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Benard C. Rusche	1 Signed	0		SENT NRC PDR XXX SENT LOCAL PDR XXX

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DESCRIPTION:

Advising the name of the Firm of Conner, Hadlock & Knotts, has been changed to Conner & Knotts....

ENCLOSURES:

ACKNOWLEDGED

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November 3, 1975

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11-6-75
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Mr. Benard C. Rusche
Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Rusche:

This is to advise you that the name of our firm has been changed to CONNER & KNOTTS. We would appreciate it if you would have this change reflected on the NRC's service lists for the cases listed on the attachment.

Sincerely,

Nancy L. Hickman

Nancy L. Hickman
Executive Assistant

Attachment

NLH/mwm

cc: Mrs. Sybil Kari
Mrs. Nancy Dube

12768

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	(McGuire 2)	50-370
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	(Catawba 2)	50-414
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	(Cherokee 2)	50-492
	(Cherokee 3)	50-493
GULF STATES UTILITIES	(River Bend 1)	50-458
	(River Bend 2)	50-459
	(Blue Hills 1)	50-510
	(Blue Hills 2)	50-511
MISSISSIPPI POWER & LIGHT COMPANY	(Grand Gulf 1)	50-416
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	(Peach Bottom 3)	50-278
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	(Limerick 2)	50-353
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	(Salem Station 2)	50-311
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	(WPPSS 1)	50-460
	(WPPSS 4)	50-513
	(WPPSS 3)	50-508
	(WPPSS 5)	50-509

OCT 15 1975

Docket Nos. 50-269
50-270
and 50-287

Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President
Steam Production
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Gentlemen:

RE: Oconee Nuclear Station Units 1, 2, and 3

The purpose of this letter is to inform you of a potential safety question which has been raised regarding the design of reactor pressure vessel support systems for pressurized water reactors (PWR`s).

On May 7, 1975 the NRC was informed by a licensee that certain transient loads on the reactor vessel support members that would result from a postulated reactor coolant pipe rupture immediately adjacent to the reactor vessel had been underestimated in their original design analyses.

It is the NRC staff`s opinion that the question related to the treatment of transient loads in the design of reactor vessel support systems may apply to other PWR facilities, especially those for which the design analyses were performed some time ago. We have therefore initiated a systematic review of this matter to determine how these loads were taken into account on other PWR facilities, and what, if any, corrective measures may be required for specific facilities.

The results of licensee studies reported to date indicate that, although the margins of safety may be less than originally intended, the reactor vessel support system would retain sufficient structural integrity to support the vessel and that the ultimate consequences of this postulated accident which could affect the general public are no worse than originally stated. We have not completed our independent evaluation of these studies. However, based on the results of our evaluation of this phenomenon to date and in recognition of the low probability of the particular pipe rupture which could lead to additional transient loads on the support systems, we conclude that continued reactor operation and continued licensing of facilities for operation are acceptable while we conduct our generic review.

We request that you review the design bases for the reactor vessel support system for your facilities to determine whether the transient loads described in the enclosure were taken into account appropriately in the design. Please inform us of the results of your review within 30 days.

The attachments to the enclosure are provided to indicate the information that could be needed, should we determine, on the basis of your review, that a reassessment of the vessel support design is required.

We are continuing to evaluate and review the methodology for calculating the subcooled blowdown loads with the nuclear steam system suppliers. You should contact your nuclear steam system supplier for information regarding these calculations if necessary to complete your review.

This request for generic information was approved by GAO under a blanket clearance number B-180225 (R0072). This clearance expires July 31, 1977.

