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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 2 [illegible] 19[illegible]

MEMORANDUM FOR: S. Miner, Project Manager, Light Water Reactors Branch No. 3, DPM

FROM: G. Lainas, Chief, Containment Systems Branch, DSS

THRU: R. Tedesco, Assistant Director for Plant Systems, DSS

SUBJECT: REVIEW OF GE TOPICAL REPORT, NEDM-20988, "CAORSO RELIEF VALVE LOADS TESTS - TEST PLAN" (TAC-2285)

As requested, the Containment Systems Branch has reviewed the GE topical report, NEDM-20988 entitled "Caorso Relief Valve Loads Tests - Test Plan," and has prepared the enclosed comments including the following significant areas of concern.

1. Effects of leaking SRV on quencher loads has been identified as an area of concern. The Caorso test plan, however, has not addressed this concern. It is our position that tests on leaking SRV should be conducted and that a test matrix addressing this concern should be provided for our review.

2. The test matrix shows that the parameters of interest will be repeated only once. We believe that this is not sufficient to demonstrate the repeatability of test data. We, therefore, suggest that additional tests be conducted to demonstrate repeatability.

It should be noted that our review has excluded consideration of fluid/structure interaction effects. Comments concerning this area should be obtained from SEB.

Gus C. Lainas, Chief
Containment Systems Branch
Division of Systems Safety

Enclosure:
As Stated

cc: See Page 2

Contact:
T. Su, CSB
492-7711

D-65

cc: S. Hanauer
F. Schroeder
R. Tedesco
J. Glynn
C. Long
O. Parr
G. Lainas
J. Knight
I. Sihweil
D. Vassallo
R. Tedesco
J. Kudrick
J. Shapaker
R. Boyd (w/o encl.)
W. McDonald (w/o encl.)
T. Su

Request for Additional Information
General Electric Company
Topical Report: NEDM-20988, Revision 2
Test Plan - December, 1976
Reviewed by: T. M. Su

1. Effects of a leaking SRV on quencher loads has been identified as an area of concern. The Caorso test plan, however, does not address this concern. It is our position that tests on a leaking SRV should be conducted. Therefore, a test plan considering a leaking SRV should be provided for our review.

2. The Caorso test matrix indicates that most of the parameters of interest will be repeated only once. Tests performed either for ramshead (Quad City and Monticello) or for quencher (NEDE-11314-08) exhibited a great degree of data scatter. Therefore, we believe that the current Caorso test matrix is insufficient to determine the repeatability of the test data. We recommend additional tests be conducted for first actuation of a SRV and subsequent actuations of the same SRV to demonstrate repeatability. The number of tests should result in test data with statistical significance.

3. Based on our evaluation of previous SRV test data, we find that the SRV discharge time and duration between first actuation and subsequent actuation influence the quencher load due to subsequent actuation. Therefore, provide the value and the basis for the selection of the times for the Caorso tests.

4. Page 4-6 states that a complete understanding of the subsequent actuation effect requires data on pool temperature in the vicinity of the quencher, pipe temperature and pressure following valve closure, flow rate of air through the vacuum breaker and dynamics of back flow of water. We agree that the air temperature history inside the pipe could be important. However, insufficient information has been given in the test plan regarding the measurement of air temperature in the pipe. Clarify what measurements or calculations will be made to monitor this temperature.

5. The sensor failure rate was found to be quite high in the Monticello Plant test program. Sensors of the same manufacturer model used in the Monticello will also be used in the Caorso. In light of this experience, we believe that redundant instrumentation is needed in critical areas. For instance, redundant sensors should be provided in the following locations:

   a. The vicinity of quencher A and the place by which combined loads from multiple SRV's actuation will be determined.

   b. SRV line between elevation 51.612 and 45.770. In addition, level probes should be added between L1 and L12.

6. Submerged structure loads have been identified as a primary design load for the Mark II containment. We believe that the analytical program indicated in the Mark II owner group meeting which was held on February 16 and 17, 1977, is insufficient to support the design loads for submerged structure without experimental data. Therefore, we recommend that additional pressure sensors should be installed on support columns and downcomers to measure the drag load during SRV operation.

7. Provide the locations for pressure sensors Nos. 19, 23, 35, 36 and 37.

T. M. Su
904 21

GENERAL ELECTRIC

NUCLEAR ENERGY

PROJECTS DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125
MC 682, (408) 925-55040

Bunch - action
pls get in PDR
and to review

December 18, 1978

"BWR-SRV
Generic"

Dr. Harold R. Denton
Director, Nuclear Regulations
U. S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, MD

Dear Dr. Denton:

This is to provide General Electric's comments on the status of the five BWR-related items in the NRC's list of "unresolved safety issues." I believe these comments show that progress is being made on these issues, and they are not of safety significance to BWR plants.

Based on discussions between Walt D'Ardenne and Del Bunch, only five of the issues involve the BWR.

We are pleased to provide you these status reports.

Yours truly,

G. G. Sherwood

G. G. Sherwood, Manager
Safety & Licensing Operation

GGS:bp/1097

Attachment

cc: L. S. Gifford
W. H. D'Ardenne
D. F. Bunch-NRC

~~7901110353~~ 7pp.

D-107

PROPOSED "UNRESOLVED SAFETY ISSUES"

	ISSUES	FOR
1.	Waterhammer (A-1)	PWR
2.	Asymmetric Blowdown Loads on the Reactor Coolant System (A-2)	PWR
3.	Pressurized Water Reactor Steam Generator Tube Integrity (A-3, A-4, A-5)	PWR
4.	BWR Mark I and Mark II Pressure Suppression Containments (A-6, A-7, A-8, A-39)	BWR
5.	Anticipated Transients Without Scram (A-9)	PWR, BWR
6.	BWR Nozzle Cracking (A-10)	BWR
7.	Reactor Vessel Fracture Toughness (A-11)	PWR
8.	Qualification of Class 1E Safety-Related Electrical Equipment (A-24)	PWR, BWR
9.	Reactor Vessel Pressure Transient Protection (A-26)	PWR
10.	Residual Heat Removal Requirements (A-31)	PWR
11.	Seismic Design Criteria (A-40)	NRC CRITERIA
12.	Pipe Cracks in Boiling Water Reactors (A-42)	BWR
13.	Emergency Sump Reliability (A-43)	PWR
14.	Station Blackout (A-44)	NRC CRITERIA

ISSUES

PWR	9
BWR	5
NRC	2

PBS:at/23M1

4. Title: BWR Mark I and II Pressure Suppression Containments (A-6, A-7, A-8, A-39)

Issue:

As a result of an ongoing GE testing program, new hydrodynamic containment loads associated with a postulated loss-of-coolant accident and the anticipated discharge of safety relief valves were identified which had not been explicitly included in the original design of Mark I and II containments.

BWR Status:

GE and the Mark I and Mark II owners groups are in the final process of defining dynamic forcing functions for Mark I and II containments. Each utility is in the process of performing appropriate evaluations to determine the response of their containments to these loads. A substantial amount of interim/preliminary analyses has already been completed and are being reviewed by the NRC. The Mark I Load Definition Report will be submitted to the NRC starting in December 1978 with the final sections provided in March 1979. The Mark II Dynamic Forcing Functions Information Report was issued September 1975 and has been periodically updated with Revision 3 having been issued in June 1978. The NRC's final Zimmer Safety Evaluation Report which includes typical lead plant Mark II assessments is expected to be completed January 1979.

Basis for Continued Operation and Licensing

The acceptability of Mark I continued operation has been evaluated based on the Mark I Short Term Program and reported in the NRC's Safety Evaluation Report issued December 1977.

The acceptability of Mark II operation has been demonstrated through licensing evaluations leading to operating licenses on Zimmer.

PBS:sj/23H4

5. Title: Anticipated Transients Without Scram, ATWS (A-9)

Issue:

The NRC has modeled LWRs using extremely conservative probability input data. The results suggest that design changes ranging from minor to significant are needed to meet NRC objectives.

BWR Status:

GE has modeled the BWR using representative probability input data. The results conclude that the failure to scram is extremely unlikely because of the redundancy in the BWR control rod system. In summary, there is no need to install additional shutdown systems, and above all, no justification for making ATWS a design basis event.

At the request of the NRC, GE has provided descriptions and costs for Reactor Pump Trip, Alternate Rod Insertion, and modified Standby Liquid Control Systems which can be utilized to mitigate the postulated ATWS event even though GE believes that the event is so unlikely that no mitigation is necessary.

Basis for Continued Operation and Licensing

The probability of the event is so small that there is no safety issue and ATWS should be treated as a class 9 event. Even based on the conservative NRC modelling, continued operation is an acceptable risk because of the low probability of an anticipated transient without scram between.

