DISTRIBUTION:
AEC PDR
Local PDR
Docket File
RP Reading
BWR-1 File
SHanauer
RSBoyd
DSkovholt
FSchroeder
RMaccary
DKnuth
RTedesco
HDenton
WButler
JStolz
RClark
RIreland
WHaass
OGC
Regulatory Operations (3)
GLear
SKari
MMaigret
JHulman
BShewmaker
CFarrell
TEssig
JHowze
ACunningham
BWaterfield
EMarkee

AUG 7 1972

Docket No. 50-331

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

Gentlemen:

In order that we may continue our review of your application for a license to operate the Duane Arnold Energy Center, additional information on those matters set forth in the enclosure is needed. Many of these matters requiring additional information were discussed with your representatives at a technical meeting on July 20, 1972.

In order to maintain our licensing review schedule we will need a completely adequate response by September 15, 1972. Please inform us within 7 days after receipt of this letter of your confirmation of the schedule or the date you will be able to meet. If you cannot meet our specified date or if your reply is not fully responsive to our requests it is highly likely that the overall schedule for completing the licensing review for this project will have to be extended. Since reassignment of the staff's efforts will require completion of the new assignment prior to returning to this project, the extent of extension will most likely be greater than the extent of delay in your response.

The questions in the enclosure have been grouped by sections that correspond to the relevant sections of the DAEC Final Safety Analysis Report. Some of these questions may have been addressed by applicants on other dockets. Your response to these questions may be made either by incorporating the information provided for other nuclear plants by reference, or you may amend your application by submitting revised pages and supplements.

Please contact us if you desire additional discussion or clarification of the material requested.

Sincerely,

for W. P. Haass

Roger S. Boyd, Assistant Director
for Boiling Water Reactors
Directorate of Licensing

Enclosure:
Request for Additional Information

cc: Mr. Charles Sandford
Vice President
Iowa Electric Light & Power Co.
General Office
Cedar Rapids, Iowa 52406

Mr. Jack Newman
Lowenstein, Newman, and Reis
1100 Connecticut Avenue, N.W.
Washington, D.C. 20036

CRESS	OFFICE ▶	L:BWR-1	L:BWR1	L:AD/BWR			
[-1010 F#1n	SURNAME ▶	GEL GELear:th	WRButler WB	WPH/pn RSBoyd			LA
7/28/72	DATE ▶	8/2/72	8/2/72	8/4/72			

Form AEC-318 (Rev. 9-53) AECM 0240

REQUEST FOR ADDITIONAL INFORMATION

IOWA ELECTRIC LIGHT AND POWER COMPANY

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

2.0 SITE AND ENVIRONS

2.12 Provide the reactor location to the nearest second of latitude and longitude and to the nearest 100 meters using Universal Transverse Mercator (UTM) coordinates.

2.13 Describe the type and weight of aircraft which use the two airfields located west and southeast of the site. Provide runway orientation, length, distance from plant, and plans for future use. Indicate the annual number of take-offs and landings and describe the location of the take-off, landing, and low level flight patterns with respect to the reactor site. Evaluate the potential hazard to the plant from the operation of these airfields.

2.14 Describe the recreational use of the Cedar River in the site vicinity for water-related activities such as swimming, boating, fishing, and water skiing. Provide data on the annual usage of the river for the various recreational activities.

2.15 The use of the Cedar River for irrigation is mentioned in FSAR Section 2.5.4.2. Provide data on locations downstream from the plant where water is withdrawn for irrigation, the quantity of water used, the acres of land irrigated, and the crops grown on the irrigated land.

2.16 Discuss the potential for upstream release of corrosive liquids including oils and evaluate their effect on the plant intake structures, the main condenser cooling system, and any other system which might be affected.

2.17 Are pipelines, quarries, mineral mines, or petroleum wells located within five miles of the plant site? If so, give their location with respect to the site and evaluate the effect on the plant of an explosion, fire, or other potential hazard from these sources.

