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ACRS JOINT SUBCOMMITTEES MEETING SUMMARY/MINUTES FOR CONTAINMENT SYSTEMS/STRUCTURAL ENGINEERING
OCTOBER 17, 1989
ROSEMONT, ILLINOIS

PURPOSE

The ACRS Subcommittees on Containment Systems and Structural Engineering held a joint meeting on October 17, 1989 in Rosemont, Illinois. The purpose of this meeting was to continue the discussion in regard to the development of an ACRS paper on containment design criteria for future plants based on present knowledge. A copy of the meeting agenda and selected slides from the presentations are attached. The meeting began at 8:30 a.m. and adjourned at 4:00 p.m., and was held entirely in open session. The principal attendees were as follows:

ATTENDEES

ACRS

D. Ward, Chairman
J. Carroll, Member
C. Wylie, Member
M. Bender, Consultant
D. Houston, Staff

INVITED SPEAKERS

R. Henry, FAI
L. Minnick, Private Consultant
P. North, EG&G-Idaho
W. Snyder, SNL
W. von Riesemann, SNL
A. Walser, Sargent and Lundy

NRC

B. Hardin, RES
G. Bagchi, NRR

REVIEW DOCUMENTS

There were no formal documents to be reviewed at this meeting. The ACRS effort on this subject is in response to a Staff Requirements Memorandum dated July 28, 1988, which was written following an ACRS meeting with the Commission on July 14, 1988.

DESIGNATED ORIGINAL

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RS01 /11

ACTIONS, AGREEMENTS, AND COMMITMENTS

None

DISCUSSION

In his opening comments, D. Ward expressed regrets that C. Siess, Chairman of the Structural Engineering Subcommittee, could not attend due to illness. He indicated that the purpose of the meeting was to discuss containment design criteria for future plants. He stated that over the last five to ten years, there has been a considerable growth of scientific information and a general understanding of the nature of severe accidents. However, this has not jelled into new guidance for designers to use when considering containments or containment systems. He indicated that this was an information gathering meeting to aid in the development of new guidance or design criteria.

W. Synder (SNL) expressed his opinion that it is very timely to develop a modern set of containment system design criteria, but he also feels that it might be too late for some of the advanced designs already on the drawing board. He believes the concept of multiple barriers should be retained. He indicated that the NSSS design is bottom-up engineering while the balance of plant (BOP) is top-down. He stated that 70 to 80 percent of outages at the plants originate in the BOP. He recommended that the total plant be designed on the bottom-up approach to achieve balanced reliability performance across the whole plant. Further, he recommended that all systems be classified as safety systems, dropping the notations of safety-related and non-safety. This approach is being taken in France. He indicated there is a reluctance in the industry to embrace these ideas because of the legacy embedded in the regulatory process. He stated that he has been close to the PRA studies over the past 15 years and that he is uneasy about the conclusions one draws from PRAs. He feels that better conclusions can be drawn from conventional reliability analysis.

P. North (EG&G) discussed the philosophical foundation for the growth of nuclear energy both in the United States and the World. He indicated that the largest growth would be expected in areas with low current per capita energy consumption (~1/8 of USA). Fossil fuel plants have become a greater concern (acid rain, CO_2, greenhouse effect). He indicated that a large number of people must support the use of nuclear energy if it is to make an appropriate contribution. He discussed a foundation to provide the basis for this support based on: (1) containment criteria linked to clear protective objectives, (2) criteria that allow progressive design innovation, and (3) an approach based on rising standards of adequacy. He felt that a judgement by the Commission at this time that a traditional containment structure is necessary would be disappointing on a technical basis. He recommended a sound engineering approach based on best estimate analyses with explicit factors of safety added. He indicated that new systems must demonstrate a robustness in achieving the containment function and that there should be a balance between prevention and mitigation. He noted that longer plant lifetimes might be possible (80 to 100 years). He discussed the approach related to protective objectives: near term similar to EPRI ALWR requirements and long term as eliminating the need for offsite emergency planning. He also recommended testing of a full scale prototype to demonstrate analytical validation and fault tolerance. With these assurances, one could allow progressive design flexibility and strong societal support.

R. Henry (FAI) addressed the question of whether design criteria for containments should be altered. He indicated that there are only two types of containment to be considered: (1) large drys and (2) pressure suppression. These can be designed to: (1) contain fission products, (2) passively contain stored energy, and (3) provide for heat removal over the long term. He provided some calculational results to support various designs to contain stored energy. He concluded that current criteria are bounding and well conceived, thus should be retained. For fission product retention, he indicated that containments must have an integral steel liner. He reviewed the observations made at TMI-2 and Chernobyl and concluded that current criteria are well conceived. He indicated that future designs should focus on: (1) adding water to the core or cavity to cool debris and protect the liner and (2) imbedding the liner in concrete to minimize thermal loads. He discussed severe accident issues and indicated how these could be addressed to enhance the capabilities in the following areas: (1) hydrogen combustion, (2) liner protection, (3) tunnel configuration to restrict debris dispersal, and (4) containment floor design to achieve maximum cooling and minimum debris accumulation.

M. Bender (Querytech) discussed the system concept to define containment. This definition included a boundary closure, a heat sink and a radionuclide trapping or stabilizing capability. He then discussed some characteristics of current containments and reviewed reactor accident experience. He indicated that neither PWR nor BWR containments would contain an ATWS. He noted that no accidents with radionuclide releases have been experienced at high power. He stated that one should have a design basis accident concept but that one should consider realistic times for accident progression and recovery activities. He recommended effective accident sensing devices and systems for controlled containment failure (venting should be considered). He indicated that not enough attention has been given to make things better if the containment failed. He expressed a concern that pressure vessel/concrete codes are not well integrated with regulations and that too few engineers really understand the codes. He discussed a number of aspects that should be revisited to determine the right basis for evaluating containments.

L. Minnick (Private Consultant) reviewed the historical development of containments for Yankee-Rowe and Connecticut Yankee. He indicated that there was a reluctance to install a pressure relief system on these early designs. He recommended that a passive means for cooling core debris and for relieving containment overpressure be considered for future reactors. He further indicated that these devices should have a minimal effect on the basic design of the plant and that these devices must provide a substantial improvement in safety assurance. There should be a careful analysis of any detrimental effects from these devices. He then discussed a self-actuated pressure-relief device for reactor containments. This device was conceived by L. Minnick and investigated for EPRI by Sol Levy, Inc. A copy of the EPRI report was provided. This system is comprised of multiple standpipes with water box seals and was reported to cost about $15M. Mr. Minnick discussed the operational features of the system to relieve pressure, to scrub fission products, and to provide water inside containment.

A. Walser (Sargent & Lundy) discussed containment design criteria from the standpoint of a structural engineer. As a designer, one needs to know the applied loads with some time dependency and probability of occurrence. Given that information and the space requirements for the plant, one can then design and build a suitable containment. He reviewed the current requirements for containment design: LOCA loads from the NRC regulations and combined LOCA plus 1/2 SSE from the ASME code. He discussed safety factors and the effect of discontinuities (penetrations, hatches, stiffners). He indicated that the effect of discontinuities can not be codified. He concluded that current design criteria are adequate and should not be changed in the near future. If they are changed, he recommended that an industrial task force with input from research, universities and NRC be formed to address the matter. Another recommendation was that ASME codes be revised from deterministic to probabilistic in terms of load factors and allowables, and with an emphasis on ductility. He estimated that the efforts in his recommendations would require about 20 years to complete.

W. von Riesemann (SNL) presented his personal thoughts on the subject of containment design criteria, mostly for LWRs. He discussed the primary and secondary purposes for containments. One secondary purpose is to protect against external threats - missiles, tornadoes and sabotage. He indicated that the containment is a system, not an isolated component. Its performance depends on the response of the parts and any possible interactions. He discussed the current approach to designs and the lessons learned from scale tests performed mostly at SNL. He noted that a decade of knowledge on containment behavior and severe accidents has not been factored into the ASME code and in agreement with A. Walser, recommended that a committee be formed to revise the code considering the containment as a system. He discussed goals and some potential difficult points for new requirements. He noted that a probabilistic design approach is beyond the current state-of-the-art.

In the wrap-up session, W. Snyder emphasized a need for better communications between the severe accident analysts and the civil engineers. He felt that civil engineers would have to change their philosophies when designing systems that may go beyond the elastic limit. He also stated that in his discussions with designers, he believes that they are already a half step beyond current requirements for the next generation of plants. R. Henry encouraged designers to think more in terms of thermal loads than pressure loads. He also endorsed a more realistic approach to integrated leak test (proposed by W. von Riesemann) and a more realistic source term analysis. M. Bender emphasized the system approach for containment design and the load conditions as a function of time. W. von Riesemann proposed an ACRS workshop with all interested parties to discuss the conclusions drawn from the joint Subcommittee meetings, a proposal also endorsed by G. Bagchi (NRC/NRR). B. Hardin (NRC/RES) discussed the status of staff activities for evolutionary plants and indicated that efforts for improving the source term are being reactivated. He also discussed the disagreement between the NRC and industry in respect to the metal water reaction for hydrogen calculations [100% MWR (NRC) vs 75% MWR (Ind)].

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<u>NOTE</u>: Additional meeting details can be obtained from a transcript of this meeting available in the NRC Public Document Room, 2120 L Street, NW, Washington, DC 20006, (202) 634-3273, or can be purchased from Ann Riley and Associates, Ltd., 1612 K Street, NW, Suite 300, Washington, DC 20006, (202) 293-3950.

ACRS JOINT SUBCOMMITTEE MEETING
CONTAINMENT SYSTEMS/STRUCTURAL ENGINEERING
OCTOBER 17, 1989
ROSEMONT, ILLINOIS

- TENTATIVE AGENDA -

CONTAINMENT DESIGN CRITERIA FOR FUTURE NUCLEAR PLANTS

A.	Subcommittee Chairmen Remarks	D. Ward/ C. Siess, ACRS	8:30 a.m.

INVITED SPEAKERS

B.	Bill Snyder, SNL	8:45 a.m.
C.	Paul North, EG&G	9:30 a.m.
	**** BREAK ****	10:15 - 10:30 a.m.
D.	Bob Henry, FAI	10:30 a.m.
E.	Mike Bender, Querytech	11:15 a.m.
	**** LUNCH ****	12:00 - 1:00 p.m.
F.	Larry Minnick, Private Consultant	1:00 p.m.
G.	Adolph Walser, Sargent & Lundy	1:45 p.m.
	**** BREAK ****	2:30 - 2:45 p.m.
H.	Walt Von Riesemann, SNL	2:45 p.m.
I.	Subcommittees Discussion	3:30 p.m.
J.	Adjournment	5:00 p.m.

Presentation
to
ACRS Joint Subcommittees' Meeting
on
Containment Systems and Structural Engineering

A. Wm. Snyder
Sandia National Laboratories

October 17, 1989

AWS: 10/15/89

Containment System Design Criteria
for
Future Generations of U. S. Nuclear Power Plants

A difficult challenge, given:

- the variety of candidate NSSS and plant concepts
- the bias of the legacy being limited to the LWR experience
  - the investment in making a success of the concepts and designs of current plants
  - current institutionalization of the U. S. nuclear power enterprise
  - the sharply focussed attention being given to
    - the understanding of severe accidents
    - the predictions of threats to and the response of contemporary containments

The Concept and Design Legacy Vis-a-Vis An Alternative Future Design Agenda

The Concept and Design Legacy:

- design approach
  - NSSS; predominantly "bottom up"
  - Balance-of-Plant; predominantly "top down"
- safety systems; mostly additions/auxiliaries to the base plant
- the containment building, last barrier of the multiple defenses-in-depth, designed to withstand a surrogate (DBA) for all plausible accidents
- the multiple sequential barriers of the defenses-in-depth susceptible to common cause and interdependent failures

The Concept and Design Legacy
Vis-a-Vis
An Alternative Future Design Agenda

An Alternative Future Design Approach

- NSSS & BOP; both designs mostly "bottom up
- no distinctions between safety, safety-related, and non-safety systems

The Concept and Design Legacy Vis-a-Vis An Alternative Future Design Agenda

An Alternative Future Design Approach (continuing)

- Total Performance Management (TPM)

  Total - complete plant system; over the full projected plant life

  - optimization of the performance of the complete plant system to all vital performance success indices (safety, economics, etc.)
  - include in design, full objective consideration of both deterministic and probabilistic events and their costs
  - excellence keyed to plant system reliabilities as metrics of quality attained in design,operations, maintenance, and management.

Translation to the objectives of safety, the "language" of containment and containment systems, and the definition of design and performance criteria w/r/t internal events

- Retain the cardinal concept of multiple barriers to attain safety-in-depth

- Define multiple reliability criteria as indices of successful performance for each of the multiple barriers to attain safety-thru-quality, e.g.,

  - the reliability of a barrier to withstand successfully credible threats from credible internal initiators

  - the reliability of the collective internal systems that credibly, thru failure and malfunction, could initiate a threat to the barrier

- Define a total plant system reliability criterion as an index of successful performance of the composite containment function

Idaho
National
Engineering
Laboratory

COMMENTS ON CONTAINMENT DESIGN CRITERIA FOR FUTURE NUCLEAR PLANTS

P. NORTH

WORLD ENERGY PICTURE

- LARGE ENERGY CONSUMPTION GROWTH PROJECTED BY WORLD ENERGY STUDIES
- LARGEST GROWTH IN AREAS WITH LOWER CURRENT PER CAPITA ENERGY CONSUMPTION THAN IN THE UNITED STATES
- GLOBAL ENVIRONMENTAL CONCERNS IF THIS GROWTH IS PROVIDED BY FOSSIL FUEL BURNING
- INDICATIONS THAT UNITED STATES, EUROPEAN AND JAPANESE NUCLEAR INDUSTRIES WILL SEEK TO SERVE THIