NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL
(TEMPORARY FORM)

CONTROL NO: 1219

FILE:

FROM: Iowa Elec. Light & Power Cedar Rapids, Iowa LeeLiu	DATE OF DOC 2-6-76	DATE REC'D 2-9-76	LTR XXX	TWX	RPT	OTHER
TO: George Lear	ORIG 1 Signed	CC 0	OTHER	SENT NRC PDR XXX SENT LOCAL PDR XXX		
CLASS / UNCLASS XXX / PROP INFO	INPUT	NO CYS REC'D 1	DOCKET NO: 50-331			

DESCRIPTION:
Letter notarized 2-6-76..trans the following...
Letter advising that the DAE Center can continue to operate without undue risk to the public health and safety ... W/Attached Evaluation of Mark I Containment Capability...

(1 Copy Received)

PLANT NAME: Duane Arnold

ENCLOSURES:

DO NOT REMOVE

ACKNOWLEDGED

FOR ACTION/INFORMATION ENVIRO SAB 2-10-76

ASSIGNED AD

BRANCH CHIEF LEAR W/6

PROJECT MANAGER PAULSON

LIC. ASST. K PARRISH W/16 CYS ACRS

ASSIGNED BRANCH CHIEF

PROJECT MANAGER

LIC ASST W/ ACRS

INTERNAL DISTRIBUTION

REG FILES
~~NRC~~ PDR
~~OELD~~
~~GOSSICK/STAFF~~
~~I&E~~ (2)
MIPC

PROJECT MANAGEMENT
~~BOYD~~
P. COLLINS
HOUSTON
PETERSON
MELTZ
HELTEMES

SYSTEMS SAFETY
HEINEMAN
SCHROEDER

ENGINEERING
MACCARY
KNIGHT
SIHWEIL
PAWLICKI

REACTOR SAFETY
~~ROSS~~
~~NOVAK~~
~~ROSETOCZY~~
CHECK

PLANT SYSTEMS
~~TEDESCO~~
BENAROYA
~~LAINAS~~
IPPOLITO

OPERATING REACTORS
STELLO

OPERATING TECH.
~~EISENHUT~~
SHAO
~~BAER~~
SCHWENCER
GRIMES

SITE SAFETY & ENVIRO ANALYSIS
DENTON
MULLER

ENVIRO TECH.
ERNST
BALLARD
SPANGLER

SITE TECH.
GAMMILL
STEPP
HULMAN

SITE ANALYSIS
VOLLMER
BUNCH
J. COLLINS
KREGER

AT&I
SALTZMAN
RUTBERG

MISCELLANEOUS
CASE

EXTERNAL DISTRIBUTION

~~LOCAL~~ PDR Cedar Rapids, Iowa
~~TIC~~
~~NSIC~~
ASLB

NATIONAL LAB W/ CYS
REGION V-I&E-(WALNUT CREEK)
LA PDR
CONSULTANTS

BROOKHAVEN NAT. LAB
ULRIKSON (ORNL)

IOWA ELECTRIC LIGHT AND POWER COMPANY

General Office
CEDAR RAPIDS, IOWA

DOCKETED
USNRC
FEB 9 1976
Mail Section
Docket Clerk

LEE LIU
VICE PRESIDENT - ENGINEERING

Regulatory

File Cy.

February 6, 1976
IE-76-209

RECEIVED
1976
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Mr. George Lear, Chief
Operating Reactors Branch 3
Division of Operating Reactors
Nuclear Regulatory Commission
Washington, D.C. 20555

50-331

Dear Mr. Lear:

During the January 28, 1976 Mark I Containment Evaluation meeting in Bethesda the following conclusions were reached:

1. The individual owners would provide a written evaluation of continued operation of their plant by February 6, 1976 in light of the information developed to date.

2. The individual owners would submit plant specific analyses within two to three months, and

3. Answers to the NRC questions dated January 7, 1976 would be forwarded by April 30, 1976.

The attachment to this letter is the response to Item 1 above. The conclusion of this letter and attachment thereto is that the Duane Arnold Energy Center can continue operation without undue risk to the public health and safety while the Mark I containment evaluation continues. This conclusion is based on a number of relevant factors including the engineering judgment that the containment would maintain its integrity, and that inherent conservatisms are contained in the load and structural analyses.

Three signed originals and 37 copies of this letter are transmitted herewith. This letter is true and accurate to the best of my knowledge and belief.

Iowa Electric Light and Power Company

1219

By: [signature]
Lee Liu
Vice President, Engineering

LL/KAM/D
cc: D. Arnold
J. Newman
W. Paulson
J. Keppler

Sworn and Subscribed to before me on this 6th day of February, 1976.

Georgia F. Marlowe
NOTARY PUBLIC
State of Iowa
Commission Expires
September 30, 1976

Georgia F. Marlowe
Notary Public in and for the State of Iowa.

EVALUATION OF MARK I CONTAINMENT CAPABILITY
AND CONTINUED OPERATION FOR THE DUANE ARNOLD ENERGY CENTER,
Dockets 50-331

I. Background

The Mark I owners group was formed as a result of the April, 1975 request by the United States Nuclear Regulatory Commission (NRC) for the purpose of obtaining additional information on the design of the Mark I containments used with the General Electric (GE) designed boiling water reactors (BWR) nuclear steam supply systems. Since its formation, Iowa Electric has been an active member of the Mark I containment owners group and followed closely the development of the program conclusions. Previous letters from GE and Iowa Electric have outlined the short and long-term evaluation which is in process.

GE was retained as the Mark I Containment Owners Group Project Manager. Bechtel was retained by GE as a consultant for the purpose of structural evaluation. Teledyne Materials Research (TMC) was retained by GE to perform an overview function for load development, structural evaluation, and structural criteria establishment. NUTECH was retained by the Owners group in November, 1975 to act as the utility group's technical representative and to keep the utilities informed of program progress on a continuing basis.

The initial task for the Mark I owners group during the short term program was evaluating the integrity of the containment vent system and vent system supports assuming most probable loads, with the governing criteria being maintenance of containment functions and ECCS piping. The results of this effort,

which concluded that the vent system integrity would be maintained when subjected to the most probable pool swell loads, are documented in the five volume report which was submitted to the NRC in September, 1975.

In order to supplement the general studies being conducted by General Electric and its consultant Bechtel for the Mark I owners group, Iowa Electric retained NUTECH as an independent consultant. NUTECH has conducted a parallel evaluation of the structural integrity of the vent systems for the Duane Arnold Energy Center (DAEC) when subjected to the most probable pool swell loads and has confirmed the work done on behalf of the Mark I owners group as it relates to this unit.

Subsequent to the submittal of the Short Term Program (STP) report, Addendum 1 to that report was prepared and submitted to the NRC in December, 1975. Documented in that addendum are analyses of the relief valve discharge piping when subjected to pool swell impact and drag loads. This addendum provides the basis for concluding that the integrity of the relief valve discharge piping is assured.

Also included in Addendum 1 is documentation of the structural integrity testing of a representative vent line bellows assembly when subjected to pool swell loads. Since the bellows assemblies on the DAEC are located outside the torus and as such are not subjected to pool swell impact loads, this test was not required to demonstrate the integrity of the DAEC bellows. Nevertheless, the observed behavior demonstrates the inherent reserve

capacity for welded steel structural components to maintain their leak tight integrity even when subjected to large deformations.

The one remaining item which is in the scope of the STP but which has not been documented in either the STP Report or its Addendum is the suppression chamber torus support system evaluation. This item was discussed with the NRC staff during meetings on January 7 and 8, 1976 in San Jose and again on January 28, 1976 in Bethesda.

On January 28, 1976, representatives of GE and of the Mark I containment owners group met with members of the NRC staff and provided the latest information developed by the Mark I containment evaluation program. At the conclusion of this meeting, the staff representatives requested that each plant owner provide a letter documenting the basis for continued operation. More specifically, the staff requested that:

(a) Each operating plant submit in writing, no later than February 6th, the plant unique basis justifying continued operation.

(b) Each plant with calculated uplift equal to or greater than 0.2 inches should include a plant specific analysis of torus related ECCS piping.

(c) Each plant identified as having a calculated downward load to capability ratio of 0.9 or greater

in the tabulation presented by H. A. Franklin at the meeting should emphasize the downward loading on the pertinent critical element.

This letter presents Iowa Electric's information with regard to the DAEC per your request.

Iowa Electric has reviewed this information and has concluded that the DAEC can continue operation without undue risk to the public health and safety while the Mark I containment evaluation continues.

The bases for this conclusion are addressed below. For the purpose of explanation, it is convenient to group the bases as follows:

(a) Load considerations - upward and downward.

(b) Structural response - upward and downward.

II. Containment Structural Evaluation

A. Load Considerations

One-twelfth scale tests of a Mark I torus were run to obtain upward and downward loads on the torus due to postulated LOCA events. Recognizing the uncertainties inherent in this scaling and the fact that the DAEC has only one downcomer, the methods used in the determination exceed a most probable load analysis approach by incorporating many conservatisms in data interpretation and analytical technique.

1. Upward Load Conservatisms

The upward pressure load is sensitive to the pressure history of the drywell following a postulated LOCA because the driving

force for the pool swell and the resulting torus air space compression are increased with a greater drywell pressurization rate. The upward pressure load on the torus has been defined for the Short Term Program by application of the calculated FSAR drywell pressurization rate. Specifically, the 1/12th scale tests were run and analyzed to obtain loads based on the FSAR pressurization history. This pressure history has been used to establish the drywell design pressure and it has been biased toward high values for this purpose. The FSAR pressure history assumes an instantaneous break (mass fluxes evaluated using the Moody Critical flow model assuming slip), no steam condensation in the drywell, and a homogenous air-steam-liquid flow mixture in the vent. This results in a high pressurization rate and increases the upward load definition.

The conservatism in the upward pressure load produced by the application of the FSAR pressure rate, is illustrated by considering the reduction in mass flux which occurs with the application of the homogeneous rather than the slip formulation of the Moody Critical flow model. Even for the 20 Btu/lb mass subcooled liquid in the recirculation system, the homogeneous model shows a reduction in the mass flux from 8100 to 7100 lb m/sec. ft^2. Using the sensitivity curves, this flow reduction produces a reduction in the upward pressure load of two percent. The other conservatisms in the FSAR pressure history will add to this margin.

Another conservatism for the upward load used in the Short Term Program is the assumption of a 100% air flow in the vent system. More consistent assumptions aimed at determining the most probable load basis are possible. One alternative is to apply the FSAR homogeneous air-steam-liquid vent flow assumption for both the pressure history and the noncondensible flow rate into the bubble. The other alternative is to assume 100% air vent flow for both the pressure history and the flow rate into the bubble. If, for example, the former is evaluated, the non-condensible bubble flow rate is reduced by a factor of three and the sensitivity analysis for (A pool/A vent) shows that the maximum upward load will be reduced by a factor of two.

Another contribution to the total upward load on the torus structure is the impact load on the vent header. The impact pressure on the vent header for the Short Term Program was determined by applying the impact velocity measured in the 1/12th Scale tests and the results of the Pool Swell Test Facility (PSTF) impact data. However, the PSTF data were obtained for the impact of a slug having a thickness greater than the diameter of the target. In contrast, the 1/12th Scale slug thickness is thinner than the vent header. The reduced slug thickness in the torus allows the liquid to be quickly decelerated under the header immediately following impact. This deceleration, which was observed in the 1/12th Scale

tests, would be expected to yield a lower impact load. Indeed, the impact pressure history measured in the 1/12th Scale test by a strain gage on the vent header was a factor of three less in magnitude and three times longer in duration. The more conservative vent header impact pressure was used in the analysis as an added conservation. The 1/12th Scale test results will be substantiated by future 1/6th Scale testing.

2. Downward Load Conservatisms

Similar conservatisms have been used in defining the downward pressure load on the torus. The calculated FSAR pressure rate was also used to establish the downward pressure load on the torus. If the finite opening time of the break, reduced mass flux at the break, and steam condensation in the drywell were accounted for, the drywell pressure at vent clearing would be less and the downward pressure load would be reduced. The reduction in the downward pressure load for using a mass flux of 7100 instead of 8100 lb m/sec. ft^2 is 6 percent.

The data used from the 1/12th Scale tests to define the downward pressure loads was also analyzed in a conservative manner. There was some variation in the maximum downward pressure loads measured for the medium orifice runs considered as a group and for the large orifice runs considered as a group. Instead of averaging the loads measured for the medium and large orifice runs, the greatest magnitude downward pressure loads were identified for both orifice sizes. The Reference Plant downward pressure load was then determined by interpolating between the maximum of the maximum downward pressure loads.

The analysis of the 1/12th Scale test results also did not take credit for any reduction in the downward force due to three dimensional effects and pressure attenuation. The submerged pressure transducers are located at the mid-width of the test section and will feel most directly the pressure of the bubble formed at the downcomers and the water jet forces. Both the bubble pressure and the water jet force will attenuate as one moves circumferentially away from directly below the downcomers. However, since the pressures measured by the transducers were assumed to act uniformly over the width of the test section, a higher than actual reaction force was calculated.

In the typical torus, the downcomers are not spaced uniformly leaving a large section below the vent pipe where the influence of the downcomers is decreased. The downward pressure load produced by the bubble pressure at the downcomers and the water jet forces will be reduced in this section because of the increased distance to the nearest downcomers. However, the pressure loads should not be significantly increased where the downcomers are closely spaced because the measured pressure load of 16.33 psid approaches the driving pressure - the drywell pressure is 17.0 psid at the time of vent clearing. Therefore, due to three dimensional effects and variable downcomer spacing the maximum downward pressure cited for the reference plant of 16.33 psid is conservative.

B. Structural Considerations

1. Downward Loads

GE and Bechtel have estimated the capabilities of the various structural elements of the torus support system, and those results

were presented to the NRC at the January 28, 1975 meeting. Bechtel performed an analysis of the reference plant using a two-dimensional ring model. The results of this analysis were ratioed to obtained unique results. NUTECH, as Iowa Electric's consultant, utilized the results of a three-dimensional finite element analysis of the reference plant to compute column loads directly for the DAEC torus support columns. The Bechtel results were modified to provide better correlation with results from the three-dimensional analysis of the reference plant performed by NUTECH. Table 1 presents the maximum load, component ultimate strength and load to capability ratio for the weakest component of each plant evaluated. Additionally, a detailed analysis of the column connections was performed by NUTECH to establish a minimum ultimate capability for the connections.

An evaluation has been made to establish a lower bound for the ultimate capability of each component in the load path of the torus support system for downward loads. Specifically, this includes the connection of the column to the torus shell, and the columns themselves, and the pin connection of the column to its base plate. This has been done for both the inside and the outside columns as the loads are different. These lower bound ultimate capabilities have then been compared with the strength required to accommodate the currently defined "most probable" downward load. It is convenient to express these results in terms of the ratio of downward load divided by lower bound capacity. This ratio is less than 0.90 for the following links in the load path:

Component	Ratio
Outside column	0.80
Outside column connection to shell	0.46
Inside column	0.66
Inside column connection to shell	0.38

It is important to recognize that the lower bound for column capacity is controlled by local yielding in the top of the column due to a combination of axial load and bending moment. At no time does the entire cross-section of the column reach a state of general yielding. Specifically for the inside columns for the DAEC, the average stress across the column area is 20.2 ksi, which is 56% of yield. For the outside columns it is 25.8 ksi or 72% of yeild.

Actual material tests have shown margins beyond calculated failure. During the short-term program, structural tests were performed on a number of separate components, including the downcomer header assembly and bellows. Comparisons of test results with analytical predictions similar to the ones used here have confirmed that analytical procedures are conservative and under estimate the actual component strength.

2. Upward Loads.

As reported by GE on January 28, 1976, uplift is not a concern for the DAEC. The hydrostatic load of the contained water and the dead weight of the suppression chamber is greater than the upward pressure load which results from the most probable pool swell load. The vent header reaction produces some additional uplift load, but

due to its short duration and oscillatory nature, the computed uplift is less than 0.1 inch. Even if this uplift were to occur, the resulting column loads at the end of the transient are less than those which result at the start of the transient. Also, as discussed below, a 0.1 inch vertical movement of the torus represents no concern relative to the ECCS piping. Clearly, thermal movements of this magnitude are routinely accommodated by the connected piping.

The results of an engineering evaluation of the ECCS piping attached to the DAEC torus show that the torus uplift deflection which can be tolerated by the ECCS piping is approximately 1.25 inches based on the results of present analyses of the piping in accordance with the ASME Section III, Class 2 stress limit of 2.4 SH. By considering the plastic load carrying capacity of the pipe based upon a 3^{o} limit on plastic bending deflection, the allowable ECCS piping deflection would be two to three times larger. Consideration of stress limits in excess of code allowables is reasonable on a temporary basis, since the subject line pressures and operating temperatures are relatively low and the actual flexibility of the piping is greater than considered in the present analyses. These factors increase the confidence that the ECCS lines could withstand upward torus deflections greater than those mentioned above and still maintain their function.

Thus, it can be concluded that our present evaluation, indicates that if torus uplift were to occur, the structural integrity of the torus and ECCS piping attached to the torus would not be compromised.

III. Probability of Occurrence

By GE calculations, the only postulated break which would release sufficient energy at a rate sufficient to load the torus to the extent here analyzed is a break equal to the Design Basis Accident LOCA. Furthermore, only breaks of greater than 90% of the Design Basis Accident LOCA develop uplift due to pressure loads. Only the 22" reactor coolant recirculation lines can generate a break of such size. When line losses are considered only the section of pipe from the vessel nozzle to the recirculation pump suction valve result in calculated uplift.

The configuration of large pipe sizes and short runs of the reactor coolant recirculating loops is comparable to that of pressure vessels. Estimates of pressure vessel failure are therefore appropriate for use in evaluating the probabilities of rupture of such large pipes. On that basis, a large LOCA due to rupture in the 22" recirculation pipelines is a highly unlikely event, especially when considered over the short term of less than a reactor year.

1. The periodic inservice inspection in accordance with ASME Section XI, coupled with the leak before break feature of the ductile 304 stainless steel material and existing containment leak detection systems designed to detect pipe leaks, provide additional assurance that an instantaneous LOCA will not occur for the 22" lines.

2. No cracks due to intergranular stress corrosion cracking -- a possible damage mechanism pertinent to stainless steel -- have been found in any BWR piping systems using sizes 18" and larger.

IV. ACTIONS TO CONFIRM OR INCREASE SAFETY MARGINS

While the previous analyses indicate that the likelihood of a major loss of coolant accident involving the 22" recirculation lines is very small and that the containment would maintain its integrity even in the event of such an accident, the following actions are being investigated on a parallel and expedited basis to confirm, and if appropriate, increase calculated margins.

A. Verification of Bechtel-General Electric Analyses

A verification of the plant specific parameters used in the Bechtel-General Electric analyses is under way. To date discrepancies have not been found that would make the calculated numbers less conservative. We expect to have this verification complete within two weeks. Additionally, a two dimensional, simple, ring beam model analysis for the DAEC is being developed. We expect to report results from this analysis in the detailed plant specific analysis which is due in three months, as requested by the Staff in the January 28 meeting.

B. Model Testing

We have authorized General Electric Company to proceed with a 1/6 scale model test in order to increase confidence in the results of the 1/12 scale model tests which were used for the most probable load determination and the GE-Bechtel analyses to date.

C. Loading Maldistribution

We have begun investigation of the possibility of measuring the individual column support loadings at the DAEC. Potential load maldistributions among the 32 columns have not been considered in the calculations to date.

D. In Service Inspection

During the refueling outage scheduled to commence February 15, 1976, the In Service Inspection Program will be accelerated to include the recirculation loop suction line, thereby providing further assurance with respect to the safe operation of the plant.

TABLE 1

DUANE ARNOLD ENERGY CENTER

	Component	Load	Estimated Minimum Ultimate Capacity	Ratio Load/Capacity
1.	Inside Column	530	858*	.62
2.	Inside Connection to Shell	530	1100	.48
3.	Inside Connection at Base	530	1520	.35
4.	Outside Column	675	858*	.79
5.	Outside Connection to Shell	675	1100	.61
6.	Outside Connection at Base	675	1520	.44

* This capacity can be adjusted by ratio of actual yield to minimum specified yield.