant system in the event of certain loss-of-coolant accidents. In the latter application, automatic depressurization for small breaks of the primary system enables low pressure coolant injection system (LPCIS) or core spray system (CSS) operation. This automatic depressurization system is a backup to the high pressure coolant injection system (HPCIS) described in paragraph 6.3.1 of this evaluation report.

There are six relief valves and two safety valves in the pressure relief system. The valves are mounted on the main steamlines between the reactor vessel and the first isolation valve inside primary containment. Operation of the relief valves will discharge steam to the suppression pool and will perform these functions: (a) limit overpressure and prevent spring safety valve opening, (b) augment spring safety valve capability by opening (self-actuated operation only), and (c) depressurize the primary system following small breaks to allow LPCI and/or CSS operation. The six relief valves are self-actuating in their overpressure safety mode but can also be operated indirectly to permit remote manual or automatic operation at lower pressures.

The two safety valves will discharge to the drywell interior and function to prevent overpressurization of the primary coolant system. The overpressure protection capacity is based on the pressure rise resulting from the following postulated events: (a) main steam flow stops after closure of the main steam line isolation valves (MSLIV's) with the plant operating at turbine-generator design condition, (b) vessel dome pressure of 1090 psig, (c) 105% rated steam flow, and (d) reactor thermal power of 1657 MW. An indirect reactor scram due to high reactor vessel pressure is also assumed. The analysis indicates that a design capacity for the spring safety valves of 10 percent rated steam flow in conjunction with the design dual safety/relief valves capacity of 61.9 percent rated steam flow, is capable of keeping an adequate pressure margin below the peak ASME Code allowable pressure of 1375 psig (110% of vessel design pressure) at the vessel bottom. Actual capacities for the two spring safety valves is 18.7% and for the six safety/relief valves is 68.4% of rated steam flow. This analysis and other aspects of the overpressure protection provided are found in the FSAR, Section 4.4, Appendix H (page H.4-25) and Amendment 3 (response to question H1.1).

We conclude that the pressure relief system, when supplemented by the action of the reactor protection system, provides adequate protection against overpressurization of the reactor coolant system.

Notwithstanding, problems with pressure relief valves have been experienced in operating BWR's. These problems, e.g., inadvertent opening of valves during certain transients, are being reviewed by consultants and the regulatory staff to determine the cause and to recommend a solution to prevent their recurrence.

5.2.3 Reactor Vessel Material Surveillance Program

A material surveillance program is required to monitor changes in the fracture toughness properties of the reactor vessel beltline material induced by neutron irradiation.

The applicant has shown in the FSAR that the proposed material surveillance program complies with the Commission's proposed regulation 50.55a, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," and is consistent with programs that have been found acceptable for other similar BWR plants. The program is acceptable with respect to the number of capsules, number and type of specimens, withdrawal schedule, and retention of archive material. We have concluded that the proposed program will adequately monitor neutron radiation-induced changes in the fracture toughness of the reactor vessel material.

5.2.4 Fracture Toughness

To assure compliance with the safety and design criteria, ferritic materials in pressure retaining components of the reactor coolant pressure boundary must exhibit adequate fracture toughness properties under normal reactor operating conditions, system hydrostatic tests, and during transient conditions to which the system may be subjected.

We have reviewed materials testing and the operating limitations proposed by the applicant. The applicant has stated in the FSAR and in Amendment No. 3 thereto that acceptance testing for ferritic materials was performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III (1965 Edition, including Addenda through Summer 1967). Dropweight NDT data has been obtained for the reactor vessel beltline plate material.

In establishing the operating pressure and temperature limitations during heatup, cooldown and inservice hydrostatic tests of the system, the applicant has agreed to follow the recommendations of Appendix G, "Protection Against Non-Ductile Failure," of the 1972 Summer Addenda of the ASME Code, Section III.

The applicant will submit specific heatup, cooldown and hydrostatic test limitation curves, which meet the current fracture toughness requirement, for our use in the final issue of the Technical Specifications.

We conclude that in view of the fracture toughness testing performed by the applicant and the planned operation of the DAEC reactor coolant system, adequate margins of safety is assured.

5.2.5 Sensitized Stainless Steel

Sensitized austenitic stainless steels exhibit increased susceptibility to stress corrosion cracking when used extensively in contact with the primary coolant such as in piping, valves, pumps, pressure vessel linings and supporting hardware.

The applicant has stated in the FSAR that all sensitized austenitic stainless steel has been replaced on the Duane Arnold Energy Center pressure vessel except the jet pump riser brace pads and recirculation inlet thermal sleeve attachment buildups. These exceptions are fabricated from weld metal with controlled ferrite content (at least 5 percent) to avoid significant sensitization.

Preheat, heat-input and interpass temperatures during welding operations are controlled to avoid local sensitization of stainless steel.

We conclude that the applicant's planning and efforts to avoid sensitization of austenitic stainless steel is acceptable.

5.2.6 Leakage Detection and Testing

5.2.6.1 Leakage Detection Systems for the RCPB

Coolant leakage within the reactor containment may be an indication of a small through-wall flaw in the reactor coolant pressure boundary (RCPB). The leakage detection system proposed for the reactor coolant pressure boundary is described in the FSAR. The system, which includes diverse leak detection methods, will have sufficient sensitivity to measure small leaks, and will have provisions for suitable control room alarms and readout. The major components of the leakage detection system are containment atmosphere particulate and gaseous radioactivity monitors and level indicators at the containment sump. Indirect indication of leakage can be obtained from the drywell humidity, pressure and temperature indicators. We conclude that the proposed leakage detection system has the capability to detect small through-wall flaws in the reactor coolant pressure boundary and that the system is acceptable.

5.2.6.2 Leakage Testing Programs for Containment

Leakage testing of the reactor primary containment and associated systems is intended to provide initial and periodic verification of the leaktight integrity of the containment.

The applicant has stated in the FSAR that the primary reactor containment and its components will be designed so that periodic integrated leakage rate testing can be conducted at a test pressure corresponding to the calculated peak accident pressure.

Penetrations, including personnel and equipment hatches and airlocks, and isolation valves, have been designed with the capability for performing individual leak tests at the calculated peak accident pressure.

We conclude that the design of the containment system will permit containment leakage rate testing in accordance with the AEC proposed "Reactor Containment Leakage Testing for Water Cooled Power Reactors," § 50.54(o), Appendix J, published in the Federal Register on August 27, 1971, and the applicant's program for leak testing the containment is acceptable. The requirement for this testing is given in the Technical Specifications.

5.2.7 Inservice Inspection Program

Selected welds and weld heat-affected zones must be inspected periodically to assure continued integrity of the reactor coolant pressure boundary during the service lifetime of the plant.

The applicant has stated that the inservice inspection program for the reactor coolant pressure boundary will comply with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for In-service Inspection of Reactor Coolant Systems" 1970 Edition. Access provisions for performing inservice inspection has been considered in the design and arrangement of pressure-containing components.

The facility has been designed to allow inspection of the reactor vessel using a remotely operable inspection tool capable of performing inspections of vessel surfaces and of circumferential, longitudinal and nozzle welds.

We conclude that the access provisions and planning for inservice inspection are acceptable. The provisions of the AEC Guideline, "Inservice Inspection Requirements for Nuclear Power Plants Constructed with Limited Accessibility for Inservice Inspection," (January 31, 1969) have been satisfied.

5.2.8 Residual Heat Removal System

The Residual Heat Removal System (RHRS) is designed for four major modes of operation besides the low pressure coolant injection (LPCI) mode, which is discussed in paragraph 6.3.4. This safety-related mode of operation of the RHR System (as LPCIS) will restore and maintain coolant inventory in the reactor vessel after a loss-of-coolant accident. Another safety-related mode of operation of the RHRS provides for containment spray for condensing steam in the containment during the post-LOCA period. For normal usage, the RHR system modes of operation include removal of reactor decay heat and residual heat from the nuclear system, supplementing fuel pool cooling capability, and condensing the reactor steam so that decay heat and residual heat can be removed if the normal heat sink is not available.

The RHR system consists of two heat exchangers, four main system pumps, four RHR service water pumps, and associated valves, piping, controls and instrumentation. All functional components are designed to satisfy Category I seismic design requirements. The main system pumps are sized on the basis of flow required during the LPCI mode of operation which is the mode requiring the maximum flow rate. The service water pumps are sized to cause the pressure at the cooling water outlet of the RHRS heat exchangers to be greater than the pressure of the reactor coolant at the inlet of the heat exchangers during the shutdown cooling and steam condensing modes of operation. With this as the design criterion, heat exchanger tube leaks will not contaminate the service water with reactor coolant water.

Each loop, consisting of one heat exchanger, two RHR pumps in parallel and ancillary equipment, is physically separated from the other. However, a cross connection by a single header make it possible to supply either loop from the pumps in the other loop. Provision also exists for pumping RHR service water either directly into the containment or into the reactor if necessary. The RHRS operational modes are described briefly below.

During reactor isolation, the RHRS can be operated in the condensing mode to condense reactor steam; hence, the RHRS operates in conjunction with the reactor core isolation cooling system (RCICS). With the reactor isolated, reactor steam normally is directed to and condensed in the suppression pool via the relief valves and the RCIC turbine exhaust piping. However, the suppression pool temperature under these conditons is limited to about 130°F in order that the water temperature rise due to a postulated, subsequent design basis loss-of-coolant accident would not cause the pool temperature to exceed 170°F during the reactor blowdown. The condensing mode of RHRS operation relieves the burden on the suppression pool by transferring a portion of the decay heat; i.e., steam energy, to the RHR service water. Reactor steam is taken to the shell side of the RHRS heat exchangers and transfers heat to the service water in the tubes. The condensate is either dumped to the suppression pool or returned to the reactor vessel through the suction of the steam-turbine driven, RCIC pump. Shortly after shutdown, both heat exchangers are used to handle essentially all of the decay heat. After about 2 hours, the capacity of one heat exchanger is adequate and the other may be transferred to the suppression pool cooling mode.

The suppression pool cooling mode utilizes the RHRS heat exchangers to cool the suppression pool water by transferring heat to the RHR service water. This mode can be used in conjunction with the condensing mode or to provide long term suppression pool cooling following a loss-of-coolant accident blowdown.

The shutdown cooling and reactor vessel head spray mode is operated during normal shutdown and cooldown. Reactor water is diverted from one of the recirculation loops, through the RHRS pumps and the RHRS heat exchangers (shell side) where heat is transferred to the RHR service water (tube side); then the cooler reactor water is returned to the reactor vessel via a recirculation loop. Part of the cooled reactor water flow is diverted to a reactor head spray nozzle where it maintains saturated conditions in the vessel head volume by condensing the steam generated by the hot vessel walls and internals.

The containment spray mode of operation is initiated manually after the LPCI requirements are satisfied and aids in reducing post-LOCA drywell pressure. The RHR pumps transfer water from the suppression pool through the RHRS heat exchangers where it is cooled by the RHR service water. The cooled water enters the containment through headers and spray nozzles in the drywell and above the suppression pool and reduces the drywell pressure by condensing existing steam. The spray water will collect in the bottom of the drywell until it overflows into the drywell vent lines and drains back to the suppression pool.

We conclude that the design of the RHRS as described above is acceptable.

6.0 ENGINEERED SAFETY FEATURES

6.1 General

The engineered safety features for the Duane Arnold Energy Center include all those provided in recently reviewed and licensed boiling water reactors. The primary containment, the vapor suppression concept embodied in the vent-downcomer-torus-wetwell combination, the containment isolation capabilities, the inerting of containment atmosphere with nitrogen during normal operation, the containment atmosphere dilution in the post-LOCA period (CAD system), the standby gas treatment system (SGTS), the emergency service water system, and the emergency core cooling systems are among the systems designed and incorporated into the facility. These engineered safety features are components, equipment, structures, and systems that are designed and installed to mitigate the effects of postulated accidents, including the design basis accidents, so that there will be no undue risk to the health and safety of the public and of the plant personnel. These systems are discussed and evaluated in the paragraphs which follow.

6.2 Containment Systems

The containment systems consist of the primary containment, a secondary containment which encloses the primary containment, containment cooling systems, isolation valves, a Standby Gas Treatment System and a combustible gas control system.

6.2.1 Containment Functional Design

6.2.1.1 Primary Containment

The primary containment is a pressure suppression system consisting of the drywell, the pressure suppression chamber and a connecting vent system. The drywell houses the reactor vessel, the reactor coolant recirculation system and other branch connections of the reactor primary system.

The drywell has a steel spherical lower portion 63 feet in diameter and a steel cylindrical upper portion 32 feet in diameter. Overall height of the drywell is about 108 feet, 9 inches. The pressure suppression chamber is a steel torus located below and encircling the drywell, with a major diameter of 98 feet, 8 inches and a cross-sectional diameter of 25 feet, 8 inches. Eight (8) vent pipes lead from the drywell to a header inside the torus, and 48 downcomer pipes (24 inch diameter) project downward from the header and terminate approximately 4 feet below the surface of the torus pool. The free air volumes in the drywell and torus are approximately 109,400 ft^3 and 94,270 ft^3. The torus contains 58,900 ft^3 of water.

In the event of a design basis loss-of-coolant accident, the released steam passes through the vent pipes, torus header, and downcomer pipes into the torus water where it is condensed.

The primary containment is designed for an internal pressure of 56 psig coincident with a temperature of 281°F. In accordance with the ASME Boiler and Pressure Vessel Code, Section III, maximum drywell pressures up to 62 psig are permissible for this design. The applicant has calculated that the peak pressures that might be reached as a result of the design basis loss-of-coolant accident are 54 psig in the drywell and 25 psig in the torus. These pressures were calculated assuming a hypothetical instantaneous break of one recirculation loop pipe. The analytical methods used are the same as those used on other recently reviewed BWR plants.

We have performed our own independent analysis of the containment pressure response using the CONTEMPT-LT computer code. The peak pressures resulting from this analysis are in agreement with those calculated by the applicant. Based on the applicant's use of the General Electric NEDO-10320[22] model and our own independent verification of the analytical results, we conclude that the applicant's analysis of the short term containment response is acceptable and that the primary containment design basis is acceptable.

The primary containment is designed for an external pressure of 2 psi greater than the internal pressure. The vacuum relief system is sized to maintain the differential containment pressure to less than 2 psi.

Vacuum in the torus is relieved by two sets of valves, each set consisting of a swing check valve in series with an air operated butterfly valve, which connect the reactor building and torus atmospheres. Vacuum in the drywell is relieved by seven (7) swing check valve vacuum breakers located on the drywell-torus vent header. The torus-drywell vacuum breakers have redundant position switches which indicate on the control panel in the main control room.

6.2.1.2 Secondary Containment

The secondary containment is designed to limit the ground level release of airborne radioactive materials and provides a means for controlled elevated release of the building atmosphere so that offsite doses from a design basis fuel handling or loss-of-coolant accident will be below the guideline values stated in 10 CFR Part 100. The secondary containment system consists of the reactor building, which is discussed in this paragraph, and the Standby Gas Treatment System, which is discussed in paragraph 6.2.3.

The reactor building encloses the primary containment system, the new and spent fuel storage facilities, the core standby cooling systems, and other reactor auxiliary protection systems. The reactor building is designed to provide protection from all postulated environmental events, including tornadoes, for all systems located within the building which are required for safe shutdown of the plant.

The applicant's design criterion for reactor building leakage is to limit inleakage to 100% of the building volume per day at a negative pressure of 1/4 inch of water while the Standby Gas Treatment System is operating, under calm wind conditions. This ensures that all reactor building leakage is leakage into the building. Penetrations of the secondary containment are designed to have leakage characteristics consistent with the above cited secondary containment leak-tightness criterion.

6.2.2 Containment Heat Removal

Containment heat removal capability is provided by a drywell fan cooler system during normal operation and by the containment cooling mode of the Residual Heat Removal (RHR) System for post-accident cooling.

The drywell fan cooler system is not required for post-accident cooling. It utilizes six (6) fan coil units, each unit consisting of two cooling coils and two motor driven fans. Cooling water is supplied from the plant service water system.

The containment cooling mode of the RHR System serves to limit temperature and pressure in the drywell and torus following a loss-of-coolant accident. When operating in the containment cooling mode, the RHR pumps take suction from the suppression pool, pump water to the RHR heat exchangers and direct the cooled water either back to the suppression pool or to the drywell and suppression chamber sprays. Details of the RHR System are found in paragraph 5.2.8 of this evaluation report.

The applicant has provided analyses of the long term post-accident containment response assuming various combinations of containment cooling availability. The results of the analyses indicate that long term containment pressures and suppression pool temperatures are within allowable limits for all cases considered.

Based on our review of the DAEC system and other similar systems, we conclude that the design of the DAEC containment heat removal system is acceptable.

6.2.3 Standby Gas Treatment System

The Standby Gas Treatment System (SGTS) provides a means for minimizing the release of radioactive material from the containment to the environs by first filtering and then exhausting the atmosphere from the reactor building. Primary containment and vent exhaust can also be directed to the SGTS for processing prior to release. For all cases, elevated release is ensured by exhausting through the main offgas stack.

The SGTS consists of two identical, parallel air filtration trains, each train having 100% capacity and consisting of a demister (moisture separator), electrical heating coil, prefilter, high efficiency particulate absorber (HEPA), charcoal filter, HEPA filter, and exhaust fan. The SGTS is designed to seismic Category I criteria, including the underground discharge pipe leading to the main offgas stack. The SGTS is enclosed in a seismic Category I structure; the redundant trains are separated by a concrete wall to minimize the potential for single failure in one train causing the loss of function of the entire system.

Each exhaust fan has a 4000 cfm design flow rate which is capable of reducing and maintaining the reactor building at 1/4-inch water negative pressure under normal wind conditions. The SGTS will start automatically upon receipt of various signals or it can be manually started from the main control room. SGTS isolation valves fail in the open position on loss of electrical power or instrument air. The operation of all active components is indicated in the control room and the failure of the system to perform satisfactorily is annunciated in the main control room.

The filters will be tested to demonstrate a removal efficiency for particulates of not less than 99%. The charcoal beds will also be tested to demonstrate that their iodine removal efficiency is not less than 99%. A test program will be conducted before reactor operation and periodically during the life of the plant to demonstrate the design capability and operability of the secondary containment and SGTS.

Based on our review of the DAEC system and other similar systems, we conclude that the design and testing of the SGTS are acceptable.

6.2.4 Containment Isolation Systems

The purpose of the containment isolation system is to provide the necessary containment integrity between the primary coolant system pressure boundary or the primary containment atmosphere and the environs, in the event of accidents or equipment failures. Where necessary, valves are provided with valve operators, and these valves are automatically closed when the sensors detect certain accident or faulted conditions. The consequences of postulated pipe failures both inside and outside the containment have been evaluated.

The isolation valves and their control systems have been reviewed as a safety system to assure that no single accident or failure can result in a loss of containment integrity. This is the double barrier concept. An exception occurs in the case of the instrument lines that connect to the reactor primary coolant system, penetrate the containment, and dead-end in instrument transducers located in the reactor building. These lines have two isolation valves, both of which are outside the containment. The inboard valve nearest the containment is a hand-operated root valve. The second valve, immediately adjacent, is an excess flow check valve with open and closed position indication. A break in the portion of the instrument line between the primary containment and the excess flow check valve would result in a blowdown directly to the reactor building.

The applicant has installed orifices in each of these instrument lines inside the primary containment. The orifice size selected (1/4 inch diameter) is sufficiently small that the quantity of coolant that would be discharged from the reactor into the reactor building in the event of a rupture of an instrument line would not result in a loss of the secondary containment function; also, if the reactor building is isolated, the operation of one standby gas treatment filter train will prevent the pressure in the reactor building from exceeding its design value. Based on our review of the system, we conclude that the isolation valves and the instrument line orifices are adequately designed and meet the intent of the Supplement to Safety Guide 11.[23]

The isolation valves for the feedwater lines which penetrate the primary containment do not meet strictly all the provisions of Criterion 55 of the General Design Criteria given in Appendix A to 10 CFR Part 50 in that the feedwater isolation valves are simple check valves. This Appendix was adopted subsequent to our issuance of the construction permit for DAEC. Each feedwater line has two isolation valves; there is a check valve located inside and a stop check globe valve (motor operated) outside the primary containment. The feedwater lines are also used by the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems; these two systems connected to the feedwater lines are the only high pressure coolant injection systems available for core cooling in addition to the normal feedwater system. These high-pressure system lines have two (2) isolation valves in series. The first is a check valve and the other is a motor-operated, automatic and remote-manually actuated valve. The use of check valves outside the primary containment on influent lines that maintain reactor coolant makeup from all sources of supply, has been accepted on previous plants because of the requirement for an assured capability to permit high pressure and low pressure coolant injection into the reactor vessel. On this basis we consider that the intent of Criterion 55 is met with the design, as described, of the isolating systems for the feedwater lines.

In cases where two (2) check valves in series provide for isolation of the containment, there is the capability to functionally test and leak check these valves. Automatic isolation valves are not used in these cases since this would introduce a potential failure mechanism that would not permit injection of makeup or cooling water to the reactor vessel.

We have reviewed the applicant's design criteria used for primary containment isolation as described in Appendix F of the FSAR. In cases where a word-by-word interpretation of the criteria do not reflect considerations of the BWR suppression pool design concept, isolation provisions were developed on another acceptable defined basis. These include improvements in accessibility, inspection, maintenance and decreased probability of failure provide additional confidence that the systems will mitigate the accident consequences.

Based upon our review and the experience at other operating plants, we conclude that DAEC containment isolation design meets the intent of General Design Criteria 55, 56 and 57[7] and is acceptable.

6.2.5 Combustible Gas Control

Following a loss-of-coolant accident (LOCA), (a) hydrogen gas could be generated inside the primary containment from a chemical reaction between the fuel rod cladding and steam (metal-water reaction) and (b) both hydrogen and oxygen would be generated as a result of radiolytic decomposition of recirculating coolant. If a sufficient amount of the hydrogen is generated and oxygen is available in stoichiometric quantities, the subsequent reaction of hydrogen with oxygen at rates rapid enough to lead to significant over-pressure could lead to failure of the containment to maintain low leakage integrity. General Design Criterion 41 of Appendix A to 10 CFR Part 50 requires that systems to control hydrogen, oxygen, and other substances which may be released into the primary containment be provided as necessary to control their concentrations following postulated accidents to ensure that containment integrity is maintained.