2.18 Provide data on the total number of dairy cows and the total milk production within a 50 mile radius of the site.

2.19 The material presented in section 2.5 of the FSAR is inadequate to allow an independent evaluation to be made of your estimate of the peak runoff rate from a probable maximum flood (PMF), and corresponding static and dynamic consequences thereof on safety related facilities. Provide the following information to substantiate the estimate of PMF runoff, static and dynamic consequences thereof, and assurance that such an event would not cause a loss of safety related function:

a. Provide time and space estimates of the probable maximum precipitation used to estimate the PMF runoff. See section 2.4.3.1 of "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" (SF&C) for guidance.

b. Describe the ground absorption capability assumed during the probable maximum precipitation. Discuss historic precipitation losses, and substantiate that the PMF losses are conservatively low. See section 2.4.3.2 of SF&C for guidance.

c. Describe the hydrologic response characteristics of the basin to precipitation (such as unit hydrographs and flood routing characteristics), and provide verification of the use of such a runoff model to conservatively estimate the PMF; see section 2.4.3.3 of the SF&C for guidance. Provide the estimated peak discharge associated with the standard project flood referred to in FSAR paragraph 2.5.2.2, and provide substantiation of this estimate.

d. On a map, show the location of all dams and lakes in the basin with respect to the site. To help verify your conclusion that such facilities have no safety implications with respect to the site, provide a tabulation of their approximate channel distances from the plant, their heights, and storage volumes. Provide several (minimum of two) combined overbank and channel cross sections normal to the estimated centerline of the Cedar River at flood stage in the vicinity of safety related facilities. Provide water level estimates and their bases for the PMF in the vicinity of each safety related facility, including a water surface profile along the river property boundary. Show that the water level estimates are conservative by comparing coefficients used in reconstituting historical floods with those used to estimate PMF water levels. Show historical flood elevations on the PMF water surface profile. See section 2.4.3.5 of the SF&C for guidance.

e. Provide estimates of the maximum wave height and resulting runup which can occur as the result of a sustained 45 mph over water wind speed from a critical direction. As a result of the occurrence of such a wind coincident with the maximum PMF water level, provide estimates of the water level above grade at each safety related facility.

f. Describe the static and dynamic consequences of the occurrence of a PMF and coincident wind generated wave activity on each safety related facility which can be affected by such an event. Provide assurance that such an event cannot cause a loss of safety related function. See section 2.4.10 of the SF&C for guidance.

g. Describe any emergency measures required to prevent a loss of safety related function in the event the PMF and coincident wave action could reach plant facilities. See section 2.4.14 of the SF&C for guidance.

2.20 For evaluation of the adequacy of cooling water supply, additional information is needed as described below:

a. Provide a verified low-flow Cedar River rating curve at the intake structure which has been extrapolated to a flow of 30 cubic feet per second.

b. Provide a cross section through the intake structure, or reference other FSAR information, which illustrates the general layout of the facility.

c. Provide the minimum safety related flow rate and submergence elevation of the safety related river water pumps below which cavitation may be expected without pump throttling.

d. Compare the minimum safety related plant requirements with historical and projected future low river flow (or cross reference appropriate PSAR material).

e. Discuss the impact on water supply dependability of potential future upstream consumptive use.

f. Compare the water supply dependability available to the plant with Safety Guide 27, Ultimate Heat Sink.

g. Is FSAR Figure 2.5-2 based on mean daily discharges, or instantaneous flows?

See section 2.4.11 of the SF&C for guidance.

2.21 Provide a tabulation of surface water users (including their locations with respect to the plant, their demand, type of use, etc.) on the Cedar River downstream of the plant where such use can constitute pathways to man. Provide a map showing the location of ground water users within three miles of the site. Tabulate the owner, depth of well, pumpage and/or use rate, and type of use for each facility. Appropriate sampling methods may be used for depicting total population of wells. Include similar information for on-site wells. See sections 2.4.1.2, 2.4.12 and 2.4.13 of the SF&C for guidance.