PBS:sj/23H5

6. Title: BWR Nozzle Cracking (A-10)

Issue:

Cracks have been reported in feedwater nozzles and in control rod hydraulic return line nozzles in operating BWRs. The cracks have been initiated by high cycle thermal fatigue.

BWR Status:

Improved nozzle designs, inspection techniques, and operating procedures have been developed to avoid the nozzle cracking problem or to detect cracks before they become a safety concern. All BWR plants are committed to incorporate design modifications which will avoid the nozzle cracking problem, to perform verification tests to assure the adequacy of the modifications, and to instrument the operational nozzles employing the design changes to confirm the adequacy of the modifications.

Basis for Continued Operation and Licensing

Operating condition envelopes imposed on operating plants and improved nozzle designs for plants under construction provides adequate safety margins for the plant life based on evaluations performed by the NRC and GE.

PBS:sj/23H6

8. Title: Qualification of Class 1E Safety-Related Equipment (A-24)

Issue:

It is the NRC's position that construction permit applicants for which a Safety Evaluation Report was issued after July 1, 1974 should qualify all Class 1 electrical equipment to the requirements established in IEEE 323-1974, "IEEE Standard for Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations".

BWR Status:

GE is developing a Licensing Topical Report (LTR) which sets forth criteria by which Class 1E Equipment is qualified per the guidance of IEEE 323-1974 and the GE interpretation of Regulatory Guide 1.89.

The Licensing Topical Report is being developed to satisfy an open issue on the GESSAR docket and will be directly applicable to GESSAR Reference plants. Other projects committed to IEEE 323-1974 will reference the Licensing Topical Report.

For Projects which did not contractually commit NSSS Class 1E equipment to IEEE 323-1974 but are or will be required to comply with that standard for licensing, a plant specific retrofit program is planned.

Basis for Continued Operation and Licensing

Present equipment qualification programs are adequate; and there is sufficient basis to continue with plant licensing and to allow plant operation pending ultimate evaluation of programs and results.

PBS:sj/23H8

12. Title: Pipe Cracks in BWRs (A-42)

Issue:

Intergranular stress corrosion cracking has occurred in BWR Type 304 stainless steel recirculation system piping weld heat affected zones.

BWR Status:

Both GE and NRC evaluations have concluded that Intergranular Stress Corrosion Cracks (IGSCC) at weld heat affected zones of the piping are not a safety concern since (1) IGSCC can be detected with current in-service inspection procedures which are highly effective; and (2) detectable leaks always precede major pipe cracks. All plants perform scheduled inspections and are equipped with continuous leak detection systems to assure plant safety.

Numerous measures to ameliorate IGSCC, including the use of alternate alloys, are currently being implemented on both operating plants and plants under construction to improve availability.

Basis for Continued Operation and Licensing

Intergranular Stress Corrosion Cracking is a BWR plant availability concern but is not a safety issue since inservice inspection programs and leak detection systems preclude major cracks.

PBS:sj/23H12

T. Su 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

File BWR-SRV Mark I

MAR 1 1979

Generic Task No. A-39

Docket Nos.: 50-220, 50-237, 50-245, 50-249, 50-254, 50-259, 50-260, 50-263, 50-265, 50-271, 50-277, 50-278, 50-293, 50-296, 50-298, 50-321, 50-324, 50-325, 50-331, 50-333, 50-341, 50-354, 50-355, and 50-366

LICENSEES: Members of Mark I Owners Group (excepting Jersey Central Power & Light Company - Oyster Creek)

SUBJECT: SUMMARY OF MEETING HELD ON FEBRUARY 13, 1979 WITH REPRESENTATIVES OF THE MARK I OWNERS GROUP PROGRAM COMMITTEE

On February 13, 1979, a meeting was held in Bethesda, Maryland, with representatives of the Program Committee of the Mark I Owners Group. The purpose of the meeting was to discuss the generic Mark I program relating to the safety/relief valve loads and its application to individual plants. It is noted, however, that the discussion will not be applied for Oyster Creek since they are not using the quencher device developed under this particular program.

An attendance list and a copy of the meeting handouts are provided as enclosures 1 and 2, respectively.

The following summarizes the significant points of the meeting:

1. Representatives of General Electric described the SRV program. It consists of several subprograms, including a 1/4 scale test program, scaling analysis, full scale testing and an analytical model development effort. Test results from the 1/4 scale testing and full scale testing (Monticello) were used to verify the analytical models. These include three key analytical models; namely, SRV discharge line reflood model, SRV discharge line clearing model and torus shell load model. The three models are closely related; the calculational results from one model serves as input information for another model. GE is responsible for the model development and provides guidance for AE's and utilities in their application of these models. We expressed our concern about the level of guidance provided. We believe that the guidance as outlined in the Mark I Containment Program Load Definition Report is too general. We will require that more specific guidance be developed and provided for our review and evaluation.

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2. Multipliers and other factors are used in various models. We questioned the justification regarding these constants. GE indicated that the justification for some of the factors and multipliers cannot be clearly identified. However, they will attempt to provide the justification in the next meeting with the staff.

3. There was also some discussion on the assumption for delay time, which is defined as the time required for the SRV discharge line to depressurize following SRV closure and as a result allow the reflood transient to begin. The reflood model assumes that the delay time is proportional to the air volume stored in the SRV line. We requested that GE provide its justification for this assumption.

4. We expressed our concerns related to the torus shell load model. These are the substantial discrepancies between the data reported in the Monticello test report and the data used to verify the model. GE indicated that clarification of this matter will be provided at the next meeting.

T. M. Su, Task Manager
Task Action Plan No. A-39
Containment Systems Branch
Division of Systems Safety

Enclosures:
As Stated

<u>Distribution</u>

w/enclosures
Docket File
NRR Reading
CSB Reading
R. Fraley, ACRS (16)
S. Hanauer
R. Tedesco
I&E (3)
NRC PDR
Local PDR
W. Butler
T. Ippolito (4)
G. Lainas
V. Noonan
D. Ziemann
J. Kudrick
C. Grimes
L. Ruth
T. Su

w/o enclosure 2
H. Denton
R. Mattson
R. Boyd
D. Ross
R. DeYoung
D. Vassallo
D. Skovholt

Enclosure 1

ATTENDANCE LIST

SUBJECT: NRC T-QUENCHER ROADMAP

DATE: February 13, 1979

LOCATION: Bethesda, Md.

NAME	REPRESENTATING/ORGANIZATION
T. Martin	NUTECH
C. Tung	NRC/BNL
C. Economos	BNL
S. Hucik	GE
C. I. Grimes	NRC/DOR
T. M. Su	NRC/CSB/DSS
L. C. Ruth	NRC/CSB/DSS
L. Steinert	GE
T. J. Mulford	GE

Enclosure 2

MARK I CONTAINMENT PROGRAM

NRC LDR REVIEW

S/RV ROADMAPS

FEBRUARY 13, 1979

SAH-1
2-13-79

LOAD DEFINITION REPORT

SCOPE

- INTEGRATES RESULTS FROM THE TESTING AND ANALYTICAL TASKS
- COMBINES METHODOLOGIES AND PLANT UNIQUE LOADS
- PROVIDES LOAD COMBINATION TIMING
- DOCUMENTS FINAL DESIGN BASIS LOADS USED BY OWNERS/AE'S WITH STRUCTURAL ACCEPTANCE CRITERIA, PROCEDURES FROM OTHER TASKS AND AE SUPPLIED LOADS FOR MARK I PLANT EVALUATION AND DESIGN OF MODIFICATIONS.
- REFERENCES MARK I CONTAINMENT LTP TASK REPORTS FOR COMPLETE JUSTIFICATION OF LOAD DEFINITIONS.