In accordance with the guidelines of the supplement to Safety Guide 7,[24] "Control of Combustible Gas Concentrations in Containment Following a loss-of-Coolant Accident," the applicant has proposed a containment atmosphere dilution (CAD) system using nitrogen gas as the diluent. The nitrogen dilution system concept or CAD system satisfies the requirement for maintenance of an oxygen deficient (inert) containment atmosphere during the post-LOCA period. This would be accomplished by addition of nitrogen gas from an external nitrogen makeup and supply system. As nitrogen is added, the containment pressure would rise during the post-LOCA period. The applicant will limit the peak repressurization pressure to 30 psig when the CAD system is used. Based on an assumption of zero leakage from the primary containment and the assumptions indicated in Safety Guide No. 7,[24] a containment pressure of 30 psig would be reached approximately 35 days after occurrence of the postulated loss-of-coolant accident.

Instrumentation and sampling stations will provide the reactor operators with the necessary information as to the radioactivity levels, the radioisotopes, the hydrogen and oxygen concentrations, and local meteorology to assure that venting operations will be carried out safely and that off-site doses from any radioactivity released will be minimized. We have calculated exposure doses resulting from such venting after 30 days and found them to be well within acceptable criteria (see paragraph 15.3).

The proposed nitrogen dilution system is designed as an engineered safety feature system and will be a redundant, independent, and Category I seismic design system. Two redundant hydrogen and oxygen analyzer systems will be provided to monitor the concentration of these gases inside the containment. During operation, the nitrogen will be injected into the torus and drywell via the existing spray systems provided for the torus and drywell compartments. The CAD system's electrical design will conform to the applicable portions of IEEE-279.[5]

We have reviewed the design criteria and operational criteria for the proposed CAD system and conclude that the system is acceptable for combustible gas control following the postulated design basis loss-of-coolant accident.

6.2.6 Main Steamline Isolation Valve Sealing System

A sealing system has not been provided to prevent direct leakage from the containment through the main steamline isolation valves. The applicant has been advised that such a system will be necessary, using staff assumptions for post-LOCA dose calculations, to satisfy the guidelines of 10 CFR Part 100.[6] The applicant has indicated that a seal system will be designed and will be described in an amendment to the FSAR. This amendment will be submitted during the first quarter of 1973. Following staff acceptance of the design, the applicant plans to install the seal system during (or not later than) the first refueling outage. We find this plan to be acceptable.

6.3 Emergency Core Cooling Systems (ECCS)

The ECCS subsystems provide emergency core cooling during those postulated accident conditions where it is assumed that mechanical failures occur in the primary coolant system piping, resulting in a loss-of-coolant from the reactor vessel at rates greater than the available coolant makeup capacity using normal operating equipment. The ECCS subsystems are provided in sufficient number, diversity, reliability, and redundancy that, even if any active component of the ECCS fails during a loss-of-coolant accident, adequate cooling of the reactor core will be maintained.

The emergency core cooling system consists of two high pressure systems and two low pressure systems. The former systems are the high pressure coolant injection system (HPCIS) and the automatic depressurization system (ADS). The latter systems are the low pressure coolant injection system (LPCIS), which is one mode of operation of the Residual Heat Removal System (RHRS), and the core spray system (CSS). The ECCS for the Duane Arnold Energy Center are functionally similar to those of other licensed General Electric 1967 product line BWR facilities. They are Category I seismic design systems and are designed and fabricated in accordance with the criteria set forth in Appendix A to the FSAR which we find acceptable.

All the emergency core cooling systems are initiated by a high drywell pressure signal or a reactor vessel low water level signal, except for the Automatic Depressurization System (ADS). Initiation of ADS requires coincidence of both of these signals and a third signal that provides discharge pressure indication and hence assurance of the operation of any low pressure cooling system pump prior to initiation of the ADS. The ECCS is designed to provide adequate core cooling and to limit the peak fuel cladding temperature for the complete accident spectrum up to and including the design basis loss-of-coolant accident. For analysis and evaluation of ECCS effectiveness, the size of the design basis break is obtained by summing the areas of a completely severed suction line to a recirculation pump and the effective area of eight jet pump nozzles. This is the worst case for ECCS analysis. An analysis has also been made to evaluate the effectiveness of the ECCS to limit the peak fuel cladding temperature in the event of a main steamline break inside the drywell, upstream of the flow limiters.

A loss of offsite power will not prevent ECCS operation and all evaluations have been made assuming that only onsite electrical power is available. In addition, ECCS performance capability has been shown to be adequate assuming a failure of any active component within the ECCS. This single failure criterion has been applied coincident with the assumed loss of offsite power.

The applicant analyzed the availability of adequate net positive suction head (NPSH) for all ECCS pumps in conformance with Safety Guide No. 1[25] which requires that there be no reliance on calculated increases in containment pressure. The most limiting case occurs during the long term transient following the design basis LOCA when one core spray and one RHR pump will be running continuously. In this operating condition, the NPSH requirement for the spray pump is the limiting parameter. The analysis shows that a containment pressure margin of about 1.5 psi will be available throughout the long-term post-LOCA period to assure adequate NPSH for the core spray pumps for the above cited conditions. Although the design does not fully meet the provisions of the safety guide, we have concluded that the applicant's analysis is conservative and that there should be adequate NPSH to the ECCS pumps, in the unlikely event of a LOCA.

6.3.1 High Pressure Coolant Injection System (HPCIS)

The HPCIS includes one 100% capacity steam turbine driven pump which injects water through one of the feedwater lines and the feedwater spargers into the reactor vessel. Steam for the HPCIS turbine is supplied from one of the main steam headers in the drywell. Exhaust steam is discharged to the suppression pool through a submerged pipe. Initially, the HPCIS pump takes suction from a common header connected to the two condensate storage tanks. These tanks have a combined capacity of 400,000 gallons, of which 75,000 gallons are held in reserve for the HPCI system. Should this supply be inadequate, suction is transferred either automatically or manually to the suppression pool.

In the event of a loss-of-coolant accident resulting from a small break (i.e. equivalent to the rupture of pipes smaller than 4 inch ID if water filled and 11.5 inch ID if steam filled), the HPCI system can provide adequate core cooling, unassisted. However, for intermediate size breaks in water filled pipes (i.e. equivalent to pipes between 4 and 6 inches in internal diameter), the HPCI system must act in conjunction with the low pressure core cooling systems. For large size breaks (i.e. equivalent to 6 inches ID or larger pipe) the HPCI system is not required since the high fluid flow rate and energy loss cause rapid, unassisted vessel depressurization to lower pressures that permit operation of the low pressure core cooling systems.

6.3.2 Auto-Depressurization System (ADS)

The auto-depressurization uses four of the six dual-purpose safety-relief valves of the Pressure Relief System described in paragraph 5.2.2 of this evaluation report. The pressure relief valves open automatically upon coincident signals of reactor vessel low water level, primary containment (drywell) high pressure, and discharge pressure indication of any LPCIS pump but only after a timer delays operation of the relief valves for two minutes. If an operator determines that the initiation signal is false or depressurization is not required, the timer may be recycled.

The ADS does not itself provide cooling, but depressurizes the reactor so that the low pressure core cooling systems can operate. The ADS is redundant to the HPCIS and is only required if the HPCIS cannot maintain the reactor water level following a loss-of-coolant accident. However, with the above-mentioned coincident signals, it will activate. Similar to the HPCIS, the ADS is not required for large breaks.

6.3.3 Core Spray System (CSS)

The CSS consists of two subsystems, each with an electric motor driven pump which can spray water drawn from the suppression pool onto the top of the reactor core. The system can be powered by either offsite power or the onsite diesel generators. Each subsystem is powered by a separate diesel-generator. No single failure of any component can affect both systems.

The CSS provides cooling water following all loss-of-coolant accidents except those resulting from small breaks that can be controlled by the HPCIS. The Core Spray System is redundant to the Low Pressure Coolant Injection System (LPCIS) and can provide adequate core cooling independently of the LPCIS.

6.3.4 Low Pressure Coolant Injection System (LPCIS)

The LPCIS is one mode of operation of the four Residual Heat Removal System (RHRS) pumps. The LPCI system injects suppression pool water into the vessel plenum below the core through the unbroken recirculation loop to reflood the core. The LPCI control system determines which recirculation loop is unbroken by measuring the pressure differential between the loops, aligns the valves to direct the flow into the unbroken loop, and opens the injection valve after the reactor pressure has fallen below the LPCI system design pressure. The system can be powered by either offsite power or the onsite diesel generators. Two of the pumps are powered by each diesel-generator.

The LPCIS provides cooling water following all loss-of-coolant accidents except those resulting from small breaks that can be controlled by the HPCIS. Although the LPCIS is considered a backup system to the CSS, its capability to provide adequate core cooling independently of the CSS is not evaluated, since no single failure can prevent operation of both subsystems of the CSS. Thus there should always be at least one functional subsystem in the CSS.

6.3.5 *Functional Performance*

In Section 6.7 of the FSAR, the applicant provided an analysis of the performance of the ECCS using the assumptions and calculational techniques described in the Commission's Interim Policy Statement dated June 19, 1971 titled "AEC Adopted Interim Acceptance Criteria for Performance of ECCS for Light-Water Power Plants." The assumptions established by the criteria were applied without deviation.

Following our review and evaluation of the discussion and analyses presented by the applicant, we have concluded that the design and intended operation of the DAEC emergency core cooling systems are acceptable. The design meets the requirements of the AEC interim acceptance criteria. These criteria require that the consequences of the loss-of-coolant accident are such that (a) the calculated maximum fuel rod cladding temperature does not exceed 2300°F, (b) the amount of fuel rod cladding that reacts chemically with water or steam does not exceed 1% of the total amount of cladding in the reactor, (c) the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling, and before the cladding is so embrittled as to fail during or after quenching, and (d) the core temperature is reduced and decay heat is removed for an extended period of time.

7.0 INSTRUMENTATION AND CONTROLS

7.1 General

Our review encompassed the reactor protection and control systems, and the engineered safety feature systems. The AEC General Design Criteria (GDC) and the proposed IEEE Criteria for Nuclear Power Plant Protection Systems (IEEE 279)[5] dated August 1968 served as the bases for evaluating the adequacy of the design of these systems.

The evaluation of the Duane Arnold Energy Center was accomplished by comparing its design with that of the previously evaluated Vermont Yankee nuclear facility.

We have also evaluated the information peculiar to Duane Arnold nuclear facility in the areas of: radiation and environmental qualification, protection system testability, incident and post accident monitoring instrumentation, indication of reactor protection and engineered safety feature bypasses, anticipated transients without scran (ATWS), APRM reactor trip at 15% power, condenser low vacuum trip, and independence of redundant plant protection system channels.

We have reviewed various schematic diagrams to confirm conformance with the design criteria and have reviewed the installation at the site.

7.2 Plant Protection and Control Systems

7.2.1 Comparison of Protection Systems

The applicant indicated that the designs of the Duane Arnold Plant reactor protection systems (RPS) and engineered safety features (ESF) are essentially identical to those of Vermont Yankee. Several changes were made to provide more complete circuit separation between redundant equipment and improve testability. The changes are listed on Pages M.3-3 thru -11 of the FSAR. We have found that the changes improve safety, conform to the criteria, and make the design consistent with recently approved designs. The remaining portions of the RPS and ESF systems were found to be essentially the same as Vermont Yankee plant and are acceptable.

7.2.2 Comparison of Control Systems

The applicant has stated that the major control systems for this plant are generally identical to the similar systems of the Vermont Yankee plant with the few minor differences listed on Pages M.3-18 thru -26 of the FSAR. We have found that these minor differences have not changed the functional design nor degraded the safety of the plant. We conclude that the control systems are acceptable.

7.2.3 Protection System Testability

The applicant included additional circuitry and features in the design to permit testing of the plant protection systems during power operation. Our review of this additional circuitry, not included for the Vermont Yankee plant, confirmed that the plant protection system and engineered safety feature system are testable during power operation. We conclude that this design is acceptable.

7.2.4 Bypass Indication for Plant Protection System and Engineered Safety Feature Equipment

The design of the instrumentation and controls for the plant protection system and engineer safety features includes control room indication to identify reduction in system redundancy which could result from operator action. Our review has determined that reasonable annunciation and indication is included at the system level for these redundant safety systems. We conclude the bypass indication systems are equivalent to those of previously licensed plants and are acceptable.

7.2.5 APRM Reactor Trip at 15% Power

The design includes an APRM reactor trip at 15% power while operating in the startup mode. Our review has found that this feature satisfies the requirements of IEEE 279[5] and we find it acceptable. This trip has the same function as the IRM trip.

However, at least for the time being, the applicant proposes to retain the IRM trip. If, in the future, the applicant proposes to disable the IRM trip, we will require analysis to justify this deletion.

7.2.6 Condenser Low Vacuum Trip

The condenser low vacuum reactor trip has been deleted. An additional circuit which closes the main steam isolation valves on low condenser vacuum has been provided to assure steam flow is restricted from the main condenser during a leak. This valve closure initiates a reactor trip. Our review has determined that this circuitry satisfies the requirements of the criteria and is acceptable.

7.3 Independence of Redundant Plant Protection System Channels

Our review of the FSAR revealed that the applicant's criteria are acceptable and the elementary diagrams indicated that the criteria are properly implemented. Our visit to the plant site revealed that discrepancies identified and corrected on previous boiling water reactors (BWRs) were also corrected at the Duane Arnold facility. These items were: 1) connection of redundant protection channels to single switches and terminal boards in the control room panels; 2) installation of redundant protection system switches on control room panels within a few inches of each other with their wiring bundled and routed together, and 3) installation of redundant protection system instruments on a common rack outside the control room.

We have reviewed the cable installation design, routing and identification criteria relating to the preservation of the independence of redundant channels. We have found these criteria and their implementation to be acceptable.

We conclude that the cable installation design, routing and separation criteria relating to the preservation of the independence of redundant channels are acceptable.

7.4 Incident and Accident Surveillance Instrumentation

The applicant has provided a list identifying the redundant instrument channels whose readouts are presented to the operator for assessing plant conditions during and subsequent to accident and operational occurrences. Our review has found that the systems are comparable to recently approved BWRs and we conclude that the incident and accident surveillance instrumentation is acceptable.

7.5 Anticipated Transients Without Scram (ATWS)

The applicant stated that provision will be made to include the function of tripping the recirculation pumps as described in the General Electric Company report NEDO-10349, March 1971, with one exception. This exception is that the recirculation pumps will be tripped on high reactor pressure only. The report (NEDO-10349) proposed tripping the pumps on coincidence of high neutron flux and high reactor pressure. General Electric has not completed the details of the final design of this generic item. We will review the final design prior to its installation in the plant.

Since no decision has been made to require this improvement on a backfit basis, we have concluded that it is acceptable to operate Duane Arnold prior to installation of the recirculation pump trip. Notwithstanding, the applicant will have the recirculation pump trip system installed prior to the initial fuel load date.

The staff agrees with the view of the Advisory Committee on Reactor Safeguards, as stated in its letter on the Peach Bottom Atomic Power Station, dated September 21, 1972, that the addition of the recirculation pump trip as proposed by the applicant represents a substantial improvement in protection of the reactor for anticipated transients without scram; however, the staff has not completed its review of all the transients discussed in the General Electric Company Topical Report NEDO-10349.[26] Completion of our review of this topic is pending receipt of the review of responses to additional information which was requested from General Electric in a letter dated June 13, 1972. The staff has not concluded that the proposed recirculation pump trip provides a completely acceptable degree of protection against anticipated transients without scram for reactors of this general type. This conclusion is pending our receipt and review of the outstanding information cited above. The General Electric Company has indicated that the information requested by the staff regarding anticipated transients without scram will be submitted as a topical report in early 1973.

7.6 Control Rod Reactivity Control

In response to the current regulatory staff concern for the control of selection and movement of control rods during reactor startup (see Control Rod Drop Accident discussed in paragraph 15.2.2 of this Safety Evaluation), the applicant has committed to adopt and install additional controls which meet the approval of the Staff. As stated in their answer to question 14.15 in Amendment 9 to the FSAR: "The technical essence of the rod sequence control system that is deemed acceptable by the AEC staff for the Brown's Ferry docket will be implemented into the DAEC plant design. Therefore, further (applicant) analysis and update of related material will be deferred until the resolution is finalized." The applicant will, however, have this system installed and operable prior to operation above 1% power level. We find this to be an acceptable approach.

7.7 Radiation and Environmental Qualification

7.7.1 Radiation Qualification

The applicant has identified the safety related equipment located inside the containment which must operate during or following a DBA. The equipment will be capable of functioning under the post-accident temperature, pressure and humidity conditions for the time periods required. This capability has been demonstrated by testing. We conclude that the environmental testing of safety related equipment is acceptable.

8.0 ELECTRIC POWER SYSTEMS

8.1 General

The AEC General Design Criteria (GDC) 17 and 18, AEC Safety Guides 6[27] and 9,[28] and IEEE Criteria for Class IE Electrical Systems for Nuclear Power Generating Stations (IEEE 308)[29] served as the bases for judging the acceptability of the DAEC electrical power systems.

8.2 Offsite Power

Duane Arnold Energy Center will be interconnected to the Iowa Electric Light and Power Company transmission grid through 345 kV and 161 kV transmission systems by two 345 kV and three 161 kV transmission lines. A 345 kV-161 kV sectionalized switchyard is provided with a 345-161-34.5 kV transformer connected between the two sections. The 34.5kV section is arranged in a ring bus scheme connected to its two transmissions lines and the 345-161-34.5 kV transformer. The 161 kV section is arranged in a breaker and one-half scheme except for the startup transformer and the 345-161-34.5 kV transformer which have a single breaker each. The two 345 kV transmission lines on independent towers emanate from the station westwardly on the same right of way. One line then extends south to the Hills substation and the other line north to the Hazleton substation. The three 161 kV transmission lines on independent towers emanate from the station on three different rights-of-way. One line extends west to the Garrison substation, one line southwest to the Beverly substation, and one line east to the Hiawatha substation. Two paths bring power from the switchyard to the plant. One overhead transmission from the 161 kV section is connected to the startup transformer and the second is an underground line from the 345-161-34.5 kV transformer connected to the standby transformer. Each of the two 4 kV essential buses can be supplied by either of these paths. Either of these separate and independent paths is capable of supplying accident loads from any of the five incoming lines.

All switchyard breakers have protective relaying and d-c control circuits. A fault on an incoming transmission line with loss of the d-c control circuit power for switchyard breaker operation could lead to a complete loss of offsite power. The applicant has reviewed this area and provided a procedure which would reestablish offsite power within one hour by manually tripping individual breakers in the switchyard to eliminate the fault. Trained plant personnel will be onsite continuously and a written procedure will be available. The results of analysis by the applicant have indicated that one and one-half hours are available after the loss of all a-c power prior to the torus temperature exceeding an unsafe limit. Torus temperature is the limiting condition which establishes the time required to restore power for safely cooling the reactor. We, therefore, conclude that this operation satisfies the requirements of GDC 17[7] and is acceptable.

The applicant has completed transient stability studies that simulated 345 and 161 kV transmission line faults and the loss of the Duane Arnold generator. The results have shown that the loss of offsite power would not occur under these conditions.

Our review has determined that the applicant's offsite power system is in accordance with GDC 17[7] and 18 and is acceptable.

8.3 Onsite Power

The engineered safety features and safe shutdown loads are divided between two independent and separate 4 kV emergency buses, either of which is capable of supplying minimum engineered safety features or safe shutdown equipment. Each of these two buses is capable of receiving power from the startup transformer, a standby transformer or a diesel generator unit. There are four 480 volt emergency buses, two supplied from one 4 kV emergency bus and the other two supplied from the other 4 kV emergency bus. Separation and independence of these redundant power systems have been maintained.

Two 2850 kW (continuous rating) emergency diesel generator units provide the onsite power supply, one unit for each 4 kV emergency bus. Each diesel generator unit is started automatically on loss of power to the emergency buses or low reactor water level or high drywell pressure. The accident loads are automatically sequenced on each 4 kV emergency bus.

Each diesel generator unit and associated auxiliaries are housed in water-tight, separated structures located within the turbine building. Each diesel generator unit is a self-sustaining entity with its own independent lube oil, fuel oil, cooling water and control systems. A common Category I diesel fuel oil storage tank is provided with two transfer pumps, one for each diesel unit. A day tank with a four-hour fuel capacity is provided for each diesel. The diesel fuel oil storage tank contains sufficient fuel for operating the diesel units for approximately seven days with the generator supplying accident and shutdown loads.

The diesel generator units for the Duane Arnold plant are identical to units provided on some presently operating plants. The assignment of electrical loads during sequencing for this plant is expected to exceed the voltage and frequency recovery time limits expressed in Safety Guide 9.[28] The applicant will demonstrate the adequacy of this system by including margin tests as part of the one hundred in-plant starting and loading verification preoperational tests. These margin tests will include adding an additional 10% load of similar electrical characteristics to the initial load increment during their testing. Secondly, sequencing intervals will be reduced a small amount in each succeeding test until the ability of the diesel generator to pick up the designated loads fails to occur. We conclude that this test program will verify that margin exists in this system and the reliability will not be degraded. We will evaluate the results of the test program upon its completion.

The applicant stated and we have confirmed that the emergency power distribution systems are split in accordance with Safety Guide 6 except for the motor operated valves associated with LPCIS and the ADS valves. The power supply to these valves is automatically transferred between redundant buses. We have determined that this design satisfies the single failure criterion and is consistent with previously approved 1967 product line BWR plants; hence, we conclude that the design is acceptable.

The 115 volt a-c systems provided for safety are arranged with two independent reactor protection system buses and are acceptable.

The d-c power supply consists of two independent 125 volt systems and a 250 volt system. Each system has a battery with its own charger and distribution panel. The battery charger is capable of keeping the battery fully charged and supplying the d-c system loads simultaneously. Each battery is located in a separate room with an independent ventilation system. Each battery is sized to supply essential loads for a period of four hours on loss of its battery charger during any plant operating or incident condition. This d-c power supply arrangement provides for adequate separation and independence of redundant supplies.

We believe the onsite power system satisfies the GDC,[5] IEEE 308[29] and Safety Guides 6[27] and 9;[28] thus, we conclude that the system is acceptable subject to successful completion of the in-plant diesel test program.

9.0 AUXILIARY SYSTEMS

The evaluation of safety related auxiliary systems, as set forth in the following subsections, is based on reactor safety requirements, radiological safety requirements, and power generation requirements. These systems are grouped in the following paragraphs to indicate the requirements that are applicable.

The auxiliary systems necessary to assure reactor safety are: (1) reactor heat removal service water system; (2) emergency service water systems; (3) river water supply system; (4) diesel auxiliary systems; (5) fuel storage and handling facilities; and (6) ventilation and air conditioning systems, and the instrumentation and controls required for their operation. These systems have been designated for Category I seismic design.

Other auxiliary systems whose failure would not prevent safe reactor shutdown, but may interrupt power generation or be a potential for a radiological release to the environment are: (1) reactor building cooling water system; (2) general service water system; (3) fire protection water system, except in HPCI and RCIC areas; (4) fuel pool cooling and cleanup systems; (5) well water system; (6) instrument and service air system (Category I components have local emergency air tanks); and (7) potable water, drainage, sampling, lighting (except local emergency), and plant heating systems. These systems have been designated for Category II seismic design.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

The new fuel storage facility was found to provide proper drainage in the event of flooding. Location, racks, and lifting devices were studied for strength, movement, and freedom from other operations in the area. Construction details were examined for the liner, cover, lifting hooks, and seal. The applicant has assured the staff that the auxiliary hoist pull-up capacity is less than the new fuel rack structural strength in this direction.

New fuel storage is provided for by a dry-vault with a rack capacity for 30% of a full core load. The loaded rack is a Category I seismic design structure, and can withstand a pull-up force equal to the capacity of the overhead crane auxiliary hoist. There is adequate dry storage vault space in the Category I structure to store an additional 13% of a full core. A water drain prevents collection of water in the vault. In a dry condition k_{eff} will not exceed 0.90. In the event of flooding, k_{eff} will not exceed 0.95. We conclude that the new fuel storage facility is acceptable for the required service.

9.1.2 Spent Fuel Storage

The spent fuel storage pool and racks provide underwater storage space for spent fuel assemblies that require shielding and/or cooling prior to shipment. The pool also provides for a shipping cask area that permits safe transfer of spent fuel into the cask. The spent fuel is stored in a Category I seismic design pool located at the operating floor of the reactor building. Storage racks are provided in the pool with a capacity to accommodate 130% of full core load in a subcritical array with a k_{eff} less than 0.90.

In our evaluation of the pool water makeup system to replace evaporative losses from the pool, it was determined that in the event of a failure of the normal seismic Category II makeup system no make-up water was available from a seismic Category I source. The applicant was required to provide a hose connection at the operating floor level which provided water from the seismic Category I emergency service water system.

The applicant provided mechanical stops to limit crane movement only to the area around the cask loading area. The applicant gave assurance that the sliding gate separating the spent fuel pool from the cask loading pool would be in-place and watertight prior to any movement of the cask. For the postulated event of a cask drop, the applicant investigated and determined that there is a finite but small probability that the liner and concrete may crack. In this event leakage would be collected by the drain system immediately beneath the cask loading pool and discharged to the radwaste system. Structural supports were examined and found adequate to remain intact following a cask drop from the maximum possible height. We conclude that the spent fuel storage facility design is acceptable.

9.1.3 Spent Fuel Cooling and Cleanup System

The spent fuel pool cooling and cleanup system is designed to maintain the water quality and clarity and to remove the decay heat generated by the spent fuel assemblies stored in the pool.

The spent fuel cooling system is a Category II system designed to maintain pool water temperature below 125°F when removing the maximum heat load. This load is derived as the sum of the decay heat released by the average spent fuel batch (one-third core) from an equilibrium fuel cycle, plus the heat being released by the batch discharged from the previous refueling.

The maximum possible heat load is the decay heat of a full core discharged from the equilibrium fuel cycle, plus the remaining decay heat from the batch (one-third core) discharged at the previous refueling. The residual heat removal system (RHRS) is required to be connected to and operated in parallel with the spent fuel pool cooling system to remove this heat load in order to maintain the pool water temperature below 150°F. A spool-piece connects the RHR system to the fuel pool cooling system to assure that the Category I RHR system is always available during normal operation of the plant. The applicant has stated that the reactor would be shutdown and maintained in a shutdown condition if the residual heat removal system is needed to control the pool water temperature.

We conclude that these systems are acceptable and, with the operating restriction of reactor shutdown when the RHR system is interconnected to the fuel pool, will provide the necessary assurance that adequate cooling will be available.

9.1.4 Fuel Handling Equipment

The subject of reactor vessel head and component handling prior to, and following fuel servicing, is under study by the applicant. An evaluation of a postulated head and shield segment drop accident is under investigation. The applicant is analyzing the issue and is scheduled to file a report on the matter by the end of January 1973. We find this equipment acceptable on the same basis as previously reviewed plants, but plan to consider the head and shield drop as a new accident situation which is generic to all BWR plants.

9.2 Water Systems

The nearby Cedar River supplies essential cooling and evaporative cooling tower makeup water. The evaluation of the systems and structures are set forth in the following sections.

9.2.1 River Water Supply System

The river water supply system provides cooling water for the RHR service water and ESWS. It also provides makeup water for the cooling towers. In addition, it provides water to the Fire Protection System and General Service Water System. The system consists of two independent and redundant pumping systems that are installed in the river intake structure with the pump motors located above the PMF level, and suctions located below the lowest calculated river flow level.

Our evaluation of this system concentrated on the requirements for cooling water upon a loss-of-coolant accident, or loss-of-offsite power. Following such an occurrence, one of the 6000 gpm pumps in each of two redundant systems will automatically connect to the emergency diesel bus. Simultaneously, pump house valves assume a fail-safe position which assures that all the river water pump output is channeled to the RHR service water and emergency service water pump wet pits located in the pump house. We conclude that the system design for emergency function is adequate and will not be compromised by its normal functions.

9.2.2 Residual Heat Removal (RHR) Service Water System

The RHR service water system provides cooling water to the reactor residual heat removal heat exchangers under post-accident conditions and a water source if post-accident flooding of the core or primary containment is required. In addition, the system provides cooling water to the RHR heat exchangers during normal operational modes.

The system consists of two Category I seismic design pumps, in two redundant systems which take suction from the Category I seismic design pump house wet pit. Sufficient redundancy of piping and active components provide protection from the single active or passive failure. Two half-size RHR service water pumps supply one full-size RHR heat exchanger. There are four half-size pumps, and two full-size heat exchangers. Two of the half-size pumps are connected to two of each of two independent and redundant diesel-generator buses.

We conclude that the system design for emergency function is adequate and will not be compromised by its normal function.

9.2.3 Emergency Service Water System (ESWS)

The ESWS is provided to supply cooling water to essential safeguards equipment under loss-of-offsite power or a loss-of-coolant accident. Components supplied are the emergency diesel generator coolers, RHR Pump Seal Coolers, RHR and Core Spray Pump Room Cooling Unit, HPCI and RCIC Room Cooling Unit, Control Building Chiller and Core Spray Pump Motor Bearing Cooler.

The system consists of two full capacity pumps each connected to a separate redundant train for supplying all the emergency needs as indicated above. One pump is connected to each of the two independent and redundant diesel generator buses to assure available emergency electric power.

Our evaluation indicates that the system design is capable of continual operation following a single active or passive failure. The structure and wet pit of the pump house, where the pumps are heated, are designed to seismic Category I criteria and protected from external missiles. We conclude that the system design is adequate to perform its safety functions.

9.2.4 Ultimate Heat Sink

Our review of the analysis of the PMF, minimum river flow, Category I sheet-pile river barrier wall, river intake structure, river water supply system, pump house structure, residual heat removal system, and emergency service water system, indicates that the intent of Safety Guide 27,[30] Ultimate Heat Sink, has been satisfied.

9.2.5 General Service Water System & Reactor Building Cooling Water System

Both systems provide for cooling of equipment to meet plant requirements for startup, normal operation and shutdown. The reactor building system is separate from the general service water system in order to minimize the possibility of reactor building radioactive fluids being released to the general service water. The Reactor Building Cooling Water System has a separate pump and heat exchanger system. The Reactor Building Cooling Water System Heat Exchanger is cooled by the General Service Water fluid for heat removal from this closed-loop system. The General Service Water System pressure exceeds the Reactor Building Cooling Water System pressure in order to prevent any release of radioactive liquid to the service water.

The General Service Water effluent is cooled by the open cooling tower heat removal capability used by the main turbine condenser.

We have reviewed the system design to assure that a potential path to the environment for radioactivity has been prevented and therefore we find that both cooling systems are acceptable for their required service during normal plant operation.

9.2.6 Well Water System

The well water system serves two functions: cooling water for ventilation cooling units and potable water for human consumption and demineralizer makeup. Capacity estimates and system design appear reasonable and reliable. The concern for backflow of potentially radioactive liquids to the well system was investigated. As a result the applicant will provide both subsystems with backflow preventers to limit the possibility of well contamination.

There are two wells, 2000 feet apart which take suction from glacial deposits 120 to 140 feet deep and which are sealed to prevent collection of less desirable groundwater from shallow aquifers. The wells are located west of the plant; hence, should liquid radwaste somehow enter the groundwater at the plant, it would flow to the east toward the river, away from the wells. We conclude the system design is acceptable for the required service during normal plant operation.

9.3 Process Auxiliaries

9.3.1 Compressed Air System (Instrument and Service Air)

The compressed air system provides air to both the service air system and instrument air system for normal operation of plant equipment, instrumentation and valves. In addition, the systems maintain the required pressure on the safety related accumulator tanks which are required for specific valve operations. Compressed air for the system is provided by two motor-driven compressors. Each compressor can provide both services.

Appropriate means of isolation of nonessential air lines is provided via spring operated pressure controlled stop valves. Essential instrument air does not have automatic isolation. Our evaluation confirms that total failure of the compressed air system can be accommodated since safety related equipment served by the Compressed Air System have the following design features:

a. All containment isolation valves and dampers have Category I air accumulators which provide reliable air supply in this event.

b. Air operated isolation dampers in the standby gas treatment system and control building ventilation system, normally open and fail open, must remotely close in the event of charcoal filter overheating. For this reason, and should the compressed air system fail, these dampers are equipped with separate Category I air tanks.

c. All other air operated valves are designed to fail in a fail-safe position.

We find the compressed air system acceptable for normal operation and separate seismic Category I air tanks for systems essential for safe plant shutdown to be an acceptable safety feature.

9.3.2 Equipment and Floor Drainage Systems

The reactor building has two drainage systems: one system from the primary containment, and the other from the secondary containment. The primary containment drainage system collects equipment and floor drainage from a gravity-fed sump. Sump pumps transfer the contents to the radwaste system. The secondary containment drainage system likewise collects drainage in the same fashion where sump pumps tranfer the liquid to the radwaste system.

Turbine building equipment and floor drains are also divided into two systems: those drains serving equipment and areas of potential radioactive leakage and those serving non-radioactive areas such as the building roof. The potentially radioactive drains collect in a sump where a pump transfers the liquid to the radwaste system. The separate non-radioactive drains empty by gravity to a separate sump and are pumped into the storm drain system. In our evaluation we also assured ourselves that a flooded sump would not backup the drainage piping system into a compartment housing Category I equipment, e.g. the diesel generator rooms.

Based on the results of our review we conclude the system design is adequate for the required service during normal and emergency operation.

9.4 Heating, Ventilating and Air-Conditioning Systems

9.4.1 Control Room Air Conditioning and Ventilation System

There are separate air conditioning systems for each major building. One system services the main battery rooms, switchgear room, cable spreading room, control room, radwaste control room, and office building and maintains a positive pressure to prevent infiltration.

This system draws in and filters varying amounts of outside air under normal conditions. Intake air radiation monitors will isolate the normal ventilation path and connect the intake to high efficiency filter trains. Power for the filtration-recirculation system can be transferred to the emergency bus. In addition, the filtration-recirculating system is seismic Category I and is located in a Category I structure.

9.4.2 Turbine and Auxiliary Buildings Ventilation Systems

Auxiliary buildings, such as the turbine building, radwaste building and pump house, have their own ventilation system for filtering, heating, and cooling.

The turbine building ventilation system is a once-through system composed of three subsystems. Two redundant exhaust fans serve those turbine building spaces that may contain radioactive materials. During normal operation, these special fans exhaust into the offgas stack. Spaces below the turbine operating floor exhaust through the main plant exhaust plenum. Spaces above the operating floor are exhausted via fans in the roof. The entire turbine building ventilation system will be isolated should a release of radioactivity in excess of existing background levels occur.

In this event building exhaust will be directed to the reactor building exhaust plenum where the atmosphere is exhausted to the standby gas treatment system and then to the offgas stack.

9.4.3 Radwaste Building Ventilation System

The radwaste building is serviced by a once-through system which exhausts through prefilters and HEPA filters to the space surrounding the torus, or the reactor building exhaust plenum. Closure for both radwaste supply and exhaust dampers is possible for building isolation.

9.4.4 Reactor Building Ventilation System

The reactor building ventilation system has two subsystems to supply air at the refueling floor level, and the other to supply air below this level. The exhaust fans supply air at a rate of 10 air changes per hour with the air being exhausted via the main plant exhaust plenum creating a negative pressure in the spaces. Should a release of radioactivity occur, the reactor building will be automatically secured, the supply and exhaust fans shutdown, and the atmosphere exhausted to the standby gas treatment system and then to the offgas stack.

We conclude that the heating, ventilation and air conditioning systems are acceptable and will route contaminated air to the standby gas treatment system prior to release to the atmosphere.

9.4.5 Engineered Safeguards Heating and Ventilating Systems

The "engineered safeguards" area heating and ventilating systems (so called by the applicant) are required to assure a suitable ambient temperature in the Emergency and Switchgear Battery Rooms, Emergency Diesel-Generator Rooms, Emergency Cooling Pump Room (RHR and ESWS), and Reactor Building RHR, RCIC, HPCI and Core Spray Pump Rooms. All of these systems or areas are required for reactor protection.

The equipment is installed in Category I seismic design structures, with supply air filtered, and methods for heating and cooling provided. The systems are redundant, and are supplied from both normal and emergency busses. Electric heaters take their power supply from the same busses to provide freeze protection. Cooling water coils are supplied from the ESWS to provide a proper ambient temperature when necessary.

We have evaluated these systems and find that no single component failure will prevent system operation during normal and accident conditions and therefore, conclude that the systems are acceptable.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

The fire protection system will consist of water, carbon dioxide and chemical fire fighting systems whose design will not prevent Category I systems from performing their function should the fire fighting system fail. In those cases where a failure of fire fighting systems could impair other Category I systems, that portion of the fire protection system is designed to Category I standards.

The fire protection water system utilizes water pressure from a loop header and distribution piping supplied from fire pumps located in the pump house. The pumping system is composed of a diesel driven fire pump, motor driven fire pump, and motor driven jockey pump. Water deluge systems, sprinkler system (wetpipe and drypipe), as well as local hose connections distribute water for extinguishing fires. We evaluated the plant arrangement of the diesel driven fire pump location which was near the main circulating pump expansion joints. The applicant will provide watertight doors around the diesel driven fire pumps to prevent flooding in the event the expansion joints fail.

The cable spreading room has an automatically actuated carbon dioxide extinguisher system. Portable extinguishers are provided throughout the remainder of the plant and are a non-toxic dry chemical type.

Fire detection is accomplished by temperature actuation of local sensors which alarm in the control room. Also alarming in the control room are smoke detectors and ionization detectors above the cable trays which detect products of combustion.

We conclude that the fire protection system is adequate for plant protection.

9.5.2 Communications Systems

There are four communications systems provided in the control room and throughout the plant for startup, operation, shutdown and maintenance under normal and emergency conditions. These systems include:

a. The public address system that has page and party channels and which can issue plant-wide instructions, and intercommunicate between two or more stations. It receives its power from the instrument bus.

b. The telephone system, installed by the local phone company, provides in-plant service as well as local and long distance calls. It is powered by batteries.

c. Sound powered jacks are located throughout the plant for testing and maintenance contact with other locations.

d. A VHF transmitter-receiver provides radio contact with other IELP stations. Battery power is provided for this system.

e. In addition, an alarm signal is also provided which can be transmitted over the public address system and from an outdoor siren to warn personnel of an emergency.

The applicant has indicated tests will be performed on the communication system during preoperation to assure background noise does not interfere with the communication capability. We find this to be acceptable.

9.5.3 Diesel Generator Fuel Oil Storage and Transfer System

Two redundant diesel generators, physically separated are provided with independent air starting, fuel day-tank, cooling system, and lube oil systems. Each has a continuous rating of 2,850 KW. A common fuel oil storage tank has fuel capacity for a single engine operating at rated power conditions for seven days. Each diesel compartment has one individual day tank with fuel capacity for four hours of operation. Replenishment sources of fuel oil are eight miles from the site and are within the time limitation established by use of day-tank fuel.

Each diesel is located in a Category I seismic design structure, entirely separated from each other by a common Category I seismic design wall. This Category I enclosure is at the southeast corner of the Category II turbine building. The applicant performed an evaluation to show that failure of any Category II structure, or Category II components within the Category II strucute, will not affect the operation of the diesel generators.

There are two 200 percent capacity diesel oil storage transfer pumps, one to supply each day tank from the common storage tank. Each day tank has 1000 gallons capacity surrounded by a fire wall and is located within each diesel generator compartment.

The day tank contains level switches which automatically operate the transfer pumps to refill the day tank from the storage tank. In the event of failure of this pump, a manually operated pump is available. Through selective manual valve operation, either transfer pump can fill both day tanks.

The Category I diesel oil storage tank is in a Category I seismic design structure outside the building and underground. The refill-hose connection, tank vent, and flame arrestor are located above the level of the probable maximum flood (PMF).

With this adequate fuel supply and flood protection to assure diesel generator operation during the PMF and the redundancy in equipment, we conclude that the design of the system is acceptable.

9.5.4 Diesel Generator Cooling Water System

Cooling water is supplied to the diesels by redundant and separated piping from the emergency service water system (ESWS). A single active or passive failure in the ESWS system will affect only the unit it serves.

We conclude that the diesel engine cooling water system is adequate.

9.5.5 Diesel Generator Air Starting System

Each diesel is provided with redundant and physically separate air starting compressors and receivers. The air tanks are capable of five successive engine starts without recharging. The air starting tanks have cross connections to start either engine. Our evaluation indicates the system has adequate flexibility and capacity to provide for reliable starting capability.

A separate 125 volt d-c battery is furnished each diesel engine with its own static type battery charger and bus to supply control power to the various systems of the diesels.

We conclude that the diesel generator starting system is adequate.

9.5.6 Diesel Engine Lubrication System

During operation each engine is provided with lube oil by its own shaft pump. In addition, each engine has its own lube oil makeup system which injects oil to the crankcase by a separate electric drive pump.

Our evaluation indicates that single failure would limit the lube oil for one diesel to only what is in the crankcase; the other unit would still be capable of operating for seven days without manual lube oil addition.

We conclude that the diesel engine lubrication system is adequate.

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The DAEC steam and power conversion system is of conventional design, similar to previously approved direct cycle BWR plants. Waste heat rejection is from mechanical induced draft cooling towers. The two main condensers serving the two low pressure turbines are connected in series, which permits the "A" condenser to operate at a lower pressure (3.38 psia) than the "B" condenser (7.10 psia), while operating at rated power. Circulating water temperature at the exit of the condensers is 112°F and is lowered to 87°F upon leaving the cooling tower when the ambient design wet-bulb temperatures is 76.5°F.

10.2 Turbine--Generator

The turbine is a tandem-compound reheat unit consisting of a single-flow high pressure turbine and two double-flow low pressure turbines with a design speed of 1800 rpm. Steam exhausted from the high pressure turbine is reheated by both main steam, and turbine extraction steam in two stages prior to entering the low pressure turbines. Including the extraction steam from main steam reheating, there are six extraction stages from the turbines to accomplish steam and feedwater heating. An automatic pressure-controlled steam turbine bypass system with two ten-inch lines can accommodate up to 25 percent of design steam flow directly to the two condensers serving the two low pressure turbines. These bypass valves control steam pressure during load rejection, reactor heatup, turbine startup and reactor cooldown.

The main generator is a direct driven, three-phase, 60 Hz, 22,000 volt, 663,500 kva unit with a maximum hydrogen cooling pressure of 45 psig. Hydrogen cooling supply is from a 24 tank manifold located external and to the east of the turbine building so as not to constitute a missile or fire threat to nearby equipment. Four tanks have a capacity to last 3 weeks or longer. Purging and refilling requires the capacity of 3 tanks. The tanks are refilled in place by a bulk hydrogen supplier.

The turbine generator has controls including a electrohydraulic control system, control valves, main stop valves, combined stop-intercept valves, initial pressure regulator and backup controller, steam bypass, and mechanical overspeed trip. There are alarms and interlocks for lube oil pressure, seal oil pressure, condenser vacuum, generator cooling, vibration, and field excitation.

10.3 Main Steam Line System

The four, twenty inch, main steam lines (MSL) are Category I seismic design up to the turbine stop valves. The applicant plans to conduct visual inspections of steam lines and turbine stop valves during outages with special attention for indications of leakage and change in position of pipe hangers. Photography will be used to determine movement or change of hanger positions. Every four to five years, but not necessarily concurrent with turbine dismantling, the turbine stop valves will be completely dismantled for inspection of the normally inaccessible parts.

As a result of recent Regulatory staff concerns regarding the effects of a rupture, outside of containment, of a pipe carrying high energy fluid, including a rupture of the largest main steam or feed water line, the applicant was requested to provide analysis of these effects. We have reviewed the information available to us on the Duane Arnold plant, and it appears that safe shutdown of the plant would not be prevented by the postulated failure of the steam or feedwater lines. However, to assure a thorough review of the consequences of the postulated accident, we asked the applicant on December 15, 1972, to provide us with an analysis of the problem and other relevant information pertaining to the accident consequences and the protection provided. The information provided by the applicant on the consequences of pipe rupture outside of the containment and the protection provided to assure that safe shutdown of the plant would not be prevented will be reviewed for acceptability by the staff prior to issuance of an operating license.

10.4 Main Condenser

The main condenser is designed as a two-pass series type with a divided water box. Each of two condenser sub-units is located on a rigid foundation. Flexible expansion joints between the condenser necks and turbine exhaust connections are provided to permit relative motion between these components.

Since the plant uses a deaerating type condenser, noncondensible gases are removed by steam jet air ejectors which maintain dissolved oxygen at less than 5 ppb at greater than 10 percent of design throttle flow.

The hotwell is sized to provide a two minute retention for short lived radioactive isotope decay, and a storage supply for the condensate pump suction.

Although the main condenser design is different from most systems, it is not unique and we conclude the design of the main condenser is adequate.

10.5 Condenser Evacuation and Sealing Systems

A motor driven mechanical vacuum pump is provided to evacuate the condenser during startup. Two full capacity steam jet air ejectors maintain vacuum during operation and deliver the noncondensible gases through a decay line to the offgas system.

The gland sealing system provides positive pressure to labyrinth sealing rings at the turbine shaft openings thereby sealing that gland from inleakage of air at the low pressure condenser glands, and against steam outleakage at the high pressure turbine shaft glands. Sealing steam is taken from the auxiliary boiler during plant startup and from turbine extraction steam during operation.

We conclude that the design of the condenser evacuation and sealing systems is similar to those used in previously reviewed plants and is acceptable.

10.6 Circulating Water System

The closed loop circulating water system is composed of the condenser cooling surface, and two evaporative type mechanical induced draft cooling towers. Since the condensers are series connected, the water flows first through the tubes of one condenser, then discharges to the suction of the second condenser prior to discharge to the cooling towers. The cooling tower sumps discharge by gravity to the wet pit sumps of the pump house where two half-capacity circulating pumps take their suction. Each of the towers has an inflow of 146,000 gpm at 112°F and an out-flow at 87°F. The total tower capacity is adequate to meet main condenser heat load and also the heat load from the service water system.

Makeup water to replace evaporation and blowdown losses comes from the seismic Category I river water system pipelines to the circulating water pump sump. Acid and chlorine to control algae and fungus are also added at the circulating water pump wet pit.

There are rubber expansion joints in the turbine building basement, and in the pump house. In the event any of these failed, the turbine building condenser room could flood, but no safety related equipment is present. The cooling tower sump capacity is insufficient to overflow the condenser compartment. Compartment water level indicators alarming in the control room will be installed. If the expansion joints in the pump house failed, the water would flow out through ventilation louvers to the site grade level.

We conclude that the design of the circulating water system is adequate.

10.7 Condensate Demineralizer System

The condensate from the condenser hotwell is processed by a demineralizer system which is the full-flow type with five vessels, including one spare which accomplishes demineralization and filtration by coating the vessel with Powdex resin or Solka Floc, or both. The system can be backwashed for resin regeneration or dumped for resin replacement. At design conditions, the limits on feedwater impurities (max.) are, Silica 5 ppb, Iron 5 ppb, Copper 2 ppb, Nickel 2 ppb, and Chloride 10 ppb.

System instrumentation and controls are locally mounted. Each vessel has instrumentation to indicate resin plugging, exhaustion and water conductivity. Specific problems annunicate at the local control panel and result in a single alarm in the main control room.

We conclude that the design of the condensate demineralizer system is adequate.

10.8 Condensate and Reactor Feedwater Systems

Both the condensate and reactor feedwater systems serve to provide as much as 115 percent of design feedwater flow to the reactor at 1100 psi and 420°F. There are six steps of feedwater heating from the condenser hotwell to the reactor. Two condensate pumps and two main feedwater pumps, each rated at 68 percent of design capacity, maintain feedwater flow. The oversize capacity gives automatic recirculation to assure positive pump suction under all projected load changes.

We conclude that the design of the condensate and reactor feedwater systems is adequate.

10.9 Condensate Storage and Transfer System

Two 200,000 gallon storage tanks supply the two 100 percent capacity pumps of 600 gpm each, and one 125 gpm jockey pump for condensate transfer. The two tanks are physically isolated by suction lines raised to an elevation which leaves a 75,000 gallon reserve. Water quality is that of reactor condensate, maintained by the makeup condensate demineralizer. The tanks will overflow to the reactor building equipment drain sump by way of an 1000 gallon overflow tank. Should the storage tank burst, the water will be retained within an enclosure dike with concrete pad to prevent condensate entry to the surrounding ground. The diked area has a portable sump pump to discharge this water ultimately to the radwaste disposal system.

We have reviewed the capacities, and design requirements and conclude that the condensate storage and transfer system is adequate.

10.10 Turbine-Generator Inservice Inspection

The Regulatory Staff is considering methods that may be developed for performing volumetric examination of the low pressure steam turbine during the performance of periodic turbine inspection. The applicant has been advised of our interest in the development of such a program on a generic basis covering all reactor plants, including the Duane Arnold plant. The applicant has informed us that he will keep abreast with technological developments in this area and will adopt a suitable program when one becomes available. We find this to be acceptable.

11.0 RADIOACTIVE WASTE TREATMENT

11.1 Introduction

The waste treatment systems are designed to provide for controlled handling and disposal of radioactive liquid, gaseous and solid wastes. The applicant's design objective for the liquid and gaseous waste systems is to keep levels of radioactivity released by these systems as low as practicable, in accordance with the requirements of 10 CFR Part 20[32] and 10 CFR Part 50[2]. The applicant's design objective for the solid waste system is to package and transport solid wastes in accordance with applicable AEC and DOT regulations.

Our evaluation is based on the "as low as practicable" criteria for radioactive contents of effluents discharged to unrestricted areas in accordance with Section 50.34a of 10 CFR Part 50. For Duane Arnold, the major pathway for exposure of an individual's thyroid from iodine is by the grass-cow-milk cycle. The gaseous radwaste system meets our current criteria for "as low as practicable discharge for plants already built since our calculations show that the annual average dose to a child's thyroid, if this child consumed milk from the nearest cow, would exceed only slightly our current limit of 5 millirem per year. The applicant will take appropriate measures through monitoring, administrative measures and/or design changes to assure that the thyroid dose to critical segments of the general population through the grass-cow-milk chain does not exceed 5 mrem/year.

11.2 Liquid Waste

11.2.1 Introduction

The liquid radioactive waste system consists of tanks, demineralizers, evaporators, miscellaneous process equipment, piping and instrumentation necessary to collect, process, monitor, store, recycle and dispose of potentially radioactive liquid wastes. The liquid waste system is divided into several subsystems. These subsystems include high purity liquid waste, low purity liquid waste, chemical waste and detergent waste. Cross-connections between the subsystems provide additional flexibility for processing the wastes by alternate methods.

Treated wastes will be handled on a batch basis as required to permit optimum control. Prior to release of any treated liquid wastes, samples will be analyzed to determine the type and amount of radioactivity in a batch. Based on the results of this analysis, the wastes will be released under controlled conditions to the cooling tower blowdown stream and then to the Cedar River, or retained for further processing.

11.2.2 System Description

High purity (low conductivity) liquid waste will be collected in the waste collector tank, and will consist of piping and equipment drains and demineralizer backwash. These wastes will be processed by filtration and ion exchange through the waste filter and waste demineralizer.

After processing, the liquid will be transferred to one of two waste sample tanks where it will be sampled. Then, if it is satisfactory for reuse, the liquid will be transferred to the condensate storage tank as makeup water. Our analysis assumed a daily input into this system of 21,000 gallons of high purity wastes. We further considered that 90% of this water will be retained for plant use and that 10% would be discharged.

Low purity (moderate conductivity) liquid wastes will be collected in the floor drain collector tank, principally from the various floor drain sumps. Processing will consist of filtration, ion exchange and subsequent transfer to the floor drain sample tank for sampling and analysis. Then, if the analysis is satisfactory, the wastes will be transferred to the condensate storage tank for reuse, or discharged. Our analysis assumed a daily input to this system of 8500 gallons of low purity wastes.

We further assumed that 70% of this water will be retained for plant use and 30% would be discharged.

Chemical wastes will be collected in the chemical waste tank, principally from decontamination, laboratory drains and cask cleaning drains. These wastes will be neutralized if required and then processed by filtration and evaporation. Evaporator bottoms will be transferred to the solid waste system. The evaporator distillate will be collected in a sampling tank for sampling and analysis. Depending on the results of the analysis, the water will be discharged or recycled for further treatment. Our analysis assumed a daily input to this system of 500 gallons of chemical waste and that all this waste will be discharged.

Detergent wastes will be collected in one of two detergent drain tanks. The source of these wastes are shop drains, personnel decontamination drains, cask cleaning drains and turbine washdown area drains. Detergent wastes will have low radioactivity concentration. They will be processed in the same manner as the chemical wastes. We have assumed a daily input of 300 gallons of detergent waste with a negligible amount of activity. In our calculations we have combined the chemical and detergent wastes.

The liquid effluent will be discharged to the cooling tower blowdown stream. Radiation monitoring equipment will automatically terminate the discharge flow if radiation levels are above a predetermined limit.

11.2.3 Evaluation of Liquid Waste Systems

Based on our evaluation of the liquid waste system, we estimate that 4 curies will be released per year, excluding tritium. This release was determined using an operating power, fission source term derived by scaling-down the source term associated with an off-gas release rate of 100,000 μCi/sec after 30 minutes holdup for a 3400 MWt reactor. Based on present operating experience, we estimate that 20 curies per year of tritium will be released from the Duane Arnold station. For comparison, the applicant estimates a yearly liquid waste release of 0.4 curies excluding tritium, based on an off-gas release rate of 50,000 microcuries per second, and a yearly tritium release of 20 curies. Our estimate for the yearly dose to an individual from the liquid waste, including drinking water ingestion, fish ingestion and immersion is 0.38 millirems.

Based on the calculated liquid release, excluding tritium being less than 5 curies, we conclude that the liquid waste system is capable of providing effluents which can be considered as low as practicable in accordance with 10 CFR Part 50; therefore the liquid waste treatment system is acceptable.

11.3 Gaseous Waste

11.3.1 Introduction

The waste gas system consists of charcoal delay beds, catalytic recombiners, heat exchangers, piping, high efficiency particulate filters, pumps and instrumentation necessary to collect, process, monitor and dispose of potentially radioactive gaseous waste.

11.3.2 Description

The primary source of gaseous radioactive waste will be the non-condensible gases removed from the main condenser by the air ejector. These gases will consist of air, hydrogen, oxygen and small volumes of radioactive gases, primarily krypton and xenon. Other sources of radioactive gases include the turbine gland seal condenser, the reactor building, turbine building and radwaste building ventilation systems and the mechanical vacuum pump used to evacuate the main condenser during startup.

Off-gases removed from the main condenser by the steam-operated air ejector will be processed through one of two catalytic recombiners in which the hydrogen and oxygen are combined to form water vapor, thereby reducing the volume of gases which must be treated. The water vapor will be condensed and removed. The non-condensible gases will be delayed for 30 minutes in a holdup pipe to allow for the decay of short-lived radioactive noble gases and activation products, and then held up for further decay in an ambient temperature charcoal system consisting of 12 charcoal beds, each of which contains 3 tons of activated charcoal, wherein xenons and kryptons will be adsorbed and delayed selectively. The residual gases will be passed through a HEPA filter and then released through a 100-meter main off-gas stack.

We calculate a delay time of 18.2 hours for krypton and 13.6 days for xenon. The applicant has calculated delay periods of 19 hours for krypton and 15 days for xenon. We calculate the yearly noble gas release from the air ejector offgas to be 27,000 curies based upon a 100,000 microcuries per second source. The applicant's number for the yearly noble gas release from the air ejector off-gas is 24,000 curies based on the release rate of 100,000 microcuries per second.

Primary steam will be used in the turbine gland seal system. Therefore the gland seal exhaust can be a source of radioactivity. These gases will be held up approximately 1.8 minutes before being exhausted into the off-gas stack without further treatment. We calculate the yearly noble gas release from this source to be approximately 3700 curies and the I-131 release to be 0.04 curie. The applicant's yearly noble gas release number from this source is 3600 curies.

During unit startup, a mechanical vacuum pump will be used to evacuate the main condenser. We assumed that the pump will operate about 10 hours per year. The exhaust gases will be discharged through the same holdup pipe into which the turbine gland seal condenser exhausts. The gases will be released through the main stack without further treatment. We estimate a noble gas release of approximately 1650 curies per year from this source.

The reactor building ventilation air will normally be discharged through the reactor building vent without treatment. If the airborne radioactivity exceeds a predetermined level, the reactor building will be isolated and its contained atmosphere will be directed at reduced flow rate through the Standby Gas Treatment System. We calculate the release of noble gases from the reactor building ventilation air to be negligible and the release of I-131 to be less than 0.01 curie per year.

The atmosphere in the drywell and suppression chambers will be purged prior to the refueling and maintenance periods. The purged gases will be discharged through the Standby Gas Treatment System, or directly to the reactor building vent if the activity is low. The expected radioactivity release from this source is insignificant.

The exhaust air from the Radwaste Building ventilation system will pass through prefilters and HEPA filters and will then be discharged through the reactor building vent.

The ventilation air flow through the Turbine Building will vary from approximately 41,000 cfm in the winter to approximately 112,000 cfm in the summer. Approximately 41,000 cfm of potentially contaminated air will be constantly exhausted from the lower areas of the turbine building to the reactor building vent. The balance of the air flow through the upper turbine building for heat removal in summer will be exhausted unfiltered through roof outlets. We calculate the noble gas released from this source to be approximately 900 curies per year and the I-131 release as 0.55 curie per year.

11.3.3 Evaluation of Gaseous Waste System

We calculate a total yearly gaseous release of approximately 33,000 curies of noble gas and 0.6 curie of I-131. We calculate the annual combined beta and gamma radiation dose due to noble gases to an individual at the site boundary to be 1.4 millirems. We further calculate that the potential dose to a child's thyroid, from radioiodine that could be ingested via the food chain, to be in excess of 5 millirems per year at a distance of 1.6 miles, where the nearest cow is presently pastured. To assure that the 5 millirem per year limit is not exceeded, the applicant will take appropriate measures through monitoring, administrative measures and/or design changes to ensure that the thyroid dose to critical segments of the general population is less than 5 millirem per year throughout the life of the plant.

11.4 Solid Wastes

Solid wastes from the plant operation will be composed primarily of spent demineralizer resins, evaporator concentrates, filter sludges and miscellaneous dry wastes such as contaminated clothing, rags and paper. The spent resins and filter sludges will be held for radioactive decay in phase separators or sludge tanks and will then be transferred to centrifuges where the excess water will be removed. The dewatered resin waste and filter sludge will then be packaged in drums. The evaporator bottoms will be placed in drums and mixed with a solidification agent. Compressible low level solid waste will be compacted by a hydraulic press. *All solid waste will be packaged and shipped offsite to a licensed burial ground in accordance with AEC and DOT regulations.*

The applicant estimates volume and activity content of waste concentrates exclusive of evaporator bottoms to be about 2200 cubic feet per year, with a total activity of about 1000 curies per year. We estimate that approximately 500 drums of spent resins, filter sludges and evaporator bottoms and 250 drums of dry and compacted waste will be shipped offsite at a total activity of approximately 1500 curies per year after 180 days of storage. Based on our evaluation, the Solid Waste Systems is acceptable.

11.5 Design

The radioactive waste treatment system will be designed in accordance with acceptable codes and standards. Tanks are designed in accordance with API Code[31] 620 or 650, or AWWA Standard D100. The other liquid waste components and piping ordered prior to July 1, 1971 are designed in accordance with ASME Section VIII Division 1 and ANSI B31.1.0.[19] The liquid waste components and piping ordered after July 1, 1971 are designed in accordance with ASME Section III, Class 3, and ANSI B 31.7, respectively. The gaseous waste components design pressure is such as to maintain the component outer wall integrity in the event of a hydrogen explosion.

The radwaste building and equipment are designed to Seismic Category II requirements. Accident analysis calculations show that failure of all liquid radwaste components will not result in offsite concentrations exceeding the limits set forth in 10 CFR Part 20. Based on our evaluation, the codes and standards, and the radwaste system design are acceptable.

11.6 Process Radiation Monitoring System

The design objective for the process radiation monitoring system is to indicate when limits of radioactivity are approached so that appropriate action may be taken where applicable. The radiation monitoring locations will include: main steamline, air ejector off-gas, plant stack, liquid waste discharge line, plant service water, reactor building closed cooling water, reactor building ventilation, and carbon bed vault area.

A high high main steamline radiation signal will initiate a reactor scram, isolate the primary containment, and shut the main steamline isolation valves. A high radiation signal in the liquid waste discharge will automatically terminate liquid waste discharge flow. A high high radiation signal in the offgas discharge will automatically terminate the offgas flow. A high radiation signal in the reactor building ventilation system will shut off the reactor building heating and ventilating system and start up the Standby Gas Treatment System. We find the process monitoring system adequate to monitor effluent discharge paths as specified in Criterion 64 of 10 CFR Part 50.

11.7 Radiological Environmental Monitoring

A pre-operational radiological environmental monitoring program has been in effect at the site since April 1971. More than two years of baseline data will be available (prior to plant start up) against which to measure and evaluate the effect of plant operation.

The monitoring program includes sampling of airborne particulates, surface water, ground water, bottom sediments, soil, vegetation, meat and poultry, milk, fish, aquatic biota, and wildlife. The analysis frequency for these samples varies from weekly to semi-annually. Airborne particulates are sampled at 16 stations which are located generally within 10 miles of the plant. In addition, thermoluminescent dosimeters are located at all air sampling locations, as well as at 32 other key locations, for the purpose of measuring ambient radiation levels. The program described above will be further defined in the Technical Specifications for the plant.

We conclude that the radiological environmental monitoring program as defined in the FSAR is acceptable.

11.8 Conclusions

Based on our model and assumptions, we calculate an expected whole body dose to an individual at the site boundary of less than 5 mrem/yr from gases and liquids. We calculate that the potential dose to a child's thyroid from the food chain to be in excess of 5 millirem per year at a distance of 1.6 miles, where the nearest cow is located. Based on our evaluation, we conclude that the liquid and solid waste treatment systems meet the requirements of "as low as practicable." We conclude that the applicant's milk monitoring program and air monitoring system will be capable of detecting I-131 concentrations equivalent to at least

5 mrem per year and the applicant's commitment to meet the 5 mrem per year limit by administrative and/or design changes is acceptable.

We also conclude that the radwaste treatment systems are designed in accordance with applicable codes and standards, and that the process and radiation environmental monitoring systems are adequate for monitoring effluent discharge paths and any associated environmental effects.

12.0 RADIATION PROTECTION

12.1 Shielding

The radiation shielding and expected personnel occupancy factors are designed to allow plant operation at the maximum calculated power level with 1% fuel defects without exceeding radiation doses permitted by 10 CFR Part 20[32] for both occupational and non-occupational personnel.

The shielding design for the Duane Arnold plant is very similar to that of previously approved boiling water reactors. We conclude on the basis of our review of the FSAR, that the shielding design is adequate to protect health and safety of the public and of the operating personnel.

12.2 Health Physics Program

The provisions for personnel monitoring, the protective equipment to be provided for use by operating and maintenance personnel, and the types of portable survey equipment and laboratory equipment available at the Duane Arnold plant are similar to that previously approved for currently operating nuclear power plants. Administrative controls and procedures are also similar. The staff discussed the applicant's health physics program at meetings and made a site visit to check the equipment, personnel, and procedures to be used at the Duane Arnold plant. The program is described in the Duane Arnold Health Physics Manual. We conclude that this program is acceptable.

13.0 CONDUCT OF OPERATIONS

13.1 Plant Organization and Staff Qualifications

Operating responsibility for the Duane Arnold Energy Center (DAEC) has been assigned to a Chief Engineer who reports to the President, Iowa Electric Light and Power Co., through the General Production Manager, and the Vice President-Engineering. An operating staff of approximately 62 full time employees is divided among four principal functional groups: operations, maintenance, technical, and administrative. The Supervisors of these groups report to the Chief Engineer through a full time Assistant Chief Engineer.

The Operations Group, under an Operations Supervisor, conducts day-to-day plant operations, fuel handling, and refueling activities. The normal shift crew complement will consist of a Shift Supervising Engineer, an Operating Engineer, a First Assistant Operating Engineer, a Second Assistant Operating Engineer, and an Auxiliaries Engineer. Each crew will include at least one Senior Reactor Operator and two Reactor Operators licensed by the AEC, in accordance with requirements to be included in the Technical Specifications.

The maintenance function is divided between the Mechanical and Electrical Maintenance Groups. The Mechanical Maintenance Group duties consist of day-by-day repairs, adjustments, equipment condition inspection, overhauls and modifications of equipment and other mechanical maintenance functions as assigned. The Electrical Maintenance Group duties consist of: maintenance of electrical equipment; modification of equipment; calibration, test, and maintenance of instruments and controls; equipment condition inspection; and other maintenance functions as assigned.

The technical staff consists of two groups: the Plant Performance Group and the Radiation Protection and Chemistry Group. The former group is headed by the Reactor and Plant Performance Engineer and has duties such as conducting tests and interpreting data logger output to determine reactor and plant performance; this group will also maintain fuel accountability. The latter group is headed by the Radiation Protection Engineer and has duties such as water treatment, waste disposal, radiation protection and shielding, radiation monitoring, and laboratory analysis.

The qualifications of the key supervisory personnel with regard to educational background, experience, and technical specialties generally meet or exceed the minimum qualifications as defined in the American National Standards Institute standard, "Selection and Training of Nuclear Power Plant Personnel," ANSI N18.1-1971.[33]

We conclude that the organization and the qualifications of the staff for operating the Duane Arnold Energy Center are acceptable.

13.2 Training

A training program and schedule for plant personnel was developed by the applicant and included major training phases conducted by the General Electric Co., in BWR Technology, BWR operator training, and specialist training for support personnel in Station Nuclear Engineering, Radiological Engineering, Nuclear Instrumentation and Control, BWR Chemistry, BWR Maintenance, and BWR Startup Testing. The schedule has provided ample opportunity for the completion of formal training elements and plant familiarization to permit full plant staff participation in the pre-operational test program. The applicant has generally committed to the execution of retraining and replacement training programs in conformance with the requirements of ANSI N18.1-1971.

We conclude that the training program for the initial plant staff should result in a sufficient number of licensed operating personnel at the time of fuel loading, and that there is reasonable assurance that continued training programs will maintain the competency of the staff. We find the training program to be acceptable.

13.3 Preparedness Plan

The applicant has formulated and submitted a Preparedness Plan which is responsive to the regulatory requirements of 10 CFR Part 50,[2] Appendix E, for plans to cope with emergencies. The plan describes the organizational structure for dealing with emergencies, and the agreements and understandings with appropriate local, State, and Federal agencies. These latter arrangements have been made with the Office of the Commissioner of Public Safety for Cedar Rapids, the office of the Sheriff of Linn County, the Linn County Health Department, the Linn County Civil Defense organization, the State of Iowa

Department of Public Safety and Department of Health, the Iowa Civil Defense Division, and the Chicago Operations Office of the AEC. The plan describes measures to be taken for a broad variety of emergency situations and includes protective action criteria for notification of offsite agencies.

Arrangements have also been made to provide for appropriate medical assistance to persons affected by onsite accidents, including those accidents having radiological effects.

Provisions for review and updating of the plan and for the conduct of periodic drills and exercises have been included.

We have reviewed the details of the Preparedness Plan submitted by the applicant and conclude that adequate arrangements have been made to cope with the possible consequences of a broad spectrum of accidents at the site including Design Basis Accidents.

13.4 Safety Review and Audit

The applicant has established a two level review process to assure that all plant operational matters, design changes, tests, and changes to approved procedures, which have a bearing on safety, are adequately assessed. An Operations Committee which is advisory to the Chief Engineer constitutes an onsite review mechanism while a second level of review is vested in a company level Safety Committee which is advisory to the General Production Manager. The Safety Committee also has an assigned responsibility for periodic audits of plant operations. These review and audit functions are generally in accord with the requirements and recommendations of ANSI N18.7-1972,[34] "Standard for Administrative Controls for Nuclear Power Plants," and their functions and responsibilities will also be incorporated in the Administrative Controls section of the Duane Arnold plant Technical Specifications.

We have concluded that the applicant's review and audit structure for plant operations is acceptable.

13.5 Plant Procedures and Records

Plant operations are to be performed in accordance with detailed written and approved procedures. These procedures will include systems check lists; startup, normal operation, and shutdown of major equipment items; systems, and integrated plant operation, alarm response procedures; surveillance and testing, refueling, radiation control; and abnormal conditions, emergency, and administrative procedures.

Records will be maintained to assure a fully documented history of facility operations. Specific requirements in these areas will be incorporated in the Administrative Controls section of the Technical Specifications.

We conclude that the provisions for preparation, review, approval, and use of written procedures and the generation and control of plant operating records are satisfactory.

13.6 Industrial Security

The applicant has submitted a copy of his Industrial Security Plan as proprietary information pursuant to Section 2.790 of the

Commission's regulations. The plan is adequately responsive to AEC guidance, including Safety Guide 17[35], and provides reasonable assurance that adequate provisions have been made by the applicant to prevent or inhibit a wide range of potential acts of industrial sabotage.

14.0 INITIAL TESTS AND OPERATION

The applicant has described a test program which will be performed to assure that the Duane Arnold Energy Center is capable of withstanding the accidents and transients analyzed in the Final Safety Analysis Report and in the design bases for the plant. This test program proceeds from component and systems acceptance and pre-operational testing through fuel loading, initial criticality, and power escalation which evaluates nuclear, process, and safety features performance at various power levels up to full power. This test program generally incorporates the components, systems, and tests described in the two guides referenced in 10 CFR Part 50,[2] Section 50.34(b)(6)(iii).

Administrative procedures have been prepared to control the test program. These procedures provide for the preparation of detailed written test procedures and instructions by Bechtel and General Electric, and review and approval by the applicant's Engineering, Quality Assurance, and Production Department personnel. Conduct of the test program is the responsibility of the applicant's Production Department (plant staff) with technical direction and coordination provided by appropriate vendor personnel.

Beginning with initial fuel loading, the applicant will observe all Technical Specification requirements including those related to staffing, even though certain personnel are not expected to be examined for operators' licenses until after commercial operation actually begins. The numbers and qualifications of supervisory, technical, and senior operating personnel on the plant staff are adequate to meet the Technical Specification requirements during this period.

We conclude that the Initial Tests and Operations program for the startup of the Duane Arnold Energy Center are acceptable.

15.0 ACCIDENT ANALYSIS

15.1 Abnormal Operational Transients

We have evaluated the applicant's analyses of various abnormal operational transients. The events that characterize these transients have been described in FSAR Section 14 and in Amendments 7, 9, and 11 as responses to questions 14.6 through 14.14. These transients include such events as process system control malfunctions, inadvertent control rod withdrawal, turbine trip, loss of electrical load, and variations in operating parameters. We have reviewed the results of the applicant's analyses of these events and conclude that the design of the facility, including the protection and control systems, is such that the occurrence of such transients would not result in damage either to the fuel or to the primary coolant boundary. Consequently, the occurrence of these abnormal transients would not lead to a significant release of fission products to the environs.

15.2 Design Basis Accidents

15.2.1 Descriptions, Assumptions, and Analysis

We and the applicant have evaluated a broad spectrum of accidents that might result from postulated failures of equipment, or their maloperation. Four highly unlikely accidents (design basis accidents) that are representative of the spectrum of types and physical locations of postulated causes and that involve the various engineered safety features systems have been analyzed in detail. The calculated consequences of these design basis accidents exceed those of all other accidents considered and are the same as those analyzed for previously licensed BWR plants. The design basis accidents analyzed were: (1) control-rod-drop, (2) refueling, (3) steam-line-break, and (4) loss-of-coolant accidents. Our evaluation of these accidents shows that the calculated doses resulting from these postulated accidents are well within the 10 CFR Part 100 guideline values. The results of this analysis are shown in Table 15.1 using the accident assumptions given in Table 15.2. Our analysis also shows that the control room design is such that the exposure guidelines of General Design Criterion 19 are met and therefore the computed dose to operating personnel during postulated design basis accidents is acceptable.

15.2.2 Control Rod Drop Accident

The analysis techniques for this accident are being revised by the General Electric Company. GE Topical Reports NEDO - 10527[20] and its Supplement 1, "Rod Drop Accident Analysis for Large BWR's," dated March 1972 and July 1972, respectively, were submitted to the Regulatory Staff. Supplement 1 presents the GE analysis of the rod drop accident for the Browns Ferry and Zimmer class of BWR's which uses gadolinium for axial power shaping and multiple enrichment fuel designs. The Duane Arnold Nuclear design is similar to that for Browns Ferry; hence, the current staff review of other applicant or

TABLE 15.1

POTENTIAL OFFSITE DOSES DUE TO DESIGN BASIS ACCIDENTS

	Two Hour Exclusion Boundary (440 meters)		Course of Accident Low Population Zone (9656 meters)	
	Thyroid (Rem)	Whole Body (Rem)	Thyroid (Rem)	Whole Body (Rem)
Loss of Coolant	32	2	98	3
Fuel Handling	<1	<1	<1	<1
Steam Line Break	37	<1	2	<1
Control Rod Drop	23	1	4	<1

TABLE 15.2

ASSUMPTIONS USED FOR STAFF ANALYSIS OF DESIGN BASIS ACCIDENTS

A. Loss-of-Coolant Accident Assumptions

1. Power Level of 1658 MWt.
2. Safety Guide No. 3 assumptions were used for evaluating the potential radiological consequences.
3. Containment leak rate of 2.0% per day.
4. Charcoal filter efficiency of 90% for elemental iodine, and 70% or organic iodine.
5. HEPA filter efficiency of 90% for particulate iodine.
6. 100 meter stack release point with Safety Guide No. 3 meteorological conditions as modified by onsite meteorological and terrain elevation data.
7. If the leak rate is zero and the CAD system is used, the doses given in Table 15.1 for this accident would be less than 30 rem (thyroid) for the LPZ course of accident.

B. Refueling Accident Assumptions

1. Rupture of 111 fuel rods.
2. All gap activity in the rods, assumed to be 10% of the noble gases and 10% of the iodine (with a peaking factor of 1.5) is released to the pool water.
3. The accident occurs 24 hours after shutdown.

4. 99% of the iodine is retained in the pool water.

5. Charcoal filter iodine removal efficiency of 90% for elemental iodine and 70% for organic iodines.

6. Elevated release as in the LOCA analyses.

7. The meteorological conditions assumed are the same as described for the 0-2 hour period following a loss-of-coolant accident.

C. Control-Rod-Drop Accident Assumptions[(1)]

1. The accident occurs due to a 2.5% ΔK control rod drop.

2. 330 fuel rods are damaged.

3. Peaking factor = 1.50.

4. 100% of the noble gases and 50% of the iodines are released from the fuel.

5. A reduction factor of 10 is allowed for iodine passing through the primary system water.

6. A plate-out factor of 2 is allowed for iodine in the turbine and condenser.

7. High radiation is detected in the steamline signaling the vacuum pump to stop and the isolation valves to close. (5 second valve closure time.)

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(1) These assumptions may be modified in the near future to conform with the results of our study and analysis of the control rod drop accident described in paragraphs 7.6 and 15.2.2 of this safety evaluation report. Preliminary results of the rod drop accident using the analysis modification recommended by our consultant, Brookhaven National Laboratory, and assuming the Rod Sequence Control System to be operable results in approximately 600 fuel rods perforation and increases the calculated doses from this accident by a factor of two, which is still well below the 10 CFR Part 100 guidline values.

8. All of the activity is contained by the turbine and condenser.
9. A constant leak rate of 0.5% per day from the turbine and condenser is assumed.
10. The total accident duration is 24 hours.
11. Safety Guide No. 3 ground level release with credit for a wake factor.

D. Steam-Line-Break Accident Assumptions

1. Accident occurs at full power level of 1658 MWt.
2. Safety Guide No. 5[36] assumptions.
3. Steamline isolation valve closes in 5 seconds.
4. Release of all activity occurs within two hours at 30 meters height.
5. Coolant concentrations are based on 1.0 Ci/Sec gaseous release rate (20 μc/cc total iodines).

vendor submitted information on the Browns Ferry and Peach Bottom Units 2/3 facilities apply also to DAEC. Following our approval of a revised analytical model and our approval of control rod system modifications, the applicant (IELP) will provide the appropriate documentation amending the DAEC operating license application and will also install the AEC approved control modifications. The applicant's commitment is discussed in FSAR Amendment 9 (response to question 14.15) and also in paragraph 7.6 of this Safety Evaluation. We find this plan of action to be acceptable and has reasonable assurance for orderly completion prior to fuel loading.

15.3 Containment Purge Dose, Post-LOCA

It has been calculated by the applicant that purging of the containment atmosphere after 35 days elapsed time in the post-LOCA period may be needed in order to prevent containment pressure buildup. We have made calculations of the exclusion area boundary dose resulting from a continuous purge of containment atmosphere beginning after 30 days, post-LOCA with purge rates equivalent to 2% and 5% of a containment volume of 2.04×10^5 ft^3, with a χ/Q of 1.35×10^{-6} sec/m^3, thermal power level of 1658 MW, and a filter efficiency for radioiodines of 90% elemental iodine and 70% organic iodines. The resulting doses from both purge rates (2.82 cfm and 7.1 cfm) are less than our current criteria of 1/10 the 10 CFR Part 100 values.

15.4 Liquid Radwaste Tanks Failure

We analyzed a postulated failure of the liquid radwaste tanks and release of their contents to the soil surrounding the Category II seismic design Radwaste Building. Using the concentrations of specific nuclides in the radwaste liquid inventory, conservative values of soil permeability, and dispersion, dilution and transit time to the Cedar River, we have determined that this remote event results in concentrations in water of specific radionuclides in the vicinity of Cedar Rapids wells to be less than the guidelines of 10 CFR Part 20, "Concentrations for Drinking Water."

16.0 TECHNICAL SPEFICATIONS

The Technical Specifications of a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the AEC. The proposed Technical Specifications for the Duane Arnold Energy Center will be similar in scope and content of recently licensed BWR's and are essentially complete. We have held meetings with the applicant to discuss their contents and some modifications to the proposed Technical Specifications have been suggested both by the staff and the applicant to more clearly describe the allowed conditions for plant operation. The finally approved Technical Specifications will be included as part of the operating license. Included are sections covering safety limits and limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls. On the basis of our review, we will assure that normal plant operation within the limits of Technical Specifications will not result in potential offsite exposures in excess of 10 CFR Part 20 limits and/or our guidance on meeting the "as low as practicable" releases of radioactivity. Furthermore, the limiting conditions of operation and surveillance requirements will assure that necessary engineered safety features for continued plant operation will be available in the event of malfunctions within the plant.

17.0 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

17.1 ACRS Construction Permit Letter

In its letter dated December 18, 1969 to the Commission, the Advisory Committee on Reactor Safeguards indicated certain matters would require resolution between the applicant and the regulatory staff. Each of these matters are discussed in this Safety Evaluation report. The matters indicated in the ACRS letter and reference in this report are: (1) Solution cavities (see paragraph 2.5.2), (2) Emergency cooling water system (see paragraph 9.2.3), (3) Main steam line inspection (see paragraph 10.3), (4) Main steam line isolation valves seal system (see paragraph 6.2.6), (5) Instrument line failure (see paragraph 6.2.4), (6) ATWS (see paragraph 7.5), (7) Combustible gas control (see paragraph 6.2.5), (8) Fuel cask drop (see paragraph 9.1.2).

The above matters were specifically identified in the ACRS construction permit letter. Other problems related to boiling water reactors which have been identified by the regulatory staff and ACRS are covered in the organization of this Safety Evaluation report. The applicant has identified and discusses those matters identified by the ACRS construction permit letters in Appendix H of the FSAR.

17.2 ACRS Operating License Letter

The report of the ACRS on this project for the operating license review will be placed in the Commission's Public Document Room and will be published in a supplement to this evaluation report.

18.0 QUALITY ASSURANCE

18.1 General

The description of the Quality Assurance (QA) Program for the operation of the Duane Arnold facility is contained in Appendix D to the FSAR, supplemented by the QA information contained in Amendments 1, 7, 10, and 11 to the application which were filed in response to our requests for additional information. Our evaluation of the QA Program is based on a review of this information and related discussions with the applicant to determine the ability of Iowa Electric and Power Company (IELP) to comply with the requirements of Appendix B to 10 CFR 50.[8]

18.2 Organization and Program

Responsibility and authority to define and direct the Quality Assurance Program is assigned by IELP to the Engineering Vice-President, who is also the Duane Arnold Project Manager. IELP's Quality Assurance Manager reports to the Project Manager and is responsible for the administration of the QA Program. The QA Manager also has direct communication with IELP's President, independent of the remainder of the Project Group, for quality matters.

The QA organization for the Duane Arnold Energy Center, (DAEC), for the period through preoperational testing will be maintained the same as during the design and construction phase.

During preoperational testing, procedures and tests on safety-related systems will be reviewed and auditied by the QA staff. Beginning with fuel loading and plant operations activities, plant staff QC personnel, reporting directly to onsite plant management, will be assigned on a full time basis. The duties of these QC personnel will include work inspection, verification of procedure implementation, plant quality planning, specification monitoring, materials and equipment control, and records monitoring. The QA Manager, who is located at IELP's headquarters office, will also have full time QA representatives at the plant site.

Based on our review of IELP's plans for the staffing and organization of the QA function, we conclude that IELP has provided an acceptable organization, adequate independence, and proper management involvement for the remainder of design, construction, and preoperational testing efforts, and for operational phase activities.

18.3 Audit Program

The QA Manager and his headquarters and onsite staff will audit safety related activities over the service life of the plant. This audit function will include audit of offsite Engineering Support Group activities and audit of onsite operational activities. The onsite activities to be audited, and the audit frequency, has been delineated by IELP. The QA organization will also provide a review of the requisition and vendor qualification for all safety related material, equipment, and services for maintenance, modification, repair, rework, and design changes.

Neither site nor headquarters QA personnel will be members of either the Plant Safety or Operating Committee who are chartered to call on IELP's QA staff as required. However, IELP requires QA personnel to audit the functioning of these committees and the implementation of their decisions.

We conclude that the audit program described in the application is adequate to provide acceptable management attention to quality related activities during the operational phase and meets the provisions of Appendix B to 10 CFR Part 50.[8]

18.4 Quality Control and Quality Assurance for Fuel Manufacturing and Performance

We have evaluated IELP's plan for review of reactor fuel to assure its long term integrity. The applicant has described the design and manufacturing features of the DAEC fuel which are intended to minimize possible fuel failures. These include restriction of possible moisture and hydrocarbon contaminants in the UO_2 pellets, inclusion of a hydrogen getter device in each fuel rod, chamfered pellet ends, and shorter pellet length. These features of fuel design represent current state of the art actions that would minimize fuel failures during plant operation. Although we consider such actions appropriate for minimizing fuel failures, we will continue our surveillance of nuclear fuel performance to evaluate its operation. IELP's actions in assuring adequacy of purchased fuel are presently limited to the audit of the fuel manufacture. We conclude that IELP's QA Program for fuel is acceptable.

18.5 Quality Assurance During Station Operation

IELP has committed to implement the requirements of Appendix B to 10 CFR 50[8] and the provisions of AEC Safety Guide 33[37] during the operational phase of the Duane Arnold Energy Center. This is acceptable.

18.6 Conclusions

Based on the applicant's commitments and on our review of the QA Program described in the FSAR and related amendments, we have concluded that the description of the QA Program complies with the requirements of Appendix B to 10 CFR 50,[8] and is acceptable for Duane Arnold facility operation.

19.0 COMMON DEFENSE AND SECURITY

The applicant states that the activities to be conducted would be within the jurisdiction of the United States and that all of its directors and principal officers including those of the other participating companies are United States citizens. We find nothing in the application to suggest that the applicant, or the other participating companies are owned, controlled, or dominated by an alien, a foreign corporation or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the regulations. The applicant will obtain fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposes is involved. For these reasons, and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

20.0 FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to financial data and information required to establish financial qualifications for an applicant (and co-applicants) for a facility operating license are presented in paragraph 50.33(f) and Appendix C of 10 CFR Part 50. We have reviewed the financial information presented in FSAR Amendment No. 10-A and its supplement and conclude that the applicant and co-applicants possess or can obtain the necessary funds to operate the Duane Arnold Energy Center and if necessary, permanently shut down the facility and maintain it in a safe shutdown condition. A detailed discussion of the basis for our conclusion is presented in Appendix D to this Safety Evaluation.

21.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

21.1 Financial Protection and Indemnity Requirements

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections), the Commission has issued regulations in 10 CFR Part 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licensees for facilities such as power reactors licensed under 10 CFR Part 50.

21.2 Preoperational Storage of Nuclear Fuel

The Commission's regulations in Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also to be the holder of a license under 10 CFR Part 70 authorizing the ownership and possession for storage only of special nuclear materials at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR Part 50), shall, during the interim storage period prior to licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement with the Commission. Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective data of the 10 CFR Part 70 license. No license authorizing the ownership and possession, for storage only, of special material at the reactor construction site for future use as fuel in the reactor will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

21.3 Operating License

Under the Commission's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished, and an indemnity agreement covering such operation (as distinguished from, for example, preoperational fuel storage only) has been executed.

Accordingly, no license authorizing operation of the Duane Arnold Energy Center will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

22.0 CONCLUSIONS

Based on our evaluation of the application as set forth above, we have concluded that:

1. The application for facility license filed by the applicant, dated November 4, 1968, as amended (PSAR with 15 amendments and FSAR with 11 amendments) comply with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter 1; and
2. The construction of the Duane Arnold Energy Center (the facility) has proceeded, and there is reasonable assurance that it will be complete, in conformity with Provisional Construction Permit No. CPPR-70, the application as amended, the provisions of the Act, and the rules and regulations of the Commission, and
3. The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and
4. There is reasonable assurance, assuming satisfactory completion of our review of those items which we have elected to defer, that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Part 1; and

5. The applicant is technically and financially qualified to engage in the activities authorized by an operating license in accordance with the regulations of the Commission set forth in 10 CFR Part 1; and

6. The issuance of an operating license for DAEC will not be inimical to the common defense and security or to the health and safety of the public.

Prior to any public hearing on the matter of the issuance of an operating license to the applicant for the Duane Arnold Energy Center, the Commission's Directorate of Regulatory Operations will prepare a supplement to this Safety Evaluation which will deal with those matters relating to the status of construction completion and conformance of that construction to the construction permit and the application. Before an operating license can be issued to the applicant the facility must be completed in conformity with the construction permits, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power level must be verified by the Commission's Directorate of Regulatory Operations prior to issuance of a license. Further, before an operating license is issued, the applicant will be required to satisfy the applicable provisions of 10 CFR Part 140.

APPENDIX A

CHRONOLOGY

REGULATORY REVIEW OF

IOWA ELECTRIC LIGHT AND POWER COMPANY'S

DUANE ARNOLD ENERGY CENTER

June 22, 1970	Construction Permit CPPR-70 issued to Applicant.
November 1971	Applicant submitted Revised Environmental Report
November 26, 1971	Determination Not to Suspend Construction Activities at the Duane Arnold Energy Center Authorized Pursuant to CPRR-70 Pending Completion of NEPA Environmental Review
March 1, 1972	Applicant tendered amended application for OL with six copies of the FSAR for distribution and start of the AEC's first three weeks preliminary safety review.
March 13, 1972	Letter to applicant requesting information on blowdown forces in torus.
March 28, 1972	Meeting with applicant to discuss results of the preliminary review of the FSAR and to hear Chicago, Bridge and Iron Co. discuss the repair of reactor vessel nozzles.
April 5, 1972	Meeting with applicant to discuss improvements in the FSAR prior to formal submission of an application for OL.
April 14, 1972	Meeting with applicant and tour of the site by Licensing Project Manager and Technical Review expert on piping restraint design.

May 8, 1972	Applicant formally submitted amended application for OL with FSAR and Amendment No. 1.
June 8, 1972	Meeting with applicant to discuss emergency plan, industrial security plan, training of operators, operating procedures, review of environmental matters, and the safety review schedule.
June 21, 1972	Letter to applicant requesting additional information for safety review.
June 29, 1972	Meeting with applicant to discuss the instrumentation and control system, the electrical system, and the review schedule.
July 5, 1972	Amendment No. 1 to the DAEC Environmental Report is submitted by applicant.
July 7, 1972	Letter to applicant requesting additional information for safety review.
July 10, 1972	Applicant filed Amendment No. 2 to the FSAR.
July 10, 1972	Applicant submitted information on blowdown forces in torus.
July 12, 1972	Applicant submitted information on radiological release source terms.
July 20 & 21, 1972	Two day meeting with applicant to discuss site analysis, hydrology, meteorology, structural engineering design, code classifications, and thermal-hydraulic analysis of core.
July 27 & 28, 1972	Two day meeting with applicant to discuss pipe whip restraint design and installation, recirculation pump seizure, RV vibration testing, seismic design, containment isolation, containment

	atmospheric dilution system, effluent treatment, the preparedness plan, and the industrial security plan.
July 31, 1972	Applicant filed Amendment No. 3 to the FSAR.
August 7, 1972	Letter to applicant requesting additional information for safety review.
August 11, 1972	Letter to applicant requesting additional information for safety review.
August 15, 1972	Applicant submitted information amending response (7/12/72) on source terms.
August 17, 1972	Letter to applicant requesting additional information for safety review.
August 30 & 31, 1972	Two day meeting with applicant to review the instrumentation and electrical systems.
September 6 & 7, 1972	Two day meeting with applicant to review the instrumentation and electrical systems and the industrial security plan.
September 15, 1972	Applicant submitted Amendment No. 4 to the FSAR.
September 16, 1972	Applicant submitted supplement to the DAEC On-Site Meteorological Data.
September 18, 1972	Letter to applicant requesting additional information for safety review.
September 28, 1972	Initial meeting with applicant to discuss the Technical Specifications.

September 29, 1972	Applicant submitted Amendment No. 5 to the FSAR.
October 3, 1972	Letter to applicant with statements of requirements.
October 3, 1972	Amendment No. 2 to the DAEC Environmental Report is submitted by applicant.
October 5 & 6, 1972	Two day meeting with applicant to review the P & I diagrams of electrical systems and instrumentation and control systems.
October 6, 1972	Applicant submitted Amendment No. 6 to the FSAR.
October 17, 1972	Letter to applicant with position statements.
October 20, 1972	Applicant submitted Amendment No. 7 to the FSAR.
October 24 & 25, 1972	Licensing project manager and staff hydrologist visited site to discuss and observe progress being made on safety review and facility construction.
November 1, 1972	Letter from applicant with response our statement of requirements dated 10/3/72.
November 3, 1972	Applicant submitted Amendment No. 8 to the FSAR.
November 13, 14 & 15	Site visit by staff electrical engineer to review and evaluate the installed electrical and instrumentation systems.
November 20, 1972	Letter to applicant with statements of requirements.
November 20, 1972	Letter to applicant requesting additional information on fuel densification.

November 20, 1972	Draft Environmental Statement is distributed.
November 21, 1972	Applicant submitted Amendment No. 9 to the FSAR.
November 30, 1972	Letter to applicant requesting financial information.
December 11, 1972	Applicant submitted Amendment 10 to the FSAR.
December 14 & 15, 1972	Two day meeting with applicant to discuss matters bearing on preparation of the Safety Evaluation Report (SER).
December 15, 1972	Letter to applicant requesting information concerning a postulated break of the main steamline outside containment.
December 18, 1972	Letter from applicant with description of plan to design, test, and install a MSL-IV seal system.
December 20, 1972	ACRS Subcommittee site visit.
December 21, 1972	Letter from applicant with financial information in response to our request dated 11/30/72: Amendment 10A.
January 2, 1973	Letter from applicant with supplement to Amendment 10A
January 8, 1973	Applicant submitted Amendment 11 to the FSAR

APPENDIX B

BIBLIOGRAPHY
(Documents referenced in the Safety Evaluation Report)

1. Final Safety Analysis Report with Amendments 1 through 11 for the Duane Arnold Energy Center, with dates from May 1972 through January 1973.

2. United States Atomic Energy Commission Rules and Regulations, 10 CFR Part 50, Licensing of Production and Utilization Facilities.

3. United States Atomic Energy Commission Rules and Regulations, 10 CFR Part 2, Rules of Practice.

4. United States Atomic Energy Commission Rules and Regulations, 10 CFR Part 9, Public Record.

5. IEEE Criteria for NPP Protection Systems (IEEE 279) August, 1968.

6. United States Atomic Energy Commission Rules and Regulations, 10 CFR Part 100, Reactor Site Criteria.

7. 10 CFR Part 50, Appendix A, General Design Criteria for Nuclear Power Plants (GDC).

8. 10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

9. Safety Guide #23, Onsite Meteorological Programs.

10. Hydrometeorological Branch, Natural Oceanic and Atmospheric Administration (NOAA), "Probable Maximum Precipitation Over South Platte River, Colorado and Minnesota River, Minnesota," January 1969 Hydrometeorological Report No. 44.

11. U. S. Corps of Engineers, EM 1110-2-1405, "Flood Hydrology Analyses and Computations."

12. Safety Guide #26, Quality Group Classifications and Standards.

13. ASCE Paper No. 3269, "Wind Forces on Structures" Structural Division, Transactions American Society of Civil Engineers, 1961.

14. Safety Guide #12, Instrumentation for Earthquakes.

15. ACI 318-63, American Concrete Institute, "Building Code Requirements for Reinforced Concrete."

16. AISC Specifications, AISC Manual of Steel Construction.

17. Safety Guide #20, Vibration Measurements on Reactor Internals.

18. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Sections III, VIII, IX and XI.

19. USAS B 31.1.0-1967, Power Piping.

20. General Electric Topical Report NEDO-10527, "Rod Drop Accident Analysis for Large BWR's" and Supplement No. 1.

21. General Electric Topical Report APED 5286, "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors" September 1966.

22. General Electric Topical Report NEDO-10320, "The General Electric Pressure Suppression Containment Analytical Model" May 1971.

23. Safety Guide #11, Instrument Lines Penetrating Primary Reactor Containment.

24. Safety Guide #7, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident.

25. Safety Guide #1, Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps.

26. General Electric Topical Report NEDO-10349, "Analysis of Anticipated Transients Without Scram" March 1971.

27. Safety Guide #6, Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems.

28. Safety Guide #9, Selection of Diesel Generator Set Capacity for Standby Power Supplies.

29. IEEE Criteria for Class IE Electrical Systems for Nuclear Power Generating Stations (IEEE-308).

30. Safety Guide 27, Ultimate Heat Sink.

31. API Code 620, or 650, Standard for Design and Construction of Low Pressure Storage Tanks or Atmospheric Storage Tanks.

32. United States Atomic Energy Commission Rules and Regulations, 10 CFR Part 20, Standards For Protection Against Radiation.

33. ANSI N 18.1-1971, "Selection and Training of Nuclear Power Plant Personnel."

34. ANSI N 18.7-1972" Standard for Administrative Controls for Nuclear Power Plants."

35. Safety Guide 17, Protection Against Industrial Sabotage.

36. Safety Guide #5, Assumptions Used For Evaluating the Potential Radiological Consequences of a Steam Line Break.

37. Safety Guide #33, Quality Assurance Program Requirements (Operations).

APPENDIX C

REPORT ON THE SITE SEISMICITY FOR THE DUANE ARNOLD ENERGY CENTER, IOWA

At the request of the Division of Reactor Licensing of the Atomic Energy Commission, the Seismology Division of the Coast and Geodetic Survey has evaluated the seismicity of the area around the proposed Duane Arnold Energy Center near Palo, Iowa, and has reviewed a similar analysis presented by the applicant in the "Preliminary Safety Analysis Report." The applicant's report on the site seismicity is adequate for the determination of the seismic factors for construction at this site.

The geologic evaluation indicates that this site is located in the northern part of the Interior Lowlands Tectonic Province of the Central Stable Region of North America. The structure of the Precambrian crystalline rocks, is poorly known. Consequently, historic earthquake activity cannot be associated with any specific tectonic structure. Therefore, it must be assumed that earthquakes characteristic of the region could occur near the plant site. There are no known active faults or other recent geologic structures that could be expected to localize seismicity in the immediate vicinity of the site.

Review of the seismicity of the area indicates that there have been no earthquake epicenters located within 75 miles of the proposed plant site. Moderate earthquakes, probably associated with the Nemaha uplift, the Sandwich Fault, and the St. Genevieve fault zone are sufficiently far from the site to have only minor influence on the seismic evaluation. In consideration of other geologic structures such as the Thurman-Wilson fault zone, and the postulated faults located 10 and 17 miles from the plant site, there is no record of seismic activity associated with these features.

Major earthquake regions, such as New Madrid, Missouri, are at least 400 miles from the proposed plant site and therefore do not have a significant affect on the determination of the seismic factor. However, within the tectonic region in which the proposed plant is to be located, there have been Intensity V events and it must be assumed that similar events could occur in the vicinity of the proposed plant.

Therefore, as a result of the review of the seismological and geological characteristics of the area, the Coast and Geodetic Survey recommends that an acceleration of 0.06g, resulting from an Intensity V (MM) earthquake, would be adequate for representing seismic disturbances likely to occur within the lifetime of the facility. Also, the Survey recommends that an acceleration of 0.12g, resulting from an Intensity VI (MM) earthquake, would be adequate for representing the ground motion from the maximum earthquake likely to affect the site. It is believed that these values would provide an adequate basis for designing protection against the loss of function of components important to safety.

U. S. Coast and Geodetic Survey
Rockville, Maryland 20852

October 24, 1969

APPENDIX D

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

December 29, 1972

W. R. Butler, Chief
BWR Projects Branch #1, Licensing
THRU: W. E. Campbell, Asst. Contr. for Accounting

IOWA ELECTRIC LIGHT AND POWER CO., ET AL: DUANE ARNOLD ENERGY CENTER, DOCKET NO. 50-331

Enclosed is my financial testimony on the subject matter. The testimony has been prepared for inclusion in the main body of the staff's Safety Evaluation with a financial analysis of each participant attached as an appendix.

Ray L. Carroll
Raymond L. Carroll
Staff Accountant
Accounting Procedures Branch, OC

FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to the financial data and information required to establish financial qualifications for applicants for an operating license are 10 CFR 50.33(f) and 10 CFR 50, Appendix C. The basic application of Iowa Electric Light and Power Company, and co-applicants Central Iowa Power Cooperative, and Corn Belt Power Cooperative, Amendment No. 10-A, and the accompanying certified annual financial statements of the applicant and co-applicants provide the financial information required by the Commission's regulations. This information includes the estimated annual costs of operating the Duane Arnold Energy Center for the first five years of operation plus the estimated cost of permanently shutting down the facility and maintaining it in a safe shutdown condition.

Our evaluation of the financial data submitted by the applicant and co-applicants, summarized below, provides reasonable assurance that they possess or can obtain the necessary funds to meet the requirements of 10 CFR 50.33(f) to operate the Duane Arnold Energy Center, and if necessary permanently shut down the facility and maintain it in a safe shutdown condition.

The applicant and co-applicants each have an undivided ownership interest in the station as follows: Iowa Electric Light and Power Company - 70%, Central Iowa Power Cooperative - 20%, and Corn Belt Power Cooperative - 10%. They have agreed to share cost of operation, cost of permanent shutdown, cost to maintain the shutdown facility in a safe condition, and power produced by the unit in the ratio of their respective ownership interests.

The Duane Arnold Energy Center will be used to augment the applicant's and co-applicants' present electrical generating capacities. The funds to be provided by each applicant or co-applicant to meet its share of operating costs, permanent shutdown costs, and costs to maintain the shutdown facility in a safe condition will be derived by each applicant from its overall operations. Estimated annual costs to operate the unit for the first five years are presently estimated by the applicants to be (in millions of dollars) $27.6, $26.1, $25.5, $25.0, and $25.0 in that order. These costs include amounts for operation and maintenance, fuel, insurance, labor and applicable overheads, property taxes, material and supplies, and depreciation. In addition, the applicants estimate the cost of permanently shutting down the facility at the conclusion of its useful life will be $9.8 million, based on 1972 dollars. It is estimated that an annual cost of $200,000 (in 1972 dollars) will be incurred to maintain the facility in a safe shutdown condition. The anticipated kilowatt hours to be generated at the Duane Arnold Energy Center, if priced at current average revenue per kwh of the respective applicants, would yield aggregate revenue of $59 million to $71 million each year from 1974 through 1978.

The information contained in Iowa Electric Light and Power Company's (Iowa) calendar year 1971 financial report indicates that operating revenues for 1971 totaled $87.6 million; operating expenses were $75.5 million, of which $7.7 million represented depreciation. The interest on long-term debt was earned 2.3 times; and the net income for the year was $8.2 million, of which $6.8 million was distributed as dividends to stockholders and the remainder of $1.4 million was used in the business. As of December 31, 1971, the Company's assets totaled $281.6 million, most of which was invested in utility plant ($248.9 million); retained earnings amounted to $33.2 million. Financial ratios computed from the 1971 statements indicate an adequate financial condition, e.g., long-term debt to total capitalization - .52, and to net utility plant - .44; net plant to capitalization - 1.16; the operating ratio - .86; and the rates of return on common - 9.1%, on stockholders' investment - 7.9%, and on total investment - 5.3%. The record of Iowa's operations over the past 4 years reflects that operating revenues increased from $64.1 million in 1967 to $87.6 million in 1971; net income increased from $7.4 million to $8.2 million; and net investment in plant from $161.2 million to $248.9 million; while the number of times long-term interest was earned declined from 3.4 to 2.3. Moody's Investors Service rates the Company's first mortgage bonds as Aa (high quality). The Company's current Dun and Bradstreet credit rating is 5A1.

The information contained in Central Iowa Power Cooperative's (Central) calendar year 1971 financial report indicates that operating revenues for 1971 totaled $7.7 million, and operating expenses were $6.7 million, of which $1.0 million represented depreciation. Interest on long-term debt was earned 2.7 times; and the net margin for the year was $.7 million. Central is exempt from Federal income tax. The rate which Central charges to members is year by year based upon actual cost of doing business, plus any margins approved by the board of directors. As of December 31, 1971, Central's assets totaled $44.7 million, most of which was invested in utility plant ($37.1 million); member equities amounted to $4.9 million. Financial ratios computed from the 1971 statements indicate an adequate financial condition, e.g., long-term debt to total capitalization - .89, and to net utility plant - 1.03; net plant to capitalization - .86; the operating ratio - .87; and the rate of return on members equity - 14.3%, and on total investment - 2.5%. The record of Central's operations over the past 4 years reflects that operating revenues increased from $3.9 million in 1967 to $7.7 million in 1971; net margin increased from $.2 million to $.7 million; and net investment in plant from $15.9 million to $37.1 million. Central's current Dun and Bradstreet rating is 3A1.

The information contained in Corn Belt Power Cooperative's (Corn Belt) calendar year 1971 financial report indicates that operating revenues for 1971 totaled $6.6 million; operating expenses were $6.3 million, of which $1.0 million represented depreciation. The interest on long-term debt was earned 1.0 time; and net margin for the year was a deficit of $9 thousand. As of December 31, 1971, Corn Belt's assets totaled $36.7 million, most of which was invested in utility plant ($31.6 million); members equity amounted to $4.3 million. Financial ratios computed from the 1971 statements indicate an adequate financial condition, e.g., long-term debt to total capitalization - .87; and to net utility plant - .95; net plant to capitalization - .92; the operating ratio - .95; and the rate of return on members equity - none, and on total investment - 1.4%. The record of Corn Belt's operations over the 4 years since 1967 reflects that operating revenues increased from $5.7 million in 1967 to $6.6 million in 1971; net margin decreased from $.7 million to $.5 million in 1970 and to a loss of $9 thousand in 1971. Net investment in plant increased from $21.5 million to $31.6 million; while the number of times long-term interest was earned declined from 2.5 to 1.0. Corn Belt's current Dun and Bradstreet credit rating is 3A2.

A copy of our financial analysis of each organization reflecting these ratios and other pertinent financial data is attached as an appendix.

IOWA ELECTRIC LIGHT AND POWER CO.
FINANCIAL ANALYSIS
DOCKET NO. 50-331

	(dollars in millions) Calendar Year Ended December 31		
	1971	1970	1969
Long-term debt	$ 110.4	$ 111.2	$ 96.5
Utility plant (net)	248.9	203.2	179.8
Ratio - debt to fixed plant	.44	.55	.54
Utility plant (net)	248.9	203.2	179.8
Capitalization	214.3	192.1	170.7
Ratio of net plant to capitalization	1.16	1.06	1.05
Stockholders' equity	103.9	80.9	74.2
Total assets	281.6	230.6	205.2
Proprietary ratio	.37	.35	.36
Earnings available to common equity	6.4	6.3	5.8
Common equity	70.6	57.6	55.9
Rate of earnings on common equity	9.1%	10.9%	10.4%
Net income	8.2	7.2	6.7
Stockholders' equity	103.9	80.9	74.2
Rate of earnings on stockholders' equity	7.9%	8.9%	9.0%
Net income before interest	15.0	13.0	11.2
Liabilities and capital	281.6	230.6	205.2
Rate of earnings on total investment	5.3%	5.6%	5.4%
Net income before interest	15.0	13.0	11.2
Interest on long-term debt	6.4	5.3	3.7
No. of times long-term interest earned	2.3	2.4	3.0
Net income	8.2	7.2	6.7
Total revenues	90.5	82.5	75.7
Net income ratio	.09	.09	.09
Total utility operating expenses	75.5	69.5	64.5
Total utility operating revenues	87.6	81.3	75.2
Operating ratio	.86	.85	.86
Utility plant (gross)	330.1	278.5	249.9
Utility operating revenues	87.6	81.3	75.2
Ratio of plant investment to revenues	3.8	3.4	3.3

Capitalization:	1971 Amount	1971 % of Total	1970 Amount	1970 % of Total
Long-term debt	$110.4	51.5%	$111.2	57.9%
Preferred stock	33.3	15.5	23.3	12.1
Common stock & surplus	70.6	33.0	57.6	30.0
Total	$214.3	100.0%	$192.1	100.0%

Moody's Bond Rating: Aa

Dun & Bradstreet Credit Rating: 5A1

CORN BELT POWER COOPERATIVE
FINANCIAL ANALYSIS
DOCKET NO. 50-331

	(dollars in millions) Calendar Year Ended December 31		
	1971	1970	1969
Long-term debt	$ 29.9	$ 25.2	$ 20.7
Utility plant (net)	31.6	26.5	22.6
Ratio - debt to fixed plant	.95	.95	.92
Utility plant (net)	31.6	26.5	22.6
Capitalization	34.2	29.5	24.5
Ratio of net plant to capitalization	.92	.90	.92
Members equity	4.3	4.3	4.3
Total assets	36.7	31.7	26.6
Proprietary ratio	.12	.14	.16
Earnings available to members equity	0	.5	.5
Members equity	4.3	4.3	3.8
Rate of earnings on members equity	0	11.6%	13.2%
Net margin before interest	.5	.9	.9
Liabilities and capital	36.7	31.7	26.6
Rate of earnings on total investment	1.4%	2.8%	3.4%
Net margin before interest	.5	.9	.9
Interest on long-term debt	.5	.4	.4
No. of times long-term interest earned	1.0	2.2	2.2
Net margin	0	.5	.5
Total revenues	6.8	6.3	5.9
Net margin ratio	0	.08	.08
Total utility operating expenses	6.3	5.4	5.0
Total utility operating revenues	6.6	6.2	5.8
Operating ratio	.95	.87	.86
Utility plant (gross)	47.6	41.8	36.9
Utility operating revenues	6.6	6.2	5.8
Ratio of plant investment to revenues	7.2	6.7	6.4

	1971		1970	
Capitalization:	Amount	% of Total	Amount	% of Total
Long-term debt	$29.9	87.4%	$25.2	85.4%
Membership capital	4.3	12.6	4.3	14.6
Total	$34.2	100.0%	$29.5	100.0%

Dun and Bradstreet Credit Rating: 3A2

CENTRAL IOWA POWER COOPERATIVE
FINANCIAL ANALYSIS
DOCKET NO. 50-331

	(dollars in millions) Calendar Year Ended December 31		
	1971	1970	1969
Long-term debt	$ 38.2	$ 26.7	$ 22.2
Utility plant (net)	37.1	28.0	23.9
Ratio - debt to fixed plant	1.03	.95	.93
Utility plant (net)	37.1	28.0	23.9
Capitalization	43.1	30.9	26.0
Ratio of net plant to capitalization	.86	.91	.92
Members equity	4.9	4.2	3.8
Total assets	44.7	32.5	27.5
Proprietary ratio	.11	.13	.14
Earnings available to members equity	.7	.4	.6
Members equity	4.9	4.2	3.8
Rate of earnings on members equity	14.3%	9.5%	15.8%
Net margin before interest	1.1	.8	1.0
Liabilities and capital	44.7	32.5	27.5
Rate of earnings on total investment	2.5%	2.5%	3.6%
Net margin before interest	1.1	.8	1.0
Interest on long-term debt	.4	.4	.4
No. of times long-term interest earned	2.7	2.0	2.5
Net margin	.7	.4	.6
Total revenues	7.8	6.5	5.9
Net margin ratio	.09	.06	.10
Total utility operating expenses	6.7	5.7	4.9
Total utility operating revenues	7.7	6.4	5.9
Operating ratio	.87	.89	.83
Utility plant (gross)	51.8	42.0	37.0
Utility operating revenues	7.7	6.4	5.9
Ratio of plant investment to revenues	6.7	6.6	6.3

	1971		1970	
Capitalization:	Amount	% of Total	Amount	% of Total
Long-term debt	$38.2	88.6%	$26.7	86.4%
Members equities	4.9	11.4	4.2	13.6
Total	$43.1	100.0%	$30.9	100.0%

Dun and Bradstreet Credit Rating: 3A1

U.S. DEPARTMENT OF COMMERCE
National Oceanic and Atmospheric Administration
ENVIRONMENTAL RESEARCH LABORATORIES

APPENDIX E

Comments on

Duane Arnold Energy Center
Iowa Electric Light and Power Company
Final Safety Analysis Report
Volumes I through VIII dated 5/8/72
and Amendment 4 dated 9/13/72

Prepared by

Air Resources Environmental Laboratory
National Oceanic & Atmospheric Administration
January 8, 1973

The basis for our evaluation of the diffusion characteristics of the site is the separate report entitled "On Site Meteorological Data." These data cover a one-year period with winds and temperatures taken at the 10 and 50-m levels. Depending on whether an effective ground release or an elevated release was assumed, the winds at 10 m and 50 m were used respectively.

For the short-term (0-2 hours) ground release we have estimated from the joint frequency of wind speed, direction and temperature gradient in the vertical that a relative concentration of 8×10^{-4} sec m^{-3} will be exceeded 5 percent of the time at the minimum exclusion distance of 540 m (fig. 1.5-1). This is equivalent to Pasquill Type F diffusion, a wind speed of 0.75 m/sec and an additional dilution factor of 3 because of building wake effect.

For the short-term elevated release we have assumed a constant effective stack height of 100 m (the actual height of the stack). We have not substracted the height of the terrain (assuming the base of the stack is at zero) from the assumed stack height. This non-conservative assumption is more than balanced by measuring the winds at 50 m as opposed to a release at 100 m. From these meteorological data we estimate a relative concentration of 4×10^{-6} sec m^{-3} will be exceeded 5 percent of the time occurring at a distance of 2500 m. This compares to a value of 2.2×10^{-6} sec m^{-3} as shown by the applicant in figure 2 in the "On Site Meteorological Data" report.

For the maximum annual average concentration as a function of direction and distance from the source, we have estimated that this will occur towards the north of the site at a distance of 2500 m with a value of 8×10^{-8} sec m^{-3}. This assumes that the routine emission will have an effective stack height of 100 m and will be released throughout the entire year. This compares with the applicant's value of 6×10^{-8} sec m^{-3} as shown in figure 11.

Miller

DOCKET-50331-- 86

March 2, 1973

RECEIVED BY TIC MAR 8 1973

SUPPLEMENT NUMBER 1

TO THE

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

IOWA ELECTRIC LIGHT AND POWER COMPANY

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

Introduction

The Atomic Energy Commission's Safety Evaluation Report (SER) on the Duane Arnold Energy Center dated January 23, 1973, identified certain matters as requiring additional information from the applicant or that were still under review by the Regulatory staff.

The purpose of this Supplement is to update the SER based on the Regulatory staff's review of information contained in Amendment 12 to the FSAR and on a discussion held with the applicant since issuance of the SER.

Each of the sequentially-numbered items in this Supplement contains a specific reference to the sub-section of the SER that is being updated, either by replacement with or addition of the material provided in this Supplement.

Appendix A of this Supplement contains an updated chronology of our review and Appendix B is a listing of errata to the SER.

Item 1 Replace Section 2.3.6 with:

2.3.6 Conclusion

The opinion of the staff is that the onsite meteorological data presented in the FSAR, and subsequently verified by the applicant, indicates that the atmospheric dispersion conditions at the plant site are much less favorable than would normally be expected for this part of the country. Since both the applicant and the staff used these less favorable dispersion conditions as presented in the FSAR in calculating relative concentrations for the site, the staff concludes that the relative concentrations used for evaluation of the site are conservative and acceptable.

Item 2 In Section 5.2.2 insert the following on page 5-4 before the first full paragraph.

The capacity of the six safety/relief valves is sufficient to prevent actuation of the spring loaded safety valves, following any anticipated operational transient with an anticipatory scram initiated by the steam line or turbine valve position switches. In addition, the combined capacity of the six safety/relief and the two safety valves is sufficient to maintain the reactor pressure below 1350 psig (a 25 psi margin below the ASME code allowable pressure of 1375 psig), following any anticipated operational transient assuming that a scram is initiated by high reactor pressure and assuming that any one safety/relief or safety valve fails to open.

Item 3 Add the following at the end of Section 6.2.6

On February 7, 1973, a meeting was held with the applicant to discuss the status of the main steam line isolation valve seal system for the Duane Arnold plant. The applicant described three alternative seal systems which were studied: a water seal system, a pressurizing nitrogen system, and a controlled leakage system. The applicant proposes to adopt the controlled leakage system for the Duane Arnold plant. The detailed design of the controlled leakage system will be submitted in Amendment 13 on about March 15, 1973.

The controlled leakage system proposed by the applicant for the Duane Arnold plant used the one-inch diameter drain pipes located on each of the four steam lines just inboard of the outer isolation valve to collect and transport any leakage from the containment through the isolation valves, to the reactor building where the leakage will be filtered by the standby gas treatment system before being released to the atmosphere via the off-gas stack. Valve actuations necessary for system operation will be remote manually initiated, and will have interlocks to prevent initiation unless the pressure in the steam line at a point between the inner and outer isolation valves is below 50 psig. Design of the system will be in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 2 requirements and seismic Category I requirements. Each of the four main steam lines will have an independent controlled leakage system and each system will be testable.

The controlled leakage system proposed would not preclude the later adoption and use of a water seal or nitrogen seal system in the event one of these alternative systems is developed and found acceptable by the Regulatory staff.

The applicant indicated that the proposed controlled leakage system could be installed prior to the first refueling outage.

Although the staff has not completed its detailed review of the proposed controlled leakage system we conclude that the proposed system would reduce the direct leakage through the main steam isolation valves. We find the approach acceptable and will review the design prior to installation at the first refueling outage.

Item 4 <u>Substitute the following for a portion of Section 9.1.2</u>

9.1.2 Spent Fuel Storage

On page 9-3, last paragraph, delete the last 8 lines starting with, "For the postulated event of...," and replace with: "The applicant has analyzed the postulated event of a cask drop and determined that the cask could penetrate the floor of the cask pool. The applicant has proposed, in Amendment 12, to install an energy absorbing material to mitigate the consequence of a cask drop on the cask pool floor. The design of the energy absorbing material will be submitted by the applicant and reviewed by the Regulatory staff, prior to its installation, which will be no later than the first refueling operation.

APRIL 9, 1973

SUPPLEMENT NUMBER 2

TO THE

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

IOWA ELECTRIC LIGHT AND POWER COMPANY

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

Introduction

The Atomic Energy Commission's Safety Evaluation Report (SER) on the Duane Arnold Energy Center (DAEC), dated January 23, 1973, identified certain matters as requiring additional information from the applicant or that were still under review by the Regulatory staff. Supplement Number 1 to the SER, dated March 2, 1973, updated the SER by addressing eight of these matters. The ACRS completed its review of the DAEC at its March 8, 1973, meeting and reported its findings in a letter to Chairman Ray dated March 13, 1973.

The purpose of this Supplement is to address the ACRS comments in its letter of March 13, 1973, and to further update the SER, based on the Regulatory staff's review of information contained in Amendment 13 to the FSAR and on further discussions held with the applicant since issuance of Supplement 1 to the SER. Part A addresses the ACRS comments and Part B updates the SER.

Each of the sequentially-numbered items in Part B of this supplement contains a specific reference to the sub-section of the SER that is being updated, either by the replacement with, or the addition of, the material provided in this supplement.

Appendix A of this supplement contains an updated chronology of our review and Appendix B is a copy of the ACRS letter on the DAEC.

PART A: ACRS COMMENTS

Section 17.1 of the SER provides a discussion of the ACRS letter dated December 18, 1969, which reports on the DAEC construction permit review by the ACRS. This part of Supplement 2 to the SER is intended to replace Section 17.2 of the SER and addresses the ACRS letter dated March 13, 1973. In its letter of March 13, 1973, (Appendix B of this Supplement), the ACRS provided comments on the eight items discussed below.

Item 1: Leakage Control System for the MSL Isolation Valves

The ACRS noted that the criteria for functional adequacy of the leak-off system, and the detailed design in conformance with the criteria are not yet fully established, and requested that the Regulatory staff assure itself that the system finally installed does satisfy all of the considerations appropriate to the enhancement of containment reliability. The Regulatory staff stated on page 4 of Supplement 1 to the SER that "Although the staff has not completed its detailed review of the control leakage system, we conclude that the proposed system would reduce the direct leakage through the main steam isolation valves. We find the approach acceptable and will review the design prior to installation at the first refueling outage." The applicant provided in Amendment 13 to the FSAR some additional information regarding the leakage control system. The

Regulatory staff reaffirms its view expressed in Supplement 1 as cited above and will require the applicant to provide for its review prior to installation, detailed design information on the applicant's proposed leakage control system. In addition to the design description, the Regulatory staff will need for its review information on:

a. reliability of the rotometer at the flow rates considered in this application;

b. calibration of the rotometer when condensing steam is present in the leakage gas;

c. effect of moisture in the leakage gas on performance of the standby gas treatment system.

Item 2: Recirculation Pump Trip (ATWS)

The ACRS noted that the applicant will employ a recirculation pump trip system for the DAEC prior to initial fuel loading and recommended that the specific means for implementing the pump trip be resolved in a manner satisfactory to the Regulatory staff. This matter is currently under review by the Regulatory staff, as we indicated in Section 7.5 of the SER.

Item 3: Rod Sequence Control System

The ACRS noted that the applicant has committed to installation of a rod sequence control system and recommended that approved measures, satisfactory to the Regulatory staff, be placed in effect prior to operation above 1% of rated power. This matter which is generic to all BWR plants as discussed in paragraph 4.2.3 of the SER, will be resolved prior to operation of the DAEC above 1% of rated power.

Item 4: Postulated Drop of Spent Fuel Shipping Cask

The ACRS noted that a postulated cask drop is calculated to result in penetration or cracking of the cask pool bottom if unprotected, and that the applicant intends to install an energy absorbing material no later than the first refueling operation, and recommended that the matter be resolved in a manner satisfactory to the Regulatory staff. The applicant will be required to submit design information on the energy absorbing material along with those measures needed for its surveillance, for Regulatory staff review prior to installation, which will be no later than the first refueling operation requiring movement of a shipping cask.

Item 5: Potential for Missiles from Recirculation Pump and Motor

The ACRS noted that the applicant is reviewing means of dealing with the possibility of the recirculation pump impellor acting as a turbine causing the pump and motor to overspeed and become potential sources of missiles. The ACRS recommended that the matter be resolved in a manner satisfactory to the Regulatory staff. This matter is currently under review by the Regulatory staff.

Item 6: Linear Fuel Heat Ratings

The ACRS noted that potential effects of some aspects of fuel performance and LOCA-related phenomena on acceptable linear fuel heat ratings for the DAEC are under study and recommended that the matter be resolved in a manner satisfactory to the Regulatory staff. The Regulatory staff is currently reviewing this matter and plans to advise the ACRS on any developments. (See Part B, Item 1 of this Supplement.)

Item 7: Protection Against Pipe Whip

The ACRS noted that provisions are made in the DAEC for protection against pipe whip in accordance with criteria proposed by the Regulatory staff and recommended that particular emphasis be devoted to the performance of the protective systems with special attention during pre-operational testing and hot startup to assure that the protective measures meet the design criteria. The Regulatory staff plans to audit the applicant's activities in this regard during its startup testing of the DAEC.

Item 8: Other Problems Relating to Large Water Reactors

The ACRS recommended that other problems relating to large water reactors cited in previous ACRS reports be dealt with appropriately by the Regulatory staff and the applicant as suitable approaches are developed. The Regulatory staff intends to follow up on these other problems and intends to deal with them appropriately as recommended by the ACRS.

PART B: FURTHER UPDATING OF THE SER

In addition to the matters cited in Part A of this Supplement, the Regulatory staff has continued its evaluation, as discussed below, in the areas of fuel densification as discussed in Section 4.2.1 of the SER, postulated rupture in high energy lines outside containment as discussed in Item 7 of Supplement Number 1 to the SER, main steamline isolation valve leakage as discussed in Item 3 of Supplement Number 1 to the SER, and hydrogen-getter in the fuel as discussed in Section 4.2.1 of the SER.

Item 1: Fuel Densification

As anticipated in Section 4.2.1 of the SER, the matter of fuel densification is under review and evaluation by the Regulatory staff for all nuclear plants. Our current objective is to complete this review for the DAEC during the Summer of 1973. The areas of review include gap conductance and the effects of densification on gap conductance, clad creepdown, clad collapse, and the power spike due to axial gaps. We plan to address these matters is a further Supplement to the SER on completion of this review.

Item 2: Postulated Rupture in High Energy Lines Outside Containment

As indicated in Item 7 of Supplement 1 to the SER, the matter of postulated high energy pipeline breaks occurring external to the primary containment building is currently under review by the Regulatory staff. The preliminary conclusion given in Supplement 1 on this matter remains valid and we plan to report our final conclusion on completion of our review of this matter for the DAEC, which is now scheduled for the Summer of 1973.

Item 3: Main Steamline Isolation Valve Leakage (Amendment 13)

The applicant's Amendment 13 to the FSAR, filed on March 20, 1973, addresses our concern regarding main steamline isolation valve (MSLIV) leakage following a postulated design basis LOCA. The applicant discusses its evaluation of three design alternatives, including a water seal system, a gaseous nitrogen seal system, and a leakage control system. The information provided by the applicant on this matter confirms the preliminary conclusion of the Regulatory staff as reported in Item 3 of Supplement 1 to the SER that the design concept proposed by the applicant for the leakage control system is acceptable. The Regulatory staff will review the detailed design when it is completed and prior to installation at the first refueling outage to assure that all appropriate design criteria are satisfied. Additional consideration of this matter by the ACRS is given in Part A, Item 1, of this Supplement.

Item 4: Hydrogen-Getter

As indicated in Section 4.2.1 of the SER, the DAEC fuel will include a hydrogen-getter material. Substantive description of this material remains outstanding. We plan to address this matter in a further Supplement to the SER, when the applicant provides the necessary information and on completion of our review of the matter.

Appendix A

Chronology after February 28, 1973

March 2, 1973	Issuance of Supplement 1 to the Safety Evaluation for the Duane Arnold Energy Center.
March 8, 1973	ACRS meeting on the Duane Arnold Energy Center application.
March 13, 1973	Issuance of the ACRS letter on the Duane Arnold Energy Center.
March 20, 1973	Received Amendment 13 to the FSAR containing additional information on the applicant's proposed leakage control system.
March 27, 1973	Prehearing conference to consider environmental matters relating to the Duane Arnold Energy Center application.

APPENDIX B

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

March 13, 1973

Honorable Dixy Lee Ray
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON DUANE ARNOLD ENERGY CENTER

Dear Dr. Ray:

At its 155th meeting, March 8-10, 1973, the Advisory Committee on Reactor Safeguards completed its review of the application by the Iowa Electric Light and Power Company for authorization to operate the Duane Arnold Energy Center at power levels up to 1658 MWt. This project was considered at a Subcommittee meeting at the site on December 20, 1972, and at a Subcommittee meeting in Washington, D. C. on January 27, 1973. During its review the Committee had the benefit of discussions with representatives and consultants of Iowa Electric Light and Power Company, General Electric Company, Bechtel Corporation, Chicago Bridge and Iron Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed. The Committee reported to the Commission on the construction of this plant in its letter of December 18, 1969 and in its supplementary letter of February 11, 1970.

The Duane Arnold Energy Center Nuclear Plant will be located on a site of approximately 500 acres adjacent to the west bank of the Cedar River in a rural area approximately eight miles northwest of the city of Cedar Rapids, Iowa.

The applicant proposes to install, no later than the first scheduled refueling outage, a leak-off system intended to reduce the potential consequences of excessive leakage from the main steam isolation valves. The criteria for functional adequacy of the leak-off system and the detailed design in conformance with the criteria are not yet fully established. The Regulatory Staff should assure itself that the system finally installed does satisfy all of the considerations appropriate to the enhancement of containment reliability.

The applicant will employ a recirculation pump trip as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The trip will be installed prior to initial fuel loading. The Committee believes that this represents a substantial improvement. The specific means for implementing the pump trip should be resolved in a manner satisfactory to the Regulatory Staff.

The applicant is committed to the installation of a rod sequence control system which will render the probability of occurrence of a postulated, high-worth control rod drop accident negligibly low. This matter is under review and should be resolved in a manner satisfactory to the Regulatory Staff. Approved measures should be placed in effect prior to operation above 1% of rated power.

The shipping cask pool is physically separated from the spent fuel pool by a wall to a height above the top of stored fuel elements and a removable gate above that level. A postulated cask drop is calculated to result in penetration or cracking of the cask pool bottom if unprotected. To avoid such damage, the applicant intends to install an energy absorbing material covering the bottom of the cask pool, no later than the first refueling operation. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

In the unlikely event that a break occurs in the recirculation pump discharge line, the pump impeller might act as a turbine causing the pump and motor to overspeed and become potential sources of missiles. The applicant is reviewing means of dealing with this possibility. The Committee believes that this matter should be resolved in a manner satisfactory to the Regulatory Staff.

The potential effects of some aspects of fuel performance and LOCA-related phenomena on acceptable linear fuel heat ratings for the Duane Arnold Energy Center are under study. This matter should be resolved in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The applicant has provided protection against pipe whip in accordance with the criteria proposed by the Regulatory Staff in the Regulatory Guide, "Protection Against Pipe Whip Inside Containment", now under preparation. The Committee has emphasized the desirability of such protective measures in several letters. The Committee also recognizes that systems for restraining against pipe whip could generate undesirable stress concentrations unless properly designed and suitably installed. Therefore, particular emphasis should be devoted to the following: (1) a better understanding of transient response in piping than is usually required; (2) quality assurance pertaining to design and installation of pipe restraints, including verification that the design computational techniques account for operational conditions and postulated transients; (3) careful examination during preoperational testing and hot startup to validate that the installation meets the design criteria.

Honorable Dixy Lee Ray March 13, 1973

Other problems relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports should be dealt with appropriately by the Regulatory Staff and the applicant as suitable approaches are developed.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that the Duane Arnold Energy Center can be operated at power levels up to 1658 MWt without undue risk to the health and safety of the public.

Sincerely yours,

H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

References Attached:

Honorable Dixy Lee Ray March 13, 1973

References

1) Final Safety Analysis Report, Duane Arnold Energy Center

2) Amendments 1-12, Final Safety Analysis Report, Duane Arnold Energy Center

3) Supplement to Amendment No. 1, dated June 6, 1972

4) Iowa Electric Light and Power Company letter dated July 10, 1972 re: Relief Valve Discharge Line

5) Iowa Electric Light and Power Company letter dated October 24, 1972 re: fuel design (proprietary)

6) Iowa Electric Light and Power Company letter dated December 18, 1972 re: installation of a main steam line isolation valve seal system in the Duane Arnold Energy Center

7) Iowa Electric Light and Power Company letter dated January 15, 1973 adopts the GE NEDM-10735 "Densification Considerations in BWR Fuel Design and Performance"

8) Iowa Electric Light and Power Company letter dated January 16, 1973, re: the gaseous effluent discharges from the Duane Arnold Energy Center being "as low as practicable" and consistent with the proposed Appendix I to 10 CFR Part 50

9) Iowa Electric Light and Power Company letter dated January 22, 1973 transmitting revised operating pressure and temperature limits for Duane Arnold Energy Center

10) Directorate of Licensing Safety Evaluation Report dated January 23, 1973

11) Directorate of Licensing Supplement No. 1 to the Safety Evaluation dated March 2, 1973

February 20, 1974

SUPPLEMENT NUMBER 3

TO THE

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

IOWA ELECTRIC LIGHT AND POWER COMPANY

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

INTRODUCTION

The Atomic Energy Commission's Safety Evaluation Report (SER) on the Duane Arnold Energy Center (DAEC), dated January 23, 1973, identified certain matters as requiring additional information from the applicant or that were still under review by the Regulatory staff. Supplement Number 1 to the SER, dated March 2, 1973, updated the SER by addressing eight of these matters. The ACRS completed its review of the DAEC at its March 8, 1973, meeting and reported its findings in a letter to Chairman Ray, dated March 13, 1973. Supplement Number 2 to the SER, dated April 9, 1973, addressed the ACRS comments and further updated the SER.

The purpose of this Supplement is to again update the SER, based on the Regulatory staff's review of the additional information provided by the applicants. Each of the sequentially numbered items of this Supplement contains a specific reference to the subsection of the SER that is being updated, either by the replacement with, or by the addition of, the material provided in this Supplement.

With the indicated resolution of these outstanding matters, the Regulatory staff has completed its review of those items for which resolution is required prior to issuance of an operating license and concludes that there is reasonable assurance that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter 1.

ITEM 1 REPLACEMENT FOR THE FOURTH PARAGRAPH OF SECTION 4.2.3 (PAGE 4-7)

The Regulatory staff requires that peak fuel enthalpies not exceed 280 (calories per gram) in the event of any postulated control rod drop accident. As described in General Electric Company's (GE) Topical Report NEDO-10527 and its Supplements, if the control rod worth does not exceed 1.43% Δk/k at low power levels (20% of rated power or less), the peak fuel enthalpy in the event of a rod drop accident will not exceed 280 cal/gm. Limiting the maximum control rod worth while at power levels below 30% of rated power to less than 1.43% Δk/k, will be accomplished by: 1) electrically restricting the removal of the first 50% of the rods to be withdrawn in a prescribed configuration, and the remaining 50% of the rods to single notch movement, as restricted by a Rod Sequence Control System (RSCS) which employs a notch group mode of operation as described in Amendment 14 to the PSAR; and 2) the Rod Worth Minimizer (RWM) which controls the specific order of control rod withdrawal. In the event of RWM inoperability, the applicant will be required to assign a second operator to monitor control rod movement to assure that the first operator follows the pre-selected order.

We conclude that the applicants' proposed system of rod movement control and the specified rod removal order adequately assure, for the first fuel cycle, that a control rod worth greater than 1.43% will not occur at power levels below 20% of rated power. Calculated results reported by the applicants in Amendment 14 indicate that the maximum worth rod, when employing the notch group mode of RSCS operation, would be significantly less than 1.0%. Nevertheless, we require that the RSCS system, as proposed by the applicants, be improved by adding an electrical interlocking circuit which assures that the rods in a particular notch group are positioned within one notch of each other. We will require the applicants to make this design change and to submit the proposed RSCS design modification for our review, prior to its installation during the first refueling outage.

ITEM 2 <u>REPLACEMENT FOR SECTION 7.6 (PAGE 7-7)</u>

7.6 <u>Control Over Maximum Rod Reactivity Worth</u>

In response to the current Regulatory staff concern for the control over selection and movement of control rods during reactor startup (see Item 3 below on Control Rod Drop Accident), the applicants have installed additional controls as described in Amendment 14 to the FSAR, which meet the requirements of the Regulatory staff for the first operating cycle (see Item 1 above). However, we require that further design improvements be developed for installation during the first refueling outage; these further design improvements will electrically restrict rod positions within a notch group. The applicants will be required to submit the details of this design change for review by the Regulatory staff prior to its installation.

FEB 2 2 1974

ML021960274

Docket No. 50-331

Iowa Electric Light and Power Company
ATTN: Duane Arnold, President
Security Building
P. O. Box 351
Cedar Rapids, Iowa 52406

Gentlemen:

The Atomic Energy Commission has issued Facility Operating License No. DPR-49. The licensees of DPR-49 are Iowa Electric Light and Power Company, Central Iowa Power Cooperative and Corn Belt Power Cooperative. DPR-49 authorizes operation of the Duane Arnold Energy Center in accordance with the Technical Specifications, Appendices A & B, attached thereto. The steady state reactor core power levels authorized by DPR-49 shall not exceed 1658 megawatts thermal. A copy of the license and technical specifications are enclosed.

Note that the Technical Specifications specify that the licensee shall not undertake initial criticality until specifically approved in writing by the Commission. Representatives of the Division of Regulatory Operation will be at the site during fuel loading and will verify that the assessment of the preoperational test data, the Surveillance Test procedures and the review of non-conformance reporting has been completed. We will inform you promptly of the results of their review.

A related notice, which is being forwarded to the Office of the Federal Register for filing and publication, is enclosed for your information.

Four signed originals of Amendment No. 1 to Indemnity Agreement No. B-68, which covers the activities authorized under License No. DPR-49, are enclosed for review and acceptance by the licensees. One copy of this agreement should be retained by each licensee and one copy signed by all licensees should be returned to this office.

Sincerely,

/S/ D. VASSALLO

for R. C. DeYoung, Assistant Director
for Light Water Reactors Group 1
Directorate of Licensing

Enclosures:
See Page Two

Appd LB

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Enclosures:
1. License No. DPR-49 w/Tech. Specs. A & B
2. Federal Register Notice
3. Indemnity Agreement - Amendment No. 1 to B-68

cc: Jack R. Newman, Esq.
Harold F. Reis, Esq.
Newman, Reis & Axelrad
1025 Connecticut Avenue, N.W.
Washington, D. C. 20036

Director
Office for Planning and Programming
523 East 12th Street
Des Moines, Iowa 50319

Mr. Dudley Henderson
Chairman, Linn County
Board of Supervisors
Cedar Rapids, Iowa 52406

Mr. Ed Vest
Environmental Protection Agency
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Mr. T. B. Abernathy
U. S. Atomic Energy Commission
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Oak Ridge, Tennessee 37830

bcc: A. Rosenthal, ASLAB
N. H. Goodrich, ASLBP

Distribution:
AEC PDR
Local PDR
Docket File
LWR 1-2 File
RP Reading(w/o Tech.Specs.)
R. Newton, OGC
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RO (3)
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M. Jinks (w/2 encls.)
R. C. DeYoung
R. Vollmer
C.Hebron,F&M(w/o Tech.Specs
D.Foster,F&M(w/o Tech.Specs
Ellen Brown, F&M
A.Braitman,OAI(w/o Tech. Specs.)
G. Owsley
M. Maigret
S.Kari (w/o Tech. Specs.)
W.Miller,DR:AO(w/o Tech Specs)
F. St. Mary, EP-4
S. Sheppard, EP-4 (w/O TechSpecs)
D. Muller, AD/EP
K. Goller, LWR 1-3
D. Vassallo, LWR 1-1
ACRS (16)

OFFICE ▶	L:LWR 1-2	L:LWR 1-2	OGC	OAI	L:AD/EP	L:AD/LWR 1
SURNAME ▶	Maigret:eg GOwsley	WButler		ABraitman	DMuller	RCDeYoung
DATE ▶	2/22/74	2/22/74	2/22/74	2/22/74	2/22/74	2/22/74

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

IOWA ELECTRIC LIGHT AND POWER COMPANY
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE
DOCKET 50-331
DUANE ARNOLD ENERGY CENTER
FACILITY OPERATING LICENSE

License No. DPR-49

1. The Atomic Energy Commission (the Commission) having found that:

   A. The application for license filed by Iowa Electric Light and Power Company, Central Iowa Power Cooperative and Corn Belt Power Cooperative (the licensees) 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 and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Duane Arnold Energy Center (facility) has been substantially completed in conformity with Construction Permit No. DPPR-70; the application, as amended; the provisions of the Act; and the rules and regulations of the Commission;

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

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

   E. Iowa Electric Light & Power Company is technically qualified and the licensees are financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;

   F. The licensees have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements", of the Commission's regulations;

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

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of Facility Operating License No. DPR-49 is in accordance with 10 CFR Part 50, Appendix D, of the Commission's regulations and all applicable requirements of said Appendix D have been satisfied;

I. The receipt, possession, and use of source, by-product and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Part 30 and 70, including 10 CFR Section 30.33, 70.23 and 70.31.

2. Facility Operating License No. DPR-49 is hereby issued to the Iowa Electric Light and Power Company (IEL&P), Central Iowa Power Cooperative (CIPCO) and Corn Belt Power Cooperative (Corn Belt) to read as follows:

   A. This license applies to the Duane Arnold Energy Center, a boiling water reactor and associated equipment (the facility), owned by the licensees and operated by IEL&P. The facility is located on the licensees' site near Palo in Linn County, Iowa. This site consists of approximately 500 acres adjacent to the Cedar River and is described in the "Final Safety Analysis Report" as supplemented and amended (Amendments 1 through 14) and the Environmental Report as supplemented and amended (Supplements 1 through 5).

   B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

      (1) Iowa Electric Light & Power Company, pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", to possess, use, and operate the facility; and CIPCO and Corn Belt to possess the facility at the designated location in Linn County, Iowa, in accordance with the procedures and limitations set forth in this license;

(2) IEL&P, pursuant to the Act and 10 CFR Part 70, "Special Nuclear Material", to receive, possess and use at any time up to 3500 kilograms of U-235 in reactor fuel assemblies enriched in the U-235 isotope in connection with operation of the facility;

(3) IEL&P, pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Licensing of Byproduct Material", to receive, possess, and use in connection with operation of the facility:

(a) Any byproduct material with Atomic Numbers 3 to 83, inclusive, without restrictions as to chemical and physical form, not to exceed 1 millicurie each, total not to exceed 50 millicuries;

(b) Cobalt 60, in sealed sources not to exceed 15 millicuries;

(c) Strontium 90, in sealed sources not to exceed 5 millicuries;

(d) Cesium 137, in sealed sources not to exceed a total of 210 curies;

(e) Antimony 124, in sealed sources not to exceed four sources each of 1200 curies;

(f) Americium 241, in sealed sources not to exceed 6 curies; and

(4) IEL&P, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such by-product and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

IEL&P is authorized to operate the Duane Arnold Energy Center at steady state reactor core power levels not in excess of 1658 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendices A & B attached hereto are hereby incorporated in this license. IEL&P shall operate the facility in accordance with the Technical Specifications.

D. This license is effective as of the date of issuance and shall expire at midnight on June 21, 2010.

FOR THE ATOMIC ENERGY COMMISSION

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachment:
Appendices A & B - Technical Specifications

Date of Issuance: FEB 22 1974

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-331

IOWA ELECTRIC LIGHT AND POWER COMPANY

CENTRAL IOWA POWER COOPERATIVE

CORN BELT POWER COOPERATIVE

(DUANE ARNOLD ENERGY CENTER)

NOTICE OF ISSUANCE OF FACILITY OPERATING LICENSE

Notice is hereby given that the Atomic Energy Commission has issued Facility Operating License No. DPR-49 to Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative authorizing operation of the Duane Arnold Energy Center in accordance with the provisions of the license and the Technical Specifications. The steady state reactor core power levels authorized by the license shall not exceed 1658 megawatts thermal. The Duane Arnold Energy Center is a boiling water nuclear reactor located at the licensees' site near Palo in Linn County, Iowa.

The Commission has made appropriate findings as required by the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license. The application for the license complies with the standards and requirements of the Act and the Commission's rules and regulations.

The license is effective as of its date of issuance and shall expire on June 21, 2010.

A copy of (1) Facility Operating License No. DPR-49, complete with Technical Specifications (Appendices "A" and "B"); (2) the report of the Advisory Committee on Reactor Safeguards, dated March 13, 1973; (3) the Directorate of Licensing's Safety Evaluation, dated January 1973; (4) Supplement No. 1 to the Safety Evaluation, dated March 2, 1973; (5) Supplement No. 2

to the Safety Evaluation, dated April 9, 1973; (6) Supplement No. 3 to the Safety Evaluation, dated February 20, 1974; (7) the Final Safety Analysis Report and amendments thereto; (8) the applicants' Environmental Report, dated April 1971, revised November 1971, and supplements thereto; (9) the Draft Environmental Statement, dated November 1972; and (10) the Final Environmental Statement, dated March 1973, are available for public inspection at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C. and at the Reference Service, Cedar Rapids Public Library, 426 Third Avenue, S.E., Cedar Rapids, Iowa 52401. A copy of the license and the Safety Evaluation and Supplements thereto may be obtained upon request addressed to the United States Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing.

Dated at Bethesda, Maryland, this 22nd day of February, 1974.

FOR THE ATOMIC ENERGY COMMISSION

Raymond R. Powell

Raymond R. Powell, Acting Chief
Light Water Reactors Projects Branch 1-2
Directorate of Licensing

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket Nos. 70-1384
50-331

AMENDMENT TO INDEMNITY AGREEMENT NO. B-68

AMENDMENT NO. 1

Effective February 22, 1974, Indemnity Agreement No. B-68 between Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative and the Atomic Energy Commission, dated May 15, 1973, is hereby amended as follows:

Item 2a of the Attachment to the indemnity agreement is deleted in its entirety and the following substituted therefor:

Item 2 - Amount of financial protection

a.	$ 1,000,000	(From 12:01 a.m., May 15, 1973, to 12:00 midnight, February 21, 1974 inclusive)
	$95,000,000	(From 12:01 a.m., February 22, 1974)

Item 3 of the Attachment to the indemnity agreement is deleted in its entirety and the following substituted therefor:

Item 3 - License number or numbers

SNM-1349	(From 12:01 a.m., May 15, 1973, to 12:00 midnight, February 21, 1974 inclusive)
DPR-49	(From 12:01 a.m., February 22, 1974)

Item 5 of the Attachment to the indemnity agreement is amended by adding the following:

Nuclear Energy Liability Policy (Facility Form) No. MF-72
issued by Mutual Atomic Energy Liability Underwriters.

FOR THE UNITED STATES ATOMIC ENERGY COMMISSION

Jerome Saltzman

Jerome Saltzman, Deputy Chief
Office of Antitrust & Indemnity
Directorate of Licensing

Accepted ____________________, 1974

By ______________________________
IOWA ELECTRIC LIGHT AND POWER COMPANY

Accepted ____________________, 1974

By ______________________________
CENTRAL IOWA POWER COOPERATIVE

Accepted ____________________, 1974

By ______________________________
CORN BELT POWER COOPERATIVE

Enclosure 3

CH KLIST FOR ISSUANCE OF FACILITY CENSE

APPLICANT Iowa Electric Light & Power Company DOCKET NO. 50-331

FACILITY Duane Arnold Energy Center

PROJECT MANAGER Gerry Owsley

LICENSING ASSISTANT Madelyn J. Maigret

	DATE
Notice of Consideration of Issuance of License:	
Published in FEDERAL REGISTER	September 29, 1972
Action Date	~~October 30, 1972~~
OR	
Initial Decision	June 20, 1973
Safety Review:	
L Safety Evaluation	January 23, 1973
ACRS Letter	March 13, 1973
Environmental Review:	
Final Environmental Statement	March 1973
Published in FEDERAL REGISTER	March 1973
Antitrust Review:	
OAI Concurrences	February 22, 1974
Notifications Required by Act & Commission Rules*:	
State Official	May 26, 1972
Local Official	May 26, 1972
Water Quality Certification: (401)	
Submittal by Applicant	April 27, 1973
Transmitted to EPA	~~April 27, 1973~~
License Fee:	
Amount: $544,705 Paid	February 21, 1974
Indemnity Agreement:	
OAI Concurrence	~~February~~ 22, 1974
Status of Outstanding Construction Items Checked w/RO	February 20, 1974
Regulatory Operations Final Report: (If Available)	February 20, 1974
Technical Specifications:	
RP Concurrence	~~February 20, 1974~~
EP Concurrence	February 5, 1974
OR Concurrence	February 22, 1974
Public Announcement (to be released): Copy Attached	
Issuance Package: OGC Concurrence	February 22, 1974
1. License	February 22, 1974
2. FEDERAL REGISTER Notice	~~February 22, 1974~~
3. Letter to Applicant	February 22, 1974
4. Information Report	February 28, 1974

* Date Initial Application Forwarded

Revised: MAY 7 1973

PROPOSED PRESS RELEASE

AEC ISSUES OPERATING LICENSE FOR NUCLEAR POWER PLANT IN IOWA

A full power, full-term operating license for the Duane Arnold Energy Center near Cedar Rapids, Iowa, was issued on _____________, 1974, to the Iowa Electric Light and Power Company, Cornbelt Power Cooperative and Central Power Cooperative by the Atomic Energy Commission's ~~Regulatory Staff.~~ Directorate of Licensing.

At full power the plant, which uses a boiling water reactor, will have a net electrical output of about 569 megawatts.

The term of the license is 40 years from June 1970 when the AEC construction permit for the plant was issued. The station is located near the City of Palo in Linn County, adjacent to the Cedar River, about 8 miles northwest of Cedar Rapids.

The license was issued after findings by the AEC that the application for the operating license complied with AEC requirements and that the plant has been satisfactorily constructed and is ready for fuel loading.

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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

FEB 20 1974

A. Giambusso, Deputy Director
For Reactor Projects
Directorate of Licensing

Docket No. 50-331

IOWA ELECTRIC LIGHT AND POWER COMPANY (DUANE ARNOLD)

We have been informed by our Region III Office that the Duane Arnold facility has been substantially completed in accordance with the amended application with the exceptions listed in the enclosure. As indicated in our memorandum to Mr. R. S. Boyd on January 4, 1974, we have found that the licensee has implemented an acceptable Q/A program for operations.

The licensee plans to complete the listed exceptions within the time frame stated in the enclosure. Assuming satisfactory resolution of the items in the enclosure and verification of their completion by Regulatory Operations, we recommend that an Operating License be issued to the applicant. We also recommend that the letter transmitting the Operating License to the applicant state that fuel loading and initial criticality shall not be commenced until verification of completion of the items in the enclosure by Regulatory Operations.

Harold D. Thornburg

John G. Davis, Deputy Director
for Field Operations
Directorate of Regulatory Operations

Enclosure:
As Stated

Enclosure

The following updated findings are the result of recent regulatory inspections at the Duane Arnold Energy Center. These inspections were performed by the identified regulatory groups on the following dates:

Group	Dates
Testing and Startup	Feb. 7-8, 15-16-19-18-19, 1974
Construction	Feb. 17-18-19, 1974
Security	Jan. 29-30, 1974
Preparedness Plan	Jan. 29-30-31, Feb. 1, 18-19, 1974
Health Physics	Jan. 30, Feb. 1, 15-16, 18-19, 1974
Special Nuclear Materials	Jan. 24, 1974

Items requiring resolution:

a. All construction and preoperational testing required for initial fuel loading and sub-critical testing has been completed. Final evaluation of the preoperational test data remains to be completed, however, and is scheduled for completion on February 22, 1974. Completion of this data is required prior to initial fuel loading and sub-critical testing.

b. The identified Cold Functional Testing Program remains to be completed. The remainder is in progress and is scheduled for completion on February 22, 1974. Completion of this item is required prior to initial fuel loading and sub-critical testing.

c. The functional verification of the identified Surveillance Test Procedures remains to be completed and is scheduled for completion on February 22, 1974. Resolution of this item is required prior to initial fuel loading and sub-critical testing.

d. Review and resolution of nonconformance Reports (NCR), Deficiency Reports (DR) and Field Change Notices (FCN) on safety related systems required for initial fuel loading and sub-critical testing are in progress. Regulatory inspections verify that these items are identified and the licensee is committed to proper evaluation and resolution prior to the start of the initial fuel loading program.

MAR 1 3 1974

ML021860271

Docket No. 50-331

Iowa Electric Light and Power Company
ATTN: Duane Arnold, President
Security Building
P. O. Box 351
Cedar Rapids, Iowa 52406

Gentlemen:

The Atomic Energy Commission has issued Amendment No. 1 to Facility Operating License No. DPR-49 (copy enclosed), which authorizes the licensees to own, possess and use an increased amount of Antimony-124 not to exceed eight sources, each of 1200 curies in sealed sources. This amendment has been issued to correct an error in the number of sources previously authorized for the Duane Arnold Energy Center site. Iowa Electric Light & Power Company previously understood that the standard startup source contained a single 1200 curie Antimony-124 source pin per source holder. However, it has since been learned that four such source holders, each containing two (2) source pins, were delivered to the site and are necessary for startup of the Duane Arnold Energy Center. Therefore, Amendment No. 1 to Facility Operating License No. DPR-49 authorizing possession and use of eight (8) Antimony-124 source pins each not to exceed 1200 curies is necessary. We have determined that this amendment does not present a significant hazards consideration.

A copy of a related notice, which has been forwarded to the Office of the Federal Register for publication, is enclosed for your information.

Sincerely,

Original Signed by
R. C. DeYoung

Richard C. DeYoung, Assistant Director
for Light Water Reactors, Group 1
Directorate of Licensing

Enclosures:
1. Amendment No. 1 to DPR-49
2. Federal Register Notice

cc's: See Next Page

OFFICE ▶	L:LWR 1-2	L:LWR 1-2	L:LWR 1-2	OGC	L:AD/LWR 1	
SURNAME ▶	MMallary:eag	GOwsley	WRButler	W. MASSAR	RCDeYoung	
DATE ▶	3/13/74	3/13/74	3/13/74	3/13/74	3/13/74	

Form AEC-318 (Rev. 9-53) AECM 0240 GPO 643-16-81465-1 445-678

cc's: Jack R. Newman, Esq.
Harold F. Reis, Esq.
Newman, Reis & Axelrad
1025 Connecticut Avenue, N.W.
Washington, D. C. 20036

Director
Office for Planning and Programming
523 East 12th Street
Des Moines, Iowa 50319

Mr. Dudley Henderson
Chairman, Linn County
Board of Supervisors
Cedar Rapids, Iowa 52406

Mr. Ed Vest
Environmental Protection Agency
1735 Baltimore Avenue
Kansas City, Missouri 64108

Mr. J. R. Buchanan
Assistant Director
Nuclear Safety Information Center
Oak Ridge National Lab
P. O. Box Y
Oak Ridge, Tennessee 37830

Mr. T. B. Abernathy
U. S. Atomic Energy Commission
Division of Technical Information Ext.
Document Management Branch
P. O. Box 62
Oak Ridge, Tennessee 37830

bcc: A. Rosenthal, ASLAB
N. H. Goodrich, ASLBP

DISTRIBUTION:
AEC PDR
Local PDR
Docket File (50-331)
LWR 1-2 File
R. Newton, OGC
W. Massar, OGC
F. St. Mary, EP-4
S. Sheppard, EP-4
RO (3)
N. Dube (w/o Tech. Specs)
M. Jinks (w/4 encls.)
R. C. DeYoung
C. Hebron, F&M(OL only)
D. Foster, F&M(OL only)
Ellen Brown,F&M(OL only)
A. Braitman, OAI(w/o Tech Specs)
S. Kari(w/o Tech Specs)
W. Miller,DR:AO(w/o T.S.)
LWR 1 Branch Chiefs(w/o Tech Specs)
ACRS (16)
D. Muller
M. Maigret
G. Owsley

OFFICE ▶						
SURNAME ▶						
DATE ▶						

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

IOWA ELECTRIC LIGHT AND POWER COMPANY
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE
DOCKET NO. 50-331
DUANE ARNOLD ENERGY CENTER
FACILITY OPERATING LICENSE

License No. DPR-49
Amendment No. 1

1. The Atomic Energy Commission (the Commission) having found that:

   A. The application for amendment, dated March 13, 1974, complies with the requirements of the Atomic Energy Act of 1954, as amended, and the Commission's regulations set forth in 10 CFR Chapter 1;

   B. The facility will operate in conformity with the license, as amended, 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 regulation;

   D. Prior public notice of proposed issuance of this amendment is not required since the amendment does not present a significant hazards consideration.

2. Accordingly, Facility Operating License No. DPR-49 issued to Iowa Electric Light & Power Company, Central Iowa Power Cooperative and Corn Belt Power Cooperative is hereby amended by revising the following paragraph thereof in its entirety to read:

   2.B.(3) (e) Antimony-124, in sealed sources not to exceed eight sources each of 1200 curies

This amendment is effective as of the date of issuance.

FOR THE ATOMIC ENERGY COMMISSION

for Roger S. Boyd

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Date of Issuance: MAR 13 1974

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-331

IOWA ELECTRIC LIGHT AND POWER COMPANY

CENTRAL IOWA POWER COOPERATIVE

CORN BELT POWER COOPERATIVE

(DUANE ARNOLD ENERGY CENTER)

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the Atomic Energy Commission (the Commission) has issued Amendment No. 1 to the Facility Operating License No. DPR-49 to the Iowa Electric Light and Power Company, Central Iowa Power Cooperative and Corn Belt Power Cooperative (the licensees). This amendment authorizes the licensees to increase the amount of byproduct material they may receive, possess, and use in connection with operation of the Duane Arnold Energy Center located on the licensees' site near Palo in Linn County, Iowa. The amendment, effective as of the date of issuance, authorizes the receipt, possession and use of an additional four sources for a total of eight sources, each of 1200 curies of Antimony 124 in sealed sources.

The licensees stated, in a letter to the Commission, dated March 13, 1974, that the existence and need for the additional four source pins was discovered subsequent to delivery of the sources to the site. Four source holders, each containing two (2) 1200 curie Antimony-124 source pins, are at the site and are necessary for startup of the Duane Arnold Energy Center. Therefore, Amendment No. 1 to Facility Operating License No. DPR-49 authorizing possession and use of eight (8) Antimony-124 source pins, each not to exceed 1200 curies is necessary.

The Staff's Safety Evaluation Report, upon the basis of which the original license was issued, is based upon the Final Safety Analysis Report which, on Page 3.3-14 of the text and in Figure 7.5.1 describes the correct number of sources for the Duane Arnold Energy Center. Accordingly, the Regulatory staff has determined that this amendment does not present a significant hazards consideration.

The Director of Regulation has made appropriate findings as required by the Act and the Commission's regulations in 10 CFR Chapter I, which are set forth in the license amendment.

The amendment is effective as of the date of issuance. The licensees' application for amendment, dated March 13, 1974, and a copy of Amendment No. 1 to Facility Operating Licensing No. DPR-49 are available for public inspection at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C. 20545, and at the Reference Service, Cedar Rapids Public Library, 426 Third Avenue, S.E., Cedar Rapids, Iowa 52401. Single copies of the amendment may be obtained upon request addressed to the United States Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing.

Dated at Bethesda, Maryland, this 13th day of March, 1974.

FOR THE ATOMIC ENERGY COMMISSION

Walter R. Butler

Walter R. Butler, Chief
Light Water Reactors Project Branch 1-2
Directorate of Licensing

CHECKLIST FOR ISSUANCE OF AMENDMENT TO
CONSTRUCTION PERMIT OR FACILITY OPERATING LICENSE

APPLICANT Iowa Electric Light & Power Company DOCKET NO. 50-331

FACILITY Duane Arnold Energy Center

PROJECT MANAGER Gerald Owsley

LICENSING ASSISTANT Madelyn J. Maigret

	DATE
Notice of Proposed Issuance Published In FEDERAL REGISTER Action Date	March 20, 1974
OR	
Order Directing Action by whom: IEL&P Co. letter requesting Amendment	March 13, 1974
Issuance Package: OGC Concurrence	
1. License Amendment	3/13/74
2. FEDERAL REGISTER Notice	3/13/74
3. Staff Evaluation	3/13/74
4. Letter to applicant	3/13/74
NO CHANGE IN POWER LEVEL	

For Amendments Affecting Power Level:

RO Notification and/or Concurrence	
OAI Notification and/or Concurrence	
Bus. Mgmt-OA Notification and/or Concurrence	
OIS Notification	