2.22 Present water table and piezometric level maps confirming your contention in FSAR Section 2.5.3.2 that all wells are not in the line of groundwater flow from the plant. If wells now, or may in the future, exist down-gradient from the plant, evaluate the effects of the worst possible spill of liquid radioactive wastes which might enter the groundwater supply to these wells.

2.23 The city of Cedar Rapids, and possibly others, depend on the Cedar River indirectly for water supply and, under certain conditions, directly for emergency water supply. At approximately the river flows that would require maximum emergency withdrawals from the Cedar River, assume the worst inadvertent spill or accidental pumpage of liquid radioactive wastes into the Cedar River at the plant site, and discuss the consequences. Include estimates of: (a) maximum concentrations of radioactivity (with appropriate estimates of dispersion and dilution) at Cedar Rapids, (b) elapsed time of travel, and (c) duration of passage past water intakes.

2.24 Provide information of the intensity and frequency of occurrence of hail, ice storms and fog. The impact of these phenomena on evacuation plans should be considered.

2.25 As pointed out in Safety Guide 23, basic meteorological information must be available for assessing potentially adverse environmental effects such as might result from cooling tower operation. FSAR Section 2.4-5 mentioned dew point instrumentation in the field measurement program; however, no data or analysis resulting from these measurements were presented. Discuss your plans for the establishment of valid onsite baseline humidity conditions prior to the operation of the cooling towers.

2.26 Safety Guide 23 and portions of Title 10, Code of Federal Regulations referenced therein point out the need for onsite meteorological measurements for the assessment of the consequences of accidental or routine emissions to the atmosphere during the operation of a power reactor. Discuss your plans for a continuing onsite meteorological program.

2.27 Assuming the effective stack height to be the difference between stack-top elevation and topography elevation, provide a table of effective stack height versus distance for each of the 22.5 degree segments. Identify in each increment of distance the height of the highest point and its height with respect to the top of the stack (in meters). Considering the effective stack height thus obtained and the meteorological data collected at the site, provide annual average χ/Q values as a function of distance from the stack out to a distance of at least 5 miles. Data should be presented in the format used for tables in Sections 11 and 12 of FSAR Appendix E - On-Site Meteorological Data.

2.28 Is there not a discrepancy between χ/Q values presented on the last page of Section 8 of FSAR Appendix E - On-Site Meteorological Data and χ/Q values graphed on Figure 2 of the same volume? Discuss this point.

2.29 Specify the minimum vertical cross-sectional areas of the reactor building, the inside diameter of the stack at its exit, and the volumetric flow rate anticipated in the stack.

2.30 Amendment No. 10 to the PSAR, dated November 26, 1969, stated that the reactor building mat would be capable of carrying the imposed loads, and load combinations including seismic effects. Beneath the mat, the solution cavity could be as large as 16 feet wide at a depth of 2 feet. Indicate whether the mat was built for this condition. Add supplemental information concerning this design in item 4, Section 2.6.3.6.2.2, FSAR page 2.6-70.

4.0 REACTOR COOLANT SYSTEM

4.6 In reference to ferritic materials (including welds) of the reactor pressure vessel beltline, indicate whether the specifications included any additional imposed limits on residual elements (reportable and nonreportable) and requirements which were intended to reduce sensitivity to irradiation embrittlement.

4.7 For unstabilized stainless steel material of the austenitic type series 3xx used for components that are part of (a) the reactor coolant pressure boundary, (b) systems required for reactor shutdown, (c) systems required for emergency core cooling, (d) reactor vessel internals which are required for emergency core cooling, and (e) reactor vessel internals which are relied upon to permit adequate core cooling for any mode of normal operation or under postulated accident conditions, the following information should be provided:

a. A description of the material inspection program used to verify the non-susceptibility of unstabilized austenitic stainless steels to intergranular attack. If the procedures of ASTM A-262, Practice E were not employed, furnish a description of the test procedures.

b. A descripton of methods used for control of delta ferrite in austenitic stainless steel welds to avoid microfissuring in welds, especially as regards filler materials. Describe the associated welding procedure qualification. Describe methods used for determining delta ferrite content of the welds.