Sincerely,

Original signed by
R. A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosure:
Statement of the Problem

cc w/enclosure:
See next page

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cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Mr. Troy B. Conner
Conner, Hadlock & Knotts
1747 Pennsylvania Avenue, NW
Washington, D. C. 20006

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ENCLOSURE

STATEMENT OF THE PROBLEM

In the unlikely event of a PWR primary coolant system pipe rupture in the immediate vicinity of the reactor vessel, transient loads originating from three principal causes will be exerted on the reactor vessel support system. These are:

1. Blowdown jet forces at the location of the rupture (reaction forces),

2. Transient differential pressures in the annular region between the vessel and the shield, and

3. Transient differential pressures across the core barrel within the reactor vessel.

The blowdown jet forces are adequately understood and design procedures are available to account for them. Both of the "differential pressure" forces, however, are three-dimensional and time dependent and require sophisticated analytical procedures to translate them into loads acting on the reactor vessel support system. All of the loads are resisted by the inertia and by the support members and restraints of other components of the primary coolant system including the reactor pressure vessel supports.

The transient differential pressure acting externally on the reactor vessel is a result of the flow of the blowdown effluent in the reactor cavity. The magnitude and the time dependence of the resulting forces depends on the nature and the size of the pipe rupture, the clearance between the vessel and the shield and the size and location of the vent openings leading from the cavity to the containment as a whole. For some time refined analytical methods have been available for calculating these transient differential pressures (multi-node analyses). The results of such analyses indicate that the consequent loads on the vessel support system calculated by less sophisticated methods may not be as conservative as originally intended for earlier designs. Attachment 1 to this enclosure provides for your information a list of information requests for which responses could be needed for a proper assessment of the impact of the cavity differential pressure on the design adequacy of the vessel support system for a power plant.

The controlling loads for design purposes, however, appear in typical cases to be those associated with the internal differential pressures across the core barrel. The internally generated loads are due to a momentary differential pressure which is calculated to exist across the core barrel when the pressure in the reactor annular region between the core barrel and vessel wall in the vicinity of the ruptured pipe is assumed to rapidly decrease to the saturation pressure of the primary coolant due to the outflow of water. Although the depressurization wave travels rapidly around the core barrel, there is a finite period of time during which the pressure in the annular region opposite the break location is assumed to remain at, or near, the original reactor operating pressure. Thus, transient asymmetrical forces are exerted on the core barrel and the vessel wall which ultimately result in transient loads on the support systems. These are the loads which were underestimated by the licensee originally reporting this problem and which may be underestimated in other cases. They are therefore of generic concern to the staff. Attachment 2 to this enclosure provides for your information a list of information requests for which responses would be needed for a proper assessment of the impact that the vessel internal differential pressure, in conjunction with the other concurrent loads, could have on the design adequacy of the support system.

In that there are considerable differences in the reactor support system designs for various facilities and probably in the design margins provided by the designers of older facilities, the underestimation of these "differential pressure" loads may or may not result in a determination that the adequacy of the vessel support system for a specific facility is questionable. Since local failures in the vessel supports (such as plastic deformation) do not necessarily lead to the failure of the supports as an integral system, there may be some limited reactor vessel motion provided that no further significant consequences would ensue and the emergency core cooling systems (ECCS) would be able to perform their design functions.

ATTACHMENT 1

CONTAINMENT SYSTEMS BRANCH

REQUEST FOR ADDITIONAL INFORMATION

In the unlikely event of a pipe rupture inside major component subcompartments, the initial blowdown transient would lead to non-uniform pressure loadings on both the structures and enclosed components. To assure the integrity of these design features, we request that you perform a compartment multi-node pressure response analysis to provide the following information:

(a) The results of analyses of the differential pressures resulting from hot leg and cold leg (pump suction and discharge) reactor coolant system pipe ruptures within the reactor cavity and pipe penetrations.

(b) Describe the nodalization sensitivity study performed to determine the minimum number of volume nodes required to conservatively predict the maximum pressure within the reactor cavity. The nodalization sensitivity study should include consideration of spatial pressure variation; e.g., pressure variations circumferentially, axially and radially within the reactor cavity.

(c) Provide a schematic drawing showing the nodalization of the reactor cavity. Provide a tabulation of the nodal net free volumes and interconnecting flow path areas.

(d) Provide sufficiently detailed plan and section drawings for several views showing the arrangement of the reactor cavity structure, reactor vessel, piping, and other major obstructions, and vent areas, to permit verification of the reactor cavity nodalization and vent locations.

(e) Provide and justify the break type and area used in each analysis.

(f) Provide and justify values of vent loss coefficients and/or friction factors used to calculate flow between nodal volumes. When a loss coefficient consists of more than one component, identify each component, its value and the flow area at which the loss coefficient applies.

(g) Discuss the manner in which movable obstructions to vent flow (such as insulation, ducting, plugs, and seals) were treated. Provide analytical justification for the removal of such items to obtain vent area. Provide justification that vent areas will not be partially or completely plugged by displaced objects.

(h) Provide a table of blowdown mass flow rate and energy release rate as a function of time for the reactor cavity design basis accident.

(i) Graphically show the pressure (psia) and differential pressure (psi) responses as functions of time for each node. Discuss the basis for establishing the differential pressures.

(j) Provide the peak calculated differential pressure and time of peak pressure for each node, and the design differential pressure(s) for the reactor cavity. Discuss whether the design differential pressure is uniformly applied to the reactor cavity or whether it is spatially varied. (Standard Review Plan 6.2.1.2, Subcompartment Analysis attached, provides additional guidance in establishing acceptable design values, for determining the acceptability of the calculated results.)

U.S. NUCLEAR REGULATORY COMMISSION February, 1975

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.2.1.2 SUBCOMPARTMENT ANALYSIS

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Mechanical Engineering Branch (MEB)
Core Performance Branch (CPB)
Auxiliary and Power Conversion Systems Branch (APCSB)

I. AREAS OF REVIEW

The CSB reviews the information presented by the applicant in the safety analysis report concerning the determination of the design differential pressure values for containment subcompartments. A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within this volume. A short-term pressure pulse would exist inside a containment subcompartment following a pipe rupture within this volume. This pressure transient produces a pressure differential across the walls of the subcompartment which reaches a maximum value generally within the first second after blowdown begins. The magnitude of the peak value is a function of several parameters, which include blowdown mass and energy release rates, subcompartment volume, vent area, and vent flow behavior. A transient differential pressure response analysis should be provided for each subcompartment or group of subcompartments that meets the above definition.

The CSB review includes the manner in which the mass and energy release rate into the break compartment were determined, nodalization of subcompartments, subcompartment vent flow behavior, and subcompartment design pressure margins. This includes a coordinated review effort with the CPB. The CPB is responsible for the adequacy of the blowdown model.

The CSB review of the mass and energy release rates includes the basis for the selection of the pipe break size and location within each subcompartment containing a high energy line and the analytical procedure for predicting the short-term mass and energy release rates.

The CSB review of the subcompartment model includes the basis for the nodalization within each subcompartment, the initial thermodynamic conditions within each subcompartment, the nature of each vent flow path considered, and the extent of entrainment assumed in the vent flow mixture. The review may also include an analysis of the dynamic characteristics of components, such as doors, blowout panels, or sand plugs, that must open or be removed to provide a vent flow path, and the methods and results of components tests performed to demonstrate the validity of these analyses. The analytical procedure to determine the loss coefficients for each vent flow path and to predict the vent mass flow rates, including flow correlations used to compute sonic and subsonic flow conditions within a vent, is reviewed. The design pressure chosen for each subcompartment is also reviewed. On request from the APCSB, the CSB evaluates or performs pressure response analyses for subcompartments outside containment.

The MEB is responsible for reviewi[illegible] the acceptability of the break locations chosen and of the design criteria and provision[illegible] methods employed to justify limited pipe motion for breaks postulated to occur within subcompartments (See Standard Review Plan 3.6.2).

II. <u>ACCEPTANCE CRITERIA</u>

1. The subcompartment analysis should incorporate the following assumptions:

    a. Break locations and types should be chosen according to Regulatory Guide 1.46 for subcompartments inside containment and to Branch Technical Position MEB 3-1 (attached to Standard Review Plan 3.6.2) for subcompartments outside containment. An acceptable alternate procedure is to postulate a circumferential double-ended rupture of each high pressure system pipe in the subcompartment.

    b. Of several breaks postulated on the basis of a, above, the break selected as the reference case for subcompartment analysis should yield the highest mass and energy release rates, consistent with the criteria for establishing the break location and area.

    c. The initial plant operating conditions, such as pressure, temperature, water inventory, and power level, should be selected to yield the maximum blowdown conditions. The selected operating conditions will be acceptable if it can be shown that a change of each parameter would result in a less severe blowdown profile.

2. The analytical approach used to compute the mass and energy release profile will be accepted if both the computer program and volume noding of the piping system are similar to those of an approved emergency core cooling system (ECCS) analysis. The computer programs that are currently acceptable include SATAN-VI (Ref. 24), CRAFT (Ref. 23), CE FLASH-4 (Ref. 25), and RELAP3 (Ref. 21), when a flow multiplier of 1.0 is used with the applicable choked flow correlation. An alternate approach, which is also acceptable, is to assume a constant blowdown profile using the initial conditions with an acceptable choked flow correlation. When RELAP-4 is accepted by the staff as an operational ECCS blowdown code, it will be acceptable for subcompartment analyses.

3. The initial atmospheric conditions within a subcompartment should be selected to maximize the resultant differential pressure. An acceptable model would be to assume air at the maximum allowable temperature, minimum absolute pressure, and zero percent relative humidity. If the assumed initial atmospheric conditions differ from these, the selected values should be justified.

Another model that is also acceptable, for a restricted class of subcompartments, involves simplifying the air model outlined above. For this model, the initial atmosphere within the subcompartment is modeled as a homogeneous water-steam mixture with an average density equivalent to the dry air model. This approach should be limited to subcompartments that have choked flow within the vents. However, the adequacy of this simplified model for subcompartments having primarily subsonic flow through the vents has not been established.

4. Subcompartment nodalization schemes should be chosen such that there is no substantial pressure gradient within a node, i.e., the nodalization scheme should be verified by a sensitivity study that includes increasing the number of nodes until the peak calculated pressures converge to small resultant changes.

5. If vent flow paths are used which are not immediately available at the time of pipe rupture, the following criteria apply:

   a. The vent area and resistance as a function of time after the break should be based on a dynamic analysis of the subcompartment pressure response to pipe ruptures.
   b. The validity of the analysis should be supported by experimental data or a testing program should be proposed at the construction permit stage that will support this analysis.
   c. The effects of missiles that may be generated during the transient should be considered in the safety analysis.

6. The vent flow behavior through all flow paths within the nodalized compartment model should be based on a homogeneous mixture in thermal equilibrium, with the assumption of 100% water entrainment. In addition, the selected vent critical flow correlation should be conservative with respect to available experimental data. Currently acceptable vent critical flow correlations are the "frictionless Moody" with a multiplier of 0.6 for water-steam mixtures, and the thermal homogeneous equilibrium model for air-steam-water mixtures.

7. At the construction permit stage, a factor of 1.4 should be applied to the peak differential pressure calculated in a manner found acceptable to the CSB for the subcompartment. The calculated pressure multiplied by 1.4 should be considered the design pressure. At the operating license stage, the peak calculated differential pressure should not exceed the design pressure. It is expected that the peak calculated differential pressure will not be substantially different from that of the construction permit stage. However, improvements in the analytical models or changes in the as-built subcompartment may affect the available margin.

III. REVIEW PROCEDURES

The procedures described below are followed for the subcompartment analysis review. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis or by adopting the results of previous reviews of plants with essentially the same subcompartment and high pressure piping design.

The CSB reviews the initial conditions selected for determining the mass and energy release rate to the subcompartments. These values are compared to the spectrum of allowable operating conditions for the plant. The CBS will ascertain the adequacy of the assumed conditions based on this review.

The CSB confirms with the MEB the validity of the applicant's analysis of subcompartments containing high energy lines and postulated pipe break locations, using elevation and plan drawings of the containment showing the routing of lines containing high energy fluids. The CSB determines that an appropriate reference case for subcompartment analysis has been identified. In the event a pipe break other than a double-ended pipe rupture is postulated by the applicant, the MEB will evaluate the applicant's justification for assuming a limited displacement pipe break.

The CSB may perform confirmatory analyses of the blowdown mass and energy profiles within a subcompartment. The analysis is done using the RELAP3 computer program (See Reference 21 for a description of this code). The purpose of the analysis is to confirm the predictions of the mass and energy release rates appearing in the safety analysis report, and to confirm that an appropriate break location has been considered in this analysis. The use of RELAP3 will continue until the RELAP4 computer code has been approved by the staff as an acceptable blowdown code. At that time, the CSB will replace RELAP3 with RELAP4 for all subsequent analyses.

The CSB determines the adequacy of the information in the safety analysis report regarding subcompartment volumes, vent areas, and vent resistances. If a subcompartment must rely on doors, blowout panels, or equivalent devices to increase vent areas, the CSB reviews the analyses and testing programs that substantiate their use.

The CSB reviews the nodalization of each subcompartment to determine the adequacy of the calculational model. As necessary, CSB performs iterative nodalization studies for subcompartments to confirm that sufficient nodes have been included in the model.

The CSB compares the initial subcompartment air pressure, temperature, and humidity conditions to the criteria of II, above, to assure that conservative conditions were selected.

The CSB reviews the bases, correlations, and computer codes used to predict subsonic and sonic vent flow behavior and the capability of the code to model compressible and uncompressible flow. The bases should include comparisons of the correlations to both experimental data and recognized alternate correlations that have been accepted by the staff.

Using the nodalization of each subcompartment as specified in the safety analysis report, the CSB performs analyses using one of several available computer programs to determine the adequacy of the calculated peak differential pressure. The computer program used will depend upon the subcompartment under review as well as the flow regime. At the present time, the two programs used by the CSB are $RELAP_3$ (Ref. 21) and CONTEMPT-LT (Refs. 7, 8, and 9). A multi-volume computer code is currently under development.

At the construction permit stage, the CSB will ascertain that the subcompartment design pressures include appropriate margins above the calculated values, as given in II, above.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this section are presented in Standard Review Plan 6.2.1.

V. REFERENCES

The references for this plan are those listed in Standard Review Plan 6.2.1, together with the following:

1a. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

2a. Standard Review Plan 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," and attached Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment."

ATTACHMENT 2

MECHANICAL ENGINEERING BRANCH

REQUEST FOR ADDITIONAL INFORMATION

Recent analyses have shown that reactor pressure vessel supports may be subjected to previously underestimated lateral loads under the conditions that would exist if an instantaneous double ended break is postulated in the reactor vessel cold leg pipe at the vessel nozzle. It is therefore necessary to reassess the capability of the reactor coolant system supports to limit the calculated motion of the reactor vessel during a postulated cold leg break within bounds necessary to assure a high probability that the reactor could be brought safely to a cold shutdown condition.

The following information is required for purposes of making the necessary reassessment of the reactor vessel supports:

1. Provide engineering drawings of the reactor support system sufficient to show the geometry of all principle elements and materials of construction.
2. Specify the detail design loads used in the original design analyses of the reactor supports giving magnitude, direction of application and the basis for each load. Also provide the calculated maximum stress in each principle element of the support system and the corresponding allowable stresses.
3. Provide the information requested in 2 above for the RV supports considering a postulated break at the cold leg nozzle. Include a summary of the analytical methods employed and specifically state the effects of short term pressure differentials across the core barrel in combination

with all external loadings calculated to result from the required postulate. This analysis should consider:

(a) limited displacement break areas where applicable

(b) consideration of fluid structure interaction

(c) use of actual time dependent forcing function

(d) reactor support stiffness.

4. If the results of the analyses required by 3 above indicates loads leading to inelastic action in the reactor supports or displacements exceeding previous design limits provide an evaluation of the following:

(a) Yield behavior (effects of possible strain energy buildup) of the material used in the reactor support design and the effect on the loads transmitted to the reactor coolant system and the backup structures to which the reactor coolant system supports are attached.

(b) The adequacy of the reactor coolant system piping, control rod drives, steam generator and pump supports, structures surrounding the reactor coolant system, reactor internals and ECCS piping to assure that the reactor can be safely brought to cold shutdown.