SAH-2
2-13-79

LOAD DEFINITION REPORT

CONTENT AND ORGANIZATION

- LOADS CONSIDERED IN LDR
  - LOCA RELATED LOADS
  - SAFETY RELIEF VALVE DISCHARGE LOADS
  - OTHER CONSIDERATIONS (SECONDARY LOADS)

- LDR SATISFIES LONG TERM PROGRAM REQUIREMENTS FOR EVALUATION OF ALL LOADS

SAH-3
2-13-79

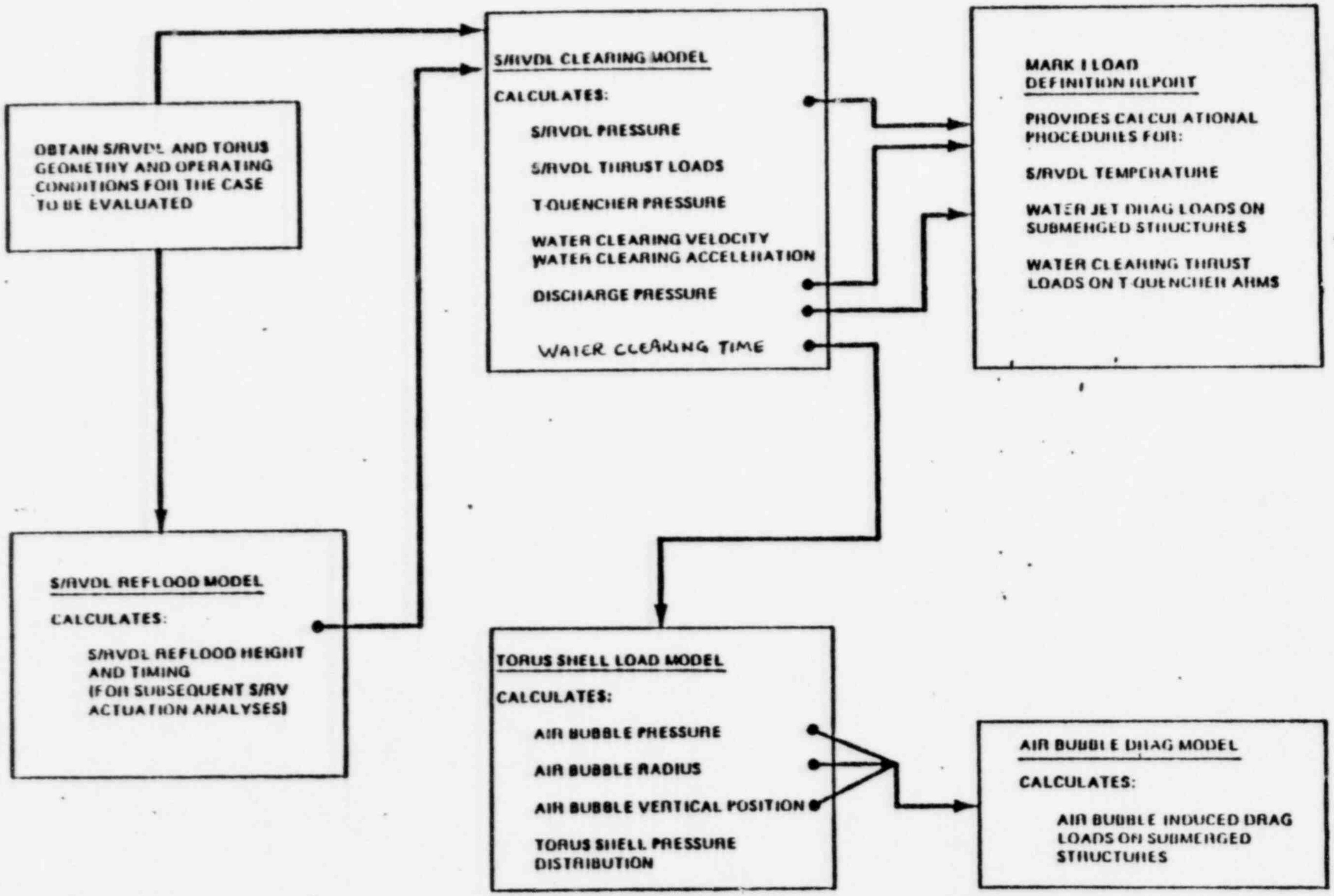


Figure 5.1-1. T-Quencher Load Definition Scheme

SAH-4
2-13-79

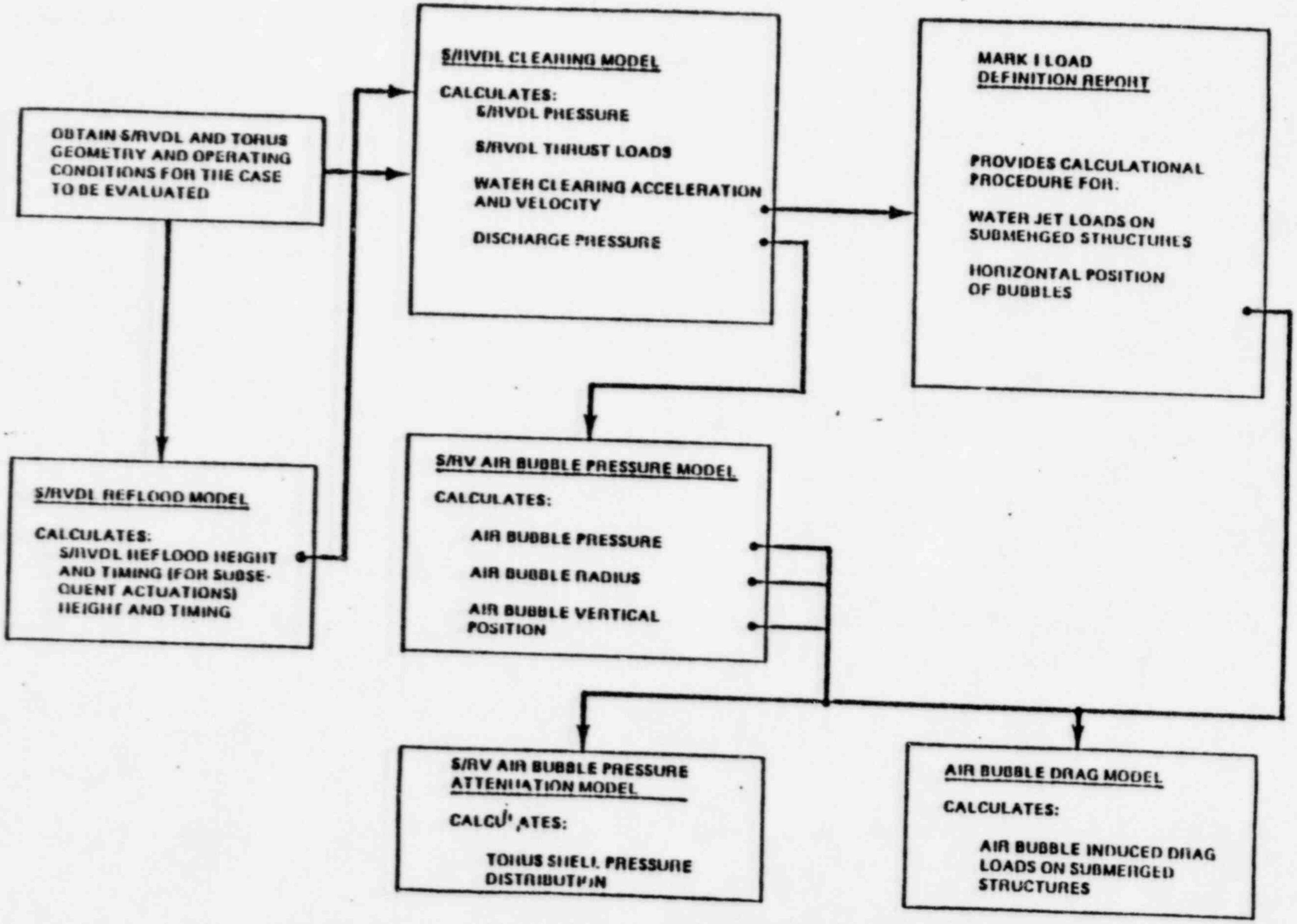


Figure 5.1-2. Ramshead Load Definition Scheme

SAH-5
2-13-79

S/RVDL CLEARING MODEL

- LDR SECTION 5.2.1

  S/RV DISCHARGE LINE CLEARING TRANSIENT LOADS

- MODEL INPUT → PLANT UNIQUE INFORMATION
- MODEL OUTPUT → LOAD DEFINITION

— S/RVDL PRESSURE

5.2.1 PIPE SEGMENT PRESSURE

5.2.7 S/RV DISCHARGE LINE MAX. TEMP.

— S/RVDL THRUST LOADS

5.2.1 THRUST LOADS ON S/RVDL PIPE SEGMENTS

— T-QUENCHER PRESSURE

5.2.1 T-QUENCHER PRESSURE LOADS.

— WATER CLEARING VELOCITY

5.2.4 WATER JET DRAG LOADS ON SUBMERGED STRUCTURES.

5.2.6 THRUST LOADS ON T-QUENCHER ARMS

SAH-6
2-13-79

S|..VDL CLEARING 1ODEL

— WATER CLEARING ACCELERATION

5.2.6 THRUST LOADS ON T-QUENCHER ARMS

— DISCHARGE PRESSURE

5.2.5 AIR BUBBLE DRAG LOADS ON SUBMERGED STRUCTURES.

5.2.2 TORUS SHELL PRESSURE

— WATER CLEARING TIME

5.2.5 AIR BUBBLE DRAG LOADS ON SUBMERGED STRUCTURES.

5.2.2 TORUS SHELL PRESSURE

- MODEL-DATA COMPARISONS

NEDE-23749-01-P

SAH-7
2-13-79

TORUS SHELL LOAD MODEL

- LDR SECTION 5.2.2

  TORUS SHELL PRESSURE

- MODEL INPUT

  — S/RVDL CLEARING MODEL
  MAXIMUM DISCHARGE PRESSURE
  WATER CLEARING TIME

  — S/RVDL AND T-QUENCHER GEOMETRY

  — PLANT UNIQUE GEOMETRY

  — PLANT UNIQUE INITIAL CONDITIONS FOR S/RVDL AND WETWELL

- MODEL OUTPUT

  — TORUS SHELL PRESSURE TRANSIENT

  — TORUS SHELL LONGITUDINAL PRESSURE DISTRIBUTION

  — TORUS SHELL RADIAL PRESSURE DISTRIBUTION

  — AIR BUBBLE PRESSURE, RADIUS, VERTICAL POSITION

  5.2.5 AIR BUBBLE DRAG LOADS ON SUBMERGED STRUCTURES

- MODEL DESCRIPTION – DATA COMPARISONS
  NEDE-21878-P

SAH-8
2-13-79

S/RVDL REFLOOD MODEL

- LDR SECTION 5.2.3

  S/RVDL REFLOOD TRANSIENT

- MODEL INPUT

  — S/RVDL AND T-QUENCHER GEOMETRY

  — PLANT UNIQUE INITIAL CONDITIONS

- MODEL OUTPUT

  — S/RVDL REFLOOD WATER TRANSIENT

  — DELAY TIME FOR REFLOOD

  INPUT TO S/RVDL CLEARING MODEL (5.2.1) FOR SUBSEQUENT S/RV ACTUATION ANALYSES

- MODEL DESCRIPTION – DATA COMPARISONS

  NEDE-23898-P

SAH-9
2-13-79

NRC APPROVAL - LDR S/RV METHODOLOGY

- S/RVDL CLEARING MODEL
  - NEDE-23749-01-P

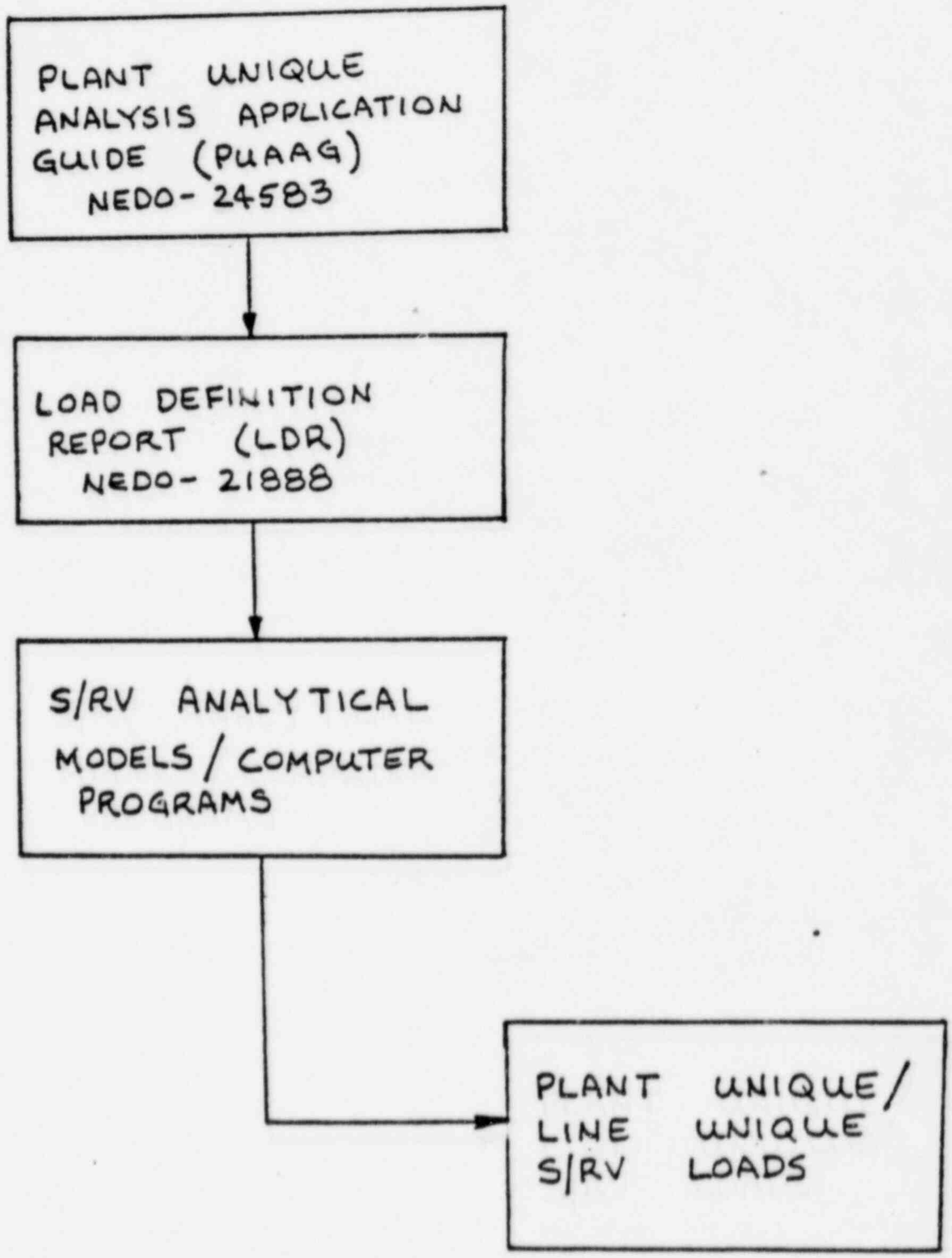
- S/RVDL REFLOOD MODEL
  - NEDE-23898-P

- TORUS SHELL PRESSURE MODEL
  - NEDE-21878-P

- S/RV SUBMERGED STRUCTURE MODELS
  - NEDO-21471
  - NEDE-21472-P
  - NEDE-24589-P

SAH-10
2-13-79

S/RV LOAD DEFINITION SCHEME



TJM
2-13-79

PUAAG - LDR S/RV LOAD EVALUATION CASES

LDR S/RV LOAD CASE		A1, A3**, C3, A2			N/A						A1, A3**, C3, A2						A1					
EVENT COMBINATIONS		SRV	SRV + EQ		SBA IBA		SBA + EQ IBA + EQ				SBA + SRV IBA + SRV		SBA + SRV + EQ IBA + SRV + EQ				DBA + SRV		DBA + EQ + SRV			
						CO, CH			CO, CH			CO, CH			CO, CH		PS	CO, CH	PS		CO, CH	
TYPE OF EARTHQUAKE			O	S			O	S	O	S			O	S	O	S			O	S	O	S
COMBINATION NUMBER		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21
LOADS Normal (2)	N	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
Earthquake	EQ		X	X			X	X	X	X			X	X	X	X			X	X	X	X
SRV Discharge	SRV	X	X	X							X	X	X	X	X	X	X	X	X	X	X	X
LOCA Thermal	T_A				X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
LOCA Reactions	R_A				X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
LOCA Quasi-Static Pressure	P_A				X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
LOCA Pool Swell	P_{PS}																X		X	X		
LOCA Condensation Oscillation	P_{CO}					X			X	X		X			X	X		X			X	X
LOCA Chugging	P_{CH}					X			X	X		X			X	X		X			X	X

** One S/RV will be considered leaky if plant is equipped with Ramshead

TJM
2-13-79

S/RV REPORTS SUBMITTED TO NRC

- S/RVDL CLEARING MODEL
  - NEDE-23749-01-P
- S/RVDL REFLOOD MODEL
  - NEDE-23898-P
- TORUS SHELL PRESSURE MODEL
  - NEDE-21878-P
- S/RV SUBMERGED STRUCTURE MODELS
  - NEDO-21471
  - NEDE-21472-P
  - NEDE-24589-P
- S/RV TEST REPORTS
  - MONTICELLO T-QUENCHER TEST REPORT -NEDE-21864-P
  - 1/4 SCALE T-QUENCHER TEST REPORT -NEDE-24549-P

F-2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 1 6 1979

Generic Task No. A-39

MEMORANDUM FOR: S. H. Hanauer, Director, Unresolved Safety Issues Program, NRR
R. P. Denise, Acting Assistant Director for Reactor Safety, DSS

FROM: T. M. Su, A-39 Task Manager, Containment Systems Branch, DSS

SUBJECT: SUMMARY OF MEETING HELD ON APRIL 4, 1979 WITH REPRESENTATIVES OF THE GENERAL ELECTRIC COMPANY TO DISCUSS SRV METHODOLOGY

On April 4, 1979, a meeting was held in Bethesda, Maryland with representatives of the General Electric Company. The purpose of the meeting was to discuss the methodology for predicting bubble phasing during multiple valve discharges for all Mark III containments where the GE designed cross quencher device is used in the safety/relief valve discharge line.

An attendance list and a copy of the meeting handouts are enclosed.

Background

In April 1978, the General Electric Company submitted an Interim Containment Loads Report, Mark III Containment (22A4365). Attachment M to the report provides an outline of the methodology for determining multiple safety/relief valves bubble-phasing. Since then a series of discussions had been held between GE, the staff and their consultants. Following these discussions, GE had gathered all staff concerns and provided justifications for each concern. GE, therefore, requested the meeting to discuss these justifications.

Summary

1. The bubble frequency distribution curve was generated on the basis of 132 data points obtained from tests at reactor pressures ranging from 150 to 1000 psia. Since the wide range of initial testing condition will affect the bubble frequency distribution, we requested that GE generate a bubble frequency curve based on initial testing pressure close to rated reactor pressure. In response to this request, GE presented the results of their study, which was based on selected reactor pressure and initial pool temperature. This selection criterion reduced the number of data points from the original 132 to 38. Analyses based on these selected data points resulted in a standard deviation of 1.7 Hz instead of 2.3 Hz as the original curve indicated. The mean also changes from 8.1 Hz to 8.9 Hz. Based on the

~~7908170270~~ 30pp.

results of this study, GE proposed a standard deviation of 1.7 Hz and a mean of 8.1 Hz as the design values. Note that the selected mean is based on all data because it results in a higher confidence level.

2. The results of the study also confirmed that line air volume is the most dominant parameter for determining bubble frequency; the other parameters such as SRV opening time, line air temperature and submergence have no statistically significant effect on bubble frequency.

3. GE will include the Caorso test results in their final analysis for predictions of bubble phasing during multiple valve actuations. The preliminary analysis indicates that the current methodology predicts conservative results when compared with the Caorso data.

4. The staff and their consultants concluded that the general approach for predicting multiple valve bubble phasing is valid. We will require, however, that GE include the following in the final analysis:

   a. Effect of pool temperature on bubble frequency;

   b. Sensitivity study of standard deviation of bubble frequency distribution and its effect on SRV loads;

   c. Effect of pool boundaries on bubble frequency; and

   d. Structural and equipment response for determining the design case.

T. M. Su, A-39 Task Manager
Containment Systems Branch
Division of Systems Safety

Enclosures:
As Stated

Distribution:

Central File	R. Martin
NRR Reading	D. Ross
CSB Reading	I&E (3)
H. Denton	NRC PDR
R. Mattson	Local PDR
F. Schroeder	S. Varga
R. Fraley, ACRS (16)	W. Butler
R. DeYoung	J. Kudrick
D. Vassallo	T. Su
D. Skovholt	
R. Denise	

Mark III SRV Meeting
April 4, 1979

Name	Organization
T. J. Su	NRC/CSB/DSS
P. Huber	MIT/BNL
Forrest Hatch	GE
F. Reuter	GE
P. Stancavage	GE
I. Uppal	GE
L. Sobon	GE
R. Patel	Bechtel
R. A. Hill	GE
P. Moskal	Bechtel
R. Beck	Bechtel
C. Tung	BNL
J. A. Kudrick	NRC/CSB/DSS

MULTIPLE QUENCHER METHODS

SUMMARY

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PPS - 1
4/4/79

VALID DATA BASE

- FREQUENCY DISTRIBUTION FROM IN - PLANT DATA
  - FULL SCALE IN TWO BWRS
  - WHOLE RANGE OF EXPECTED CONDITIONS
- DISTRIBUTION CONFIRMED AT TYPICAL CONDITIONS
  - FULL REACTOR PRESSURE
  - MODERATE POOL TEMPERATURE
  - FIRST ACTUATION
- CAORSO TEST VALIDATES DISTRIBUTION
  - FREQUENCY IS RANDOM
  - STRONG SIGNALS 5 TO 10 Hz

FLUID - STRUCTURE INTERACTION

- GLOBAL EFFECTS SMALL
  - TORUS MODEL SHOWS NO FSI IF MINOR DIA/THICKNESS LESS THAN 600
  - LICENSEE PLANTS HAVE D/T < 600
  - MARK II/III PLANTS HAVE D/T < 300
- LOCAL EFFECTS SMALL
  - TRANSDUCERS ON CONCRETE AND STEEL SHOW SAME FREQUENCIES
  - NATURAL FREQUENCIES OF LOCAL STRUCTURES MUCH HIGHER THAN BUBBLE FREQUENCIES

PPS - 3
4/4/79

CAORSO DATA

- PRESSURE AMPLITUDES BOUNDED BY PREDICTION

- MEAN VALUES	PREDICTED	MEASURED
FIRST ACTUATION	+9.1/-6.5	+4.3/-2.8
SUBSEQUENT ACTUATION	+15.9/-9.2	+7.2/-4.5

- DESIGN VALVES	PREDICTED (90/90)	MEASURED (MAX)
FIRST ACTUATION	+12.8/-8.1	+5.0/-4.3
SUBSEQUENT ACTUATION	+29.6/-11.6	+8.0/-5.7

- TIME AND DISTANCE ATTENUATION MORE RAPID THAN PREDICTED

- BUBBLE FREQUENCY IS RANDOM
  - CYCLE TO CYCLE
  - TEST TO TEST
    - INTEGRAL
    - SPECIAL DENSITY
  - VALVE TO VALVE

BUBBLE PHASING

- SMALL INTERACTION EXPECTED
  - BUBBLES SEPARATED BY 5 DIA
  - INFLUENCE ONLY 10%
  - NO EFFECT ON INITIAL CYCLES
- SMALL INTERACTION CONFIRMED BY TEST
  - MULTIPLE VALVE PRESSURE BOUNDED BY SINGLE VALVE PRESSURE
    - MONTICELLO
    - CAORSO
  - MULTIPLE VALVE WAVEFORMS MIXED

CONCLUSIONS

- DATA BASE IS VALID
  - FULL RANGE OF CONDITIONS
  - CONFIRMED AT TYPICAL OPERATING STATE
  - VERIFIED BY CAORSO TESTS
- FLUID STRUCTURE INTERACTION IS SMALL
  - GLOBAL EFFECT UNIMPORTANT D/T < 600
  - LOCAL EFFECTS NOT OBSERVED
- CAORSO TESTS CONFIRM METHODS
  - PRESSURE AMPLITUDES
  - TIME AND DISTANCE ATTENUATION
  - RANDOM FREQUENCIES

MULTIPLE

QUENCHER

METHODS

NRC QUESTIONS

I. S. UPPAL
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APRIL 4, 1979

QUESTION 1: DISCUSS THE STOCHASTIC NATURE OF BUBBLE FREQUENCY

RESPONSE:

BUBBLE FREQUENCY IS RANDOM DUE TO VARIATIONS IN

- o INITIAL CONDITIONS
  - LINE TEMPERATURE
  - WATER LEVEL
  - STEAM CONTENT
- o DYNAMIC PROCESS
  - TAYLOR INSTABILITY
  - BUBBLE FORMATION

EXPERIMENTAL EVIDENCE SHOWS FREQUENCY VARIATION EVEN WITH SIMILAR INITIAL CONDITIONS

SELECT DATA BASE

QUESTION 2: THE DATA WERE OBTAINED AT REACTOR PRESSURES RANGING FROM 150 TO 1000 PSIA. HOW DID THIS AFFECT THE QBF OBTAINED? A SEPARATE QBF INCLUDING THOSE DATA OBTAINED AT FULL REACTOR PRESSURE SHOULD BE GENERATED.

RESPONSE:

A SUBSET OF THE DATA BASE WAS SELECTED FROM THE IN-PLANT TESTS WHICH MOST CLOSELY REPRESENTS EXPECTED CONDITIONS FOR AN ALL VALVE ACTUATION EVENT. THE SELECTION CRITERIA ARE:

- FIRST ACTUATION, SINGLE VALVE
- POOL TEMPERATURE BELOW 110°F
- REACTOR PRESSURE ABOVE 950 PSIG

	NUMBER OF TESTS	MEAN (Hz)	STANDARD DEVIATION (Hz)	
ALL DATA MEETING CRITERIA	38	8.9	1.7	
ALL DATA	132	8.1	2.3	USE TWO SIGNIFICANT FIGURES
DESIGN USE	N/A	8.1	1.7	

THERE IS REASONABLE AGREEMENT BETWEEN SELECTED DATA AND ALL DATA.

- MEAN FREQUENCY OF DATA MEETING CRITERIA IS WITHIN TEN PERCENT OF MEAN FREQUENCY OF ALL DATA. MEAN FREQUENCY (8.1 Hz) USED IS BASED ON ALL DATA BECAUSE 132 TESTS GIVE HIGHER CONFIDENCE.

- STANDARD DEVIATION OF 1.7 Hz IS SELECTED FOR DESIGN USE
  - 1.7 Hz IS LESS THAN THE STANDARD DEVIATION BASED ON ALL DATA AND HENCE CONSERVATIVE
  - 1.7 Hz IS BASED ON TEST DATA WITH FULL REACTOR PRESSURE AND OTHER CONDITIONS THAT ARE TYPICAL OF BWR PLANTS

FREQUENCY DEPENDENCE

QUESTION 3: WAS THE QBF OBTAINED UNDER CONDITIONS OF CONSTANT AIR LINE VOLUME, VALVE OPENING TIMES, SRV LINE LENGTH AND HYDRAULIC RESISTANCE, PIPE TEMPERATURE ETC.? SEPARATE CREDIT IS TAKEN FOR THE POSSIBLE MITIGATING EFFECTS OF SOME OF THESE VARIABLES: IT IS, THEREFORE, IMPORTANT TO ESTABLISH THAT THEIR INFLUENCE NOT ALREADY IMPLICIT IN THE QBF.

RESPONSE:

THE FREQUENCY DISTRIBUTION WAS OBTAINED FROM TWO IN-PLANT TESTS WITH COMPARABLE CONDITIONS

PARAMETER	PLANT A	PLANT B	DATA MEETING CRITERIA
LINE AIR VOLUME (FT^3)	50	47	47-50
SUBMERGENCE (FT)	15	13	13-15
POOL TEMPERATURE (°F)	85-104	92-169	85-110
VALVE OPENING TIME (MSEC)	150-1000	220-1475	275-1475
REACTOR PRESSURE (PSI)	13-1066	120-1030	950-1066

VARIOUS PARAMETERS WERE INVESTIGATED VIA REGRESSION ANALYSIS FOR THEIR EFFECT ON FREQUENCY. PRESSURE RISE RATE, VALVE SETPOINT, VALVE OPENING TIME, AND BUBBLE FREQUENCY ARE INDEPENDENT VARIABLES.

RESULTS OF REGRESSION ANALYSIS

- o BUBBLE FREQUENCY IS NOT INFLUENCED BY VALVE OPENING TIME OR REACTOR PRESSURE
- o RESULTS SHOW NO TREND REGARDING EFFECT OF OTHER PARAMETERS ON BUBBLE FREQUENCY
- o LINE AIR TEMPERATURE WAS FAIRLY CONSTANT
- o LINE AIR VOLUME WAS (47-50 FT^3) CONSTANT
  - EFFECT OF VOLUME NOT IN DATA BASE
  - EFFECT OF VOLUME IS SIGNIFICANT
  - VOLUME IS INCLUDED SEPARATELY

CAORSO DATA

QUESTION 4: THE QBF SHOULD BE UPDATED ON THE BASIS OF THE CAORSO TEST AS SOON AS THESE ARE AVAILABLE.

RESPONSE:

- o CAORSO SVA DATA IS IN PRELIMINARY FORM. ANALYSIS SHOWS CAORSO MEASURED FREQUENCY IS RANDOM. CAORSO MEASURED FREQUENCY IS WITHIN RANGE OF PREDICTED FREQUENCIES.
- o TWO WAYS TO OBTAIN BUBBLE FREQUENCY
  - $F = \frac{\text{TOTAL OF BUBBLE OSCILLATION CYCLES}}{\text{TOTAL TIME OF CYCLES}}$
  - POWER SPECTRAL DENSITY PLOT (SEE FIGURES 6 & 7)
- o PSD USED TO DEVELOP CURRENT QBF
- o PSD PLOT SHOWS THAT MORE THAN ONE FREQUENCY HAS SIGNIFICANT ENERGY
- o PSD FREQUENCY SPREAD IS MUCH LARGER THAN TIME AVERAGED FREQUENCY (TABLE 4)
- o OVERALL CAORSO PRESSURE IS A FACTOR OF 2 BELOW MEAN PREDICTED

- IN TABLE 4, PREDICTED MEAN FREQUENCY

  $= 8.1 \times \sqrt[3]{\frac{50}{66.1}} = 7.4$ Hz

- MEASURED FREQUENCY IS ONE PREDOMINANT FREQUENCY PER TEST

TABLE 4

	MEASURED	PREDICTED
MEAN (Hz)	6.05	7.4
S.D. (Hz)	± .41	1.6
LOWER BOUND (Hz)	5.3	4.6
UPPER BOUND (Hz)	6.8	10.9

- TABLE 3 SHOWS THAT EACH CYCLE HAS ITS OWN FREQUENCY
- CAORSO MVA MODEL/DATA COMPARISON UNDERWAY. THIS COMPARISON WILL SHOW THAT SRVA IS CONSERVATIVE BY A LARGE MARGIN

FREQUENCY DATA BASE

QUESTION 5: HOW WELL IS THE PROBABILITY DISTRIBUTION KNOWN? WHAT IS THE DATA BASE?

RESPONSE:

THE FREQUENCY PROBABILITY IS BASED ON 132 IN-PLANT QUENCHER TESTS AT TWO LICENSEE FACILITIES. THESE TESTS PROVIDE THE DATA FOR:

- MEAN 8.1 HERTZ
- STANDARD DEVIATION 1.7 HERTZ
- UPPER BOUND 12 HERTZ
- LOWER BOUND 5 HERTZ

A CHI - SQUARE TEST SHOWS THAT THE NORMAL DISTRIBUTION IS APPROPRIATE AT 5% LEVEL OF SIGNIFICANCE.

QUESTION 6: THE POSSIBILITY THAT THE TEST DATA (ESPECIALLY THOSE RELATING TO BUBBLE FREQUENCY) WERE AFFECTED BY FLUID STRUCTURE INTERACTION SHOULD BE ADDRESSED.

RESPONSE:

GLOBAL FSI EFFECTS

- APPLIED SRV FORCING FUNCTION TO A COUPLED FLUID STRUCTURE MODEL OF MARK I TORUS
- MODEL SHOWED FOR MINOR TORUS DIAMETER TO SHELL THICKNESS (D/T) RATIO UP TO 600, FSI IS NEGLIGIBLE
- PLANTS A & B $\frac{D}{T} \sim 600$. CONCLUSION OF REFERENCED STUDY IS APPLICABLE
- HENCE GLOBAL FSI EFFECTS IN PLANTS A & B ARE NEGLIGIBLE

LOCAL FSI EFFECTS

- LOCAL STRUCTURAL INFLUENCE IS SHOWN BY TRANSDUCERS AT DIFFERENT LOCATIONS
- FIGURE 10 SHOWS THE LOCATIONS OF PRESSURE TRANSDUCERS DA 13, 14, & 16
- FIGURE 11 SHOWS THAT EACH HAS A BUBBLE FREQUENCY ~8 Hz
- THEREFORE, THIS IS A GENUINE BUBBLE FREQUENCY, NOT AFFECTED BY LOCAL FSI

FREQUENCY DATA FROM TYPICAL PLANTS

- ADDITIONALLY PLANTS A & B ARE TYPICAL OF MARK II AND III PLANTS. FSI EFFECTS (NEGLIGIBLE) PRESENT IN A & B WILL ALSO BE PRESENT IN MARK II & III. (SEE FIGURE 12)

FREQUENCY DEPENDENCE

QUESTION 7: TO WHAT EXTENT IS THE DISTRIBUTION PLANT - SPECIFIC OR SRV - SPECIFIC? DOES IT DEPEND ON LINE LENGTH? LINE TEMPERATURE? FIRST OR SUBSEQUENT ACTUATION?

RESPONSE:

o DESIGN DISTRIBUTION VARIES WITH LINE VOLUME ONLY

o DISTRIBUTION IS BROAD ENOUGH TO ACCOUNT FOR ALL OTHER VARIABLES

o MARK II AND III PLANTS HAVE SAME BASIC GEOMETRY AS FAR AS SRV LOADS ARE CONCERNED. THIER FREQUENCY IS NOT EXPECTED TO VARY DUE TO PLANT GEOMETRY

o FREQUENCY IS A FUNCTION OF LINE VOLUME

o LINE TEMPERATURE EFFECT IS IMPLICIT IN THE DATA BASE

o CAORSO SECOND ACTUATION FREQUENCY HIGHER THAN FIRST ACTUATION. EARLIER TEST SHOWED THAT FREQUENCY DOES NOT DEPEND ON FIRST OR SUBSEQUENT ACTUATION

QUESTION 8: WHAT ARE THE DATA BASES FOR THE PROBABILITY DISTRIBUTIONS OF VALVE SETPOINT AND VALVE OPENING TIME?

RESPONSE:

VALVE SETPOINT

- FOR TESTABLE INSTRUMENTATION, SD - 2 PSI APPLIED TO BOTH CROSBY & DIKKERS VALVES
- FOR NON-TESTABLE INSTRUMENTATION SD = 8 PSI IS USED BASED ON 24 SHOP TESTS
- FOR TARGET ROCK VALVES SD - 5.9 PSI BASED ON 77 SHOP TESTS. FIGURE 13 SHOWS THE DISTRIBUTION IS CLOSE TO NORMAL.

VALVE OPENING TIME

- FOR CROSBY VALVES SD = .0092 SEC BASED ON 408 TESTS
- FOR DIKKERS VALVES D = .0097 BASED ON 50 TESTS. THEREFORE, STANDARD DEVIATION FOR BOTH CROSBY AND DIKKERS IS SPECIFIED AS 0.009 SEC.
- FOR TARGET ROCK VALVES 187 DATA POINTS GAVE SD = .013 SECONDS

AIR VOLUME ON FREQUENCY

QUESTION 9: IT IS PROPOSED THAT THE QBF DISTRIBUTION BE SHIFTED TO ACCOUNT FOR SRV LINE VOLUMES THAT DIFFER FROM THE 50 FT^3 LINES USED TO OBTAIN THE DATA. THE PROPOSED ADJUSTMENT IS BASED ON A SIMPLISTIC AND POSSIBLY NON-CONSERVATIVE ANALYSIS WHICH NEGLECTS THE KNOWN DEPENDENCE OF BUBBLE PRESSURE ON AIR LINE VOLUME. A REASSESSMENT OF THIS ASSUMPTION IS REQUIRED.

RESPONSE:

THE RELATIONSHIP GOVERNING FREQUENCY AND AIR VOLUME IS, FROM RAYLEIGH'S EQUATION

$$\text{FREQUENCY} \propto \sqrt[-3]{\text{AIR VOLUME}}$$

THE ENTIRE RELATIONSHIP HAS BEEN EXPERIMENTALLY CONFIRMED IN 1/4-SCALE T-QUENCHER TESTS FOR LINE VOLUMES RANGING FROM 24 FT^3 TO 99 FT^3 (SEE FIGURE 14)

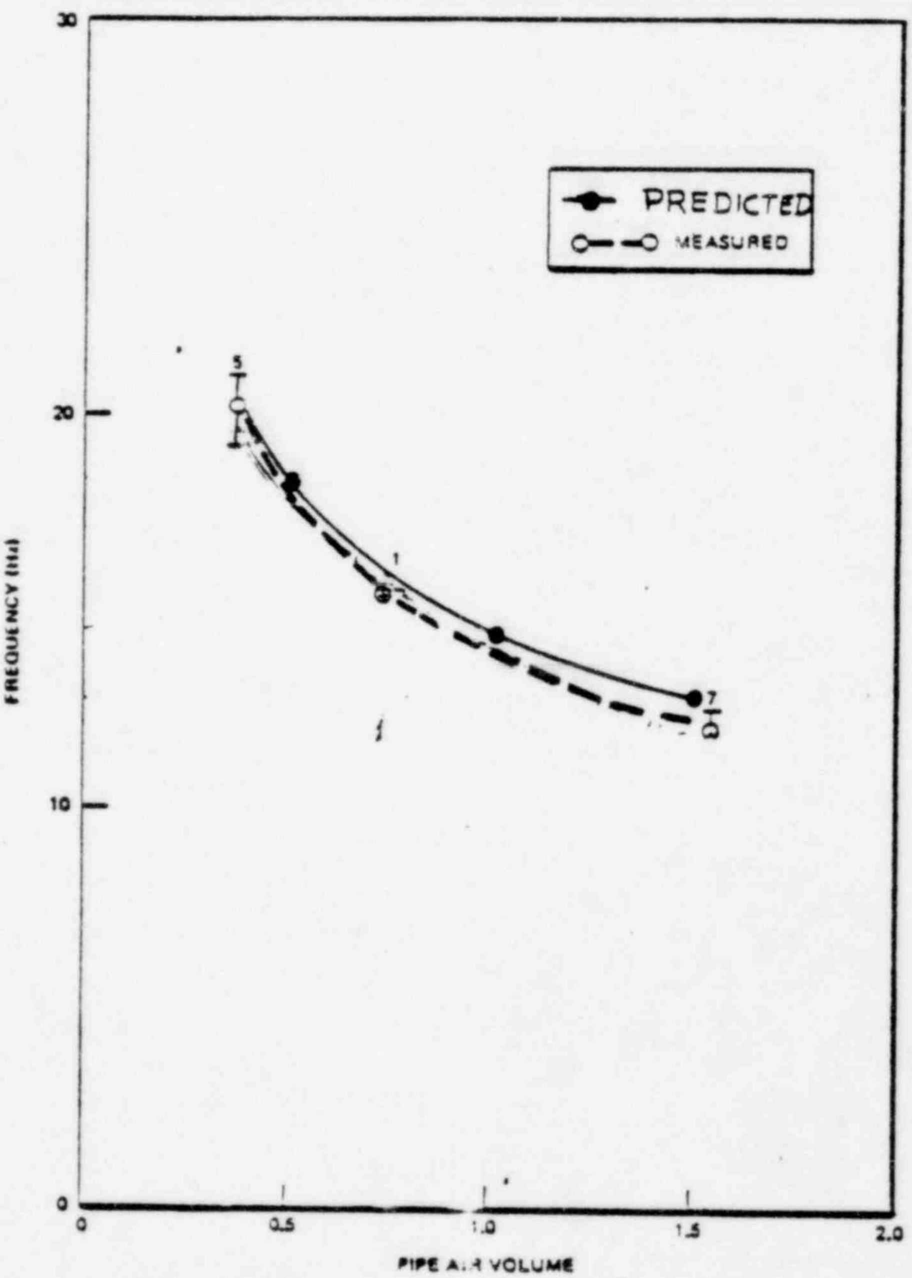

Figure 1? . Effect of Pipe Air Volume on Frequency for Initial Pipe and Wetwell Pressure of 3.7 psia

QUESTION 10: ONE CENTRAL ASSUMPTION OF THE PROPOSED METHODOLGY IS THAT BUBBLES OSCILLATING SIMULTANEOUSLY IN THE POOL DO NOT INTERACT IN ANY WAY THAT WOULD TEND TO INCREASE THE LOAD AMPLITUDES TO MODIFY THE PHASE DIFFERENCE BETWEEN THE BUBBLE OSCILLATIONS OR TO HARMONIZE THE OSCILLATION FREQUENCIES. THE BASIS FOR THIS ASSUMPTION IS ONE OF OUR MAIN PRESENT CONCERNS.

RESPONSE:

- THEORY PREDICTS LITTLE INTERACTION
  - QUENCHERS ARE 14 FEET APART
  - BUBBLE DIAMETERS ARE 2.7 FEET
  - MEAN SPACING IS 5 DIAMETERS
  - EFFECT (1/R) IS 10%
- TESTS SHOW NO INTERACTION
  - MONTICELLO TESTS SHOW SVA SHELL PRESSURES GENERALLY LESS THAN MVA SHELL PRESSURES
  - CAORSO 2 AND 3 VALVE MVA PRESSURES LESS THAN SVA PRESSURES
  - CAORSO 4 VALVE MVA PRESSURES ~20% HIGHER THAN SVA BUT WAVEFORM INDICATES NON-PHASED BUBBLES
  - CAORSO 8 VALVE MVA PRESSURES ~6% HIGHER THAN SVA AND WAVEFORM INDICATES NON-PHASED BUBBLES

- MULTIPLE QUENCHER METHOD ALLOWS APPROXIMATE PHASING
  - EXAMPLE IN ATTACHMENT M
  - 3 BUBBLES AT 0.123 SEC AND 9.3 Hz
  - 2 BUBBLES AT 0.128 SEC AND 8.6 Hz

Valve No.	IVOT (sec)	Valve No.	IVOT (sec)	Valve No.	IVOT (sec)
1	0.067	7	0.067	13	0.056
2	0.069	8	0.051	14	0.061
3	0.065	9	0.062	15	0.056
4	0.059	10	0.065	16	0.065
5	0.063	11	0.058	17	0.057
6	0.038	12	0.057	18	0.071
				19	0.069

Note that a mean value of 0.057 sec is included in the above numbers. Adding these values to the group T_i calculated in Step 3 and normalizing to have the first bubble arrive at zero time results in the following bubble arrival times:

Valve No.	Arrival Time (sec)	Valve No.	Arrival Time (sec)	Valve No.	Arrival Time (sec)
1	0.125	7	0.125	13	0.243
2	0.256	8	0.238	*14	0.127
3	0.123	9	0.120	15	0.243
4	0.247	10	0.0	16	0.124
5	0.122	11	0.246	17	0.245
6	0.225	12	0.116	*18	0.129
				19	0.256

M.3 BUBBLE FREQUENCIES

Bubble frequencies for individual quenchers are randomly selected from a random number generator code using the distribution shown in Figure M2-4. Typical random bubble frequency values for the 19 quenchers are:

Valve No.	Frequency (Hz)	Valve No.	Frequency (Hz)
1	6.56	11	7.22
2	9.77	12	5.39
3	9.15	13	5.68
4	5.01	14	*8.60
5	9.33	15	9.86
6	6.88	16	7.04
7	9.41	17	11.08
8	9.10	18	*8.68
9	7.92	19	8.52
10	11.14		

NOTE: For this example, all lines are considered as uniform in length and frequencies are randomly selected from one Quencher Bubble Frequency (QBF) distribution curve (Figure M2-4). In this example, mean = 8.23 Hz and σ = 1.80 Hz. With nonuniform line lengths, Subsection M3.2.1 is used to develop unique QBF distribution curves from which a frequency is randomly selected for each line.

EFFECT OF LINE ON BUBBLE ARRIVAL TIME

QUESTION 11: AT PRESENT NO CREDIT IS TAKEN FOR POSSIBLE CHANGES IN BUBBLE ARRIVAL TIME DUE TO DIFFERENCES IN SRV LINE LENGTH OR HYDRAULIC RESISTANCE. THESE FACTORS COULD, HOWEVER, TEND TO NEGATE THE FAVORABLE EFFECT OF DIFFERENT VALUE SETPOINTS. THEY SHOULD BE ADDRESSED.

RESPONSE:

- o LINE AIR VOLUME AFFECTS BUBBLE ARRIVAL TIME SOMEWHAT BY CHANGING THE AIR AND WATER CLEARING TIMES

- o FOR EXAMPLE, AN INCREASE IN AIR VOLUME FROM 57 FT^3 TO 88 FT^3 CAUSES A DELAY OF 56 MSEC IN AIR CLEARING TIME

- o IN INDIVIDUAL PLANTS, AIR VOLUME USUALLY LIES WITHIN 25%

- o RESULT OF INCLUDING LINE VOLUME FOR A TYPICAL PLANT (57 FT^3 88 FT^3)

  - THE AVERAGE FOURIER SPECTRA REMAINED ESSENTIALLY UNCHANGED (FIGURES 15-20)

FORCING FUNCTION SELECTION

QUESTION 12: DISCUSS IN GREATER DETAIL HOW A FREQUENCY DEPENDENT "BOUNDING" FORCING FUNCTION IS DEDUCED FROM THE COMPUTED 59 MONTE CARLO SIMULATIONS.

RESPONSE:

- o THE FORCING FUNCTION IS BOUNDING IN THE SENSE THAT 59 TRIALS GIVE 95% CONFIDENCE THAT THE PEAK BOUNDS 95% OF ALL EXPECTED RESULTS. FEWER TRIALS WILL GIVE LESS CONFIDENCE

- o SIGNIFICANT FREQUENCY RANGE IS DIVIDED INTO 3 FREQUENCY AND LARGEST SPECTRAL VALUE WITHIN EACH FREQUENCY INTERVAL IS SELECTED FOR DETERMINATION OF EQUIPMENT RESPONSE

- o ADDITIONAL CONFIDENCE IN THE BOUNDING CHARACTERISTIC OF THE FORCING FUNCTION IS PROVIDED BY:

  - HIGHEST BUBBLE PRESSURE GIVEN BY ANY DISCHARGE LINE IS USED FOR ALL DISCHARGE LINES. THE MAXIMUM DESIGN BUBBLE IS EXTREMELY CONSERVATIVE. THE PREDICTED BUBBLE PRESSURE FOR CAORSO SVA IS 15.1/-8.9 PSI AS COMPARED TO MEASURED MAXIMUM PEAK BUBBLE PRESSURE OF 5.0/-4.5 PSI.

  - THE DISTANCE ATTENUATION OF GESSAR/DFFR BOUNDS CAORSO DISTANCE ATTENUATION

- GESSAR/DFFR TIME ATTENUATION BOUNDS TIME ATTENUATION OBSERVED AT CAORSO (SEE FIGURE 21)

- o FIGURE 21 ALSO SHOWS THAT CAORSO DATA IS BOUNDED BY PREDICTIONS BY A LARGE MARGIN

- o FOR LINEAR SYSTEMS, HIGHEST INPUT FOR EACH FREQUENCY GIVES HIGHEST OUTPUT

CONFIRMATORY WORK

QUESTION 13: IT IS OUR OPINION THAT COMPARISONS BETWEEN PREDICTIONS BASED ON THIS METHODOLGY AND MVA IN-PLANT LOAD DATA ALREADY AVAILABLE AND TO BE OBTAINED FROM THE CAORSO TESTS IS AN ESSENTIAL PART OF THE REVIEW PROCESS. BOUNDING FORCING FUNCTION PREDICTIONS FOR DISCHARGE CONDITIONS CORRESPONDING PRECISELY TO THOSE ACTUALLY TESTED (TWO, THREE, FOUR AND EIGHT VALVE DISCHARGES AT CAORSO, ALL MULTIPLE VALVE TESTS AT BRUNSBUTTEL) SHOULD BE GENERATED AND COMPARED WITH THE IN-PLANT LOAD DATA. IT IS RECOGNIZED THAT THE IN-PLANT DATA CONSISTS OF DISCRETE PRESSURE MEASUREMENTS A BEST ESTIMATE OF THE INTEGRATED LOAD MUST NEVERTHELESS BE OBTAINED.

WE EMPHASIZE THE NEED FOR COMPARISONS BETWEEN TEST DATA AND PREDICTIONS BASED ON THE PROBALISTIC PROCEDURE AS APPLIED TO DISCHARGE CONDITIONS IDENTICAL TO THOSE TESTED.

RESPONSE:

- o A QUICK LOOK INDICATES THAT CAORSO MVA DATA IS BOUNDED BY PREDICTIONS BY A LARGE MARGIN
- o MODEL/DATA COMPARISON UNDERWAY

3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 8 1979

MEMORANDUM FOR: A. Thadani, Task Manager, A-9, Reactor Systems Branch, DSS

FROM: T. M. Su, Task Manager, A-39, Containment Systems Branch, DSS

SUBJECT: PRELIMINARY QUESTION LIST FOR GE REPORT ON ATWS

Per your request on August 1, 1979, I have prepared the preliminary question list for the GE report titled "Assessment of BWR Mitigation of ATWS, May 1979." A copy of the list is enclosed. Please note that the enclosed question list is intended only for discussion purposes during the coming meeting with GE on August 10. It does not, however, represent the result of our final review on that report. Based on the current review schedule, we expect to complete our review by early September, 1979. We will provide you the results of our evaluation by that time.

T. M. Su

T. M. Su, A-39 Task Manager
Containment Systems Branch
Division of Systems Safety

Enclosure:
As Stated

cc: S. Hanauer
W. Butler
J. Kudrick
L. Ruth

~~8001110672~~

3pp.

D-110

Preliminary Question List
Relating to Sections 5.2.2 and
5.2.3 of GE Report on ATWS

1. Provide a detailed description of the methodology used to calculate ATWS SRV loads. Information should include the following:

   a. All assumptions used in the methodology;

   b. The initial conditions such as SRV line temperature, SRV line volume, suppression pool temperature, and suppression chamber pressure;

   c. Data base and the associated interpretation and justification for the use of the data for ATWS conditions.

2. It is noted that the Caorso test results were used directly for the ATWS SRV loads for both Mark II and Mark III containments. It should be recognized, however, the Caorso plant has its own plant unique conditions related to SRV design such as SRV line volume, submergence, pool area per quencher etc. In addition, the Caorso primary systems and containment conditions during the tests were not near the ATWS conditions. Therefore, we believe that the Caorso test results should not be used without justification. Furthermore, the complete information on the Caorso tests will not be made available for staff review until the end of this year. A generic approach for the use of the test results has not been proposed either by GE or the Mark II owners group. Based on this status, until the above information is provided, the current statistical approach described in NEDO-21061-P (Mark II Containment Dynamic Forcing Functions) or NEDO-11314-08 for Mark III containment should be used. Provide the calculated results for Mark II and III ATWS SRV loads.

3. Provide justification for establishing the temperature difference of 14°F between local and bulk pool temperature. It should be noted that the Monticello T-Quencher tests on February, 1978 were performed under certain specific operational conditions and plant unique geometry. The test results should not be used in a generic sense. Justification is required for the application of this data base for Mark I design.

4. The service water temperature and initial pool temperature of 75°F were used in the calculation of pool temperature response to ATWS events. The rational used to justify these temperatures, i.e., the T-quencher has the capability to condense steam up to saturated local temperature, has not been substantiated. Therefore, additional information is required to justify the use of temperature below design levels.

5. Provide the Caorso data base which was used to establish the 11°F temperature difference for Mark II and III containments.