4.8 To demonstrate compliance with AEC General Design Criterion 30, which requires that means be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage, the following information should be provided:

a. The sensitivity of methods that will be used to determine coolant leakage from the reactor coolant pressure boundary.

b. With reference to your proposed maximum allowable unidentified leakage rate in the reactor coolant pressure boundary, if the limit is greater than 1 gpm:

(1) The length of a through-wall crack that would leak at the rate of the proposed limit as a function of wall thickness.

(2) The ratio of that length to the length of a critical through-wall crack, based on the application of the principles of fracture mechanics.

(3) The mathematical model and data used in such analyses.

(4) Experimental data confirming validity of the analyses described in b (1), (2) and (3).

4.9 To demonstrate compliance with Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Reactor Coolant Systems," provide a description of equipment and procedures under development for remote inservice inspection using the access provisions afforded by the plant design.

4.10 Additional information is needed to evaluate the degree of compliance with the test methods and acceptance criteria of the recently revised ASME Boiler and Pressure Vessel Code, Section III, fracture toughness rules (Code Case 1514), as applied to the fracture toughness data obtained for all ferritic materials of the reactor vessel. For the reactor vessel plates, forgings, and qualification welds, provide data depicting temperatures at which "weak" direction Charpy V-notch specimens exhibit at least 35 mils lateral expansion and not less than 50 ft-lbs absorbed energy.

4.11 The "worst case" curve relating change in transition temperature to neutron fluence and used to construct the "Minimum Reactor Pressurization Temperature" curve as shown in Figure 3.6.1, Appendix B of the FSAR, is not sufficiently conservative. Provide a statement that proposed operating limitations during startup and shutdown of the reactor vessel will use as a guide Appendix G, "Protection Against Non-Ductile Failure," of the recently revised ASME Boiler and Pressure Vessel Code, Section III, fracture toughness rules (Code Case 1514).

4.12 The curves set forth in Figure 4.2-5 set forth minimum cold hydrotest temperatures from NEDO 10115. This General Electric document, NEDO 10115, is neither listed as a reference nor as a report submitted to the AEC. Thus, the report is neither identifiable nor available. Provide a copy of the report and include the full title as a reference for Section 4.

4.13 On page 4.2-21 of the FSAR it is mentioned that the reactor vessel pressurization at end-of-life cannot begin until the coolant temperature exceeds 212°F. In the discussion on nil-ductility transition (NDT) temperature, it is evident that metal temperature is the key parameter for determining the NDT temperature of the reactor vessel. Discuss use of water coolant temperature as a measure of reactor vessel metal temperature in assuring adequate margin above the NDT temperature of the reactor vessel.

12.0 PLANT STRUCTURES AND SHIELDING

12.5 Describe the design philosophy and the methods used to provide structural strength to the numerous interior concrete block walls. Specifically discuss design to resist seismic forces. Discuss the design and provide drawings for the concrete block walls of the Category I seismic design diesel-generator rooms.

12.6 In sketch form provide the connection details that are designed to connect the precast concrete wall panels to the exterior of the reactor building. Indicate the controlling failure mechanism for the connections and the safety margin against the loss of an individual panel.

12.7 In Item 4, Section 12.2.1, FSAR page 12.2-2, it is stated that the spent fuel storage pool, the reactor basin cavity and dryer-separator pool consist of lined, reinforced concrete structures. The lining was noted as being either stainless steel or epoxy. If epoxy is used, provide information concerning its ability to strain and span the expected crack widths. Provide the magnitude and spacing of predicted cracks and the engineering properties of the epoxy.

APPENDIX G - RESPONSE TO THE AEC SAFETY GUIDES

QG15.1 The criteria on deformations for reinforcing bars have been stated to be not applicable. The criteria set forth in ASTM A-615 are to be met, including the deformation criteria. It is not necessary to meet the deformation criteria by demonstrating the adequacy of a Cadweld splice in a tension test since no Cadwelds were used. It is, however, necessary to assure that the reinforcing steel furnished has deformations in accordance to Section 6 of ASTM A-615 by some other means such as random measurements. Indicate how the deformation requirements have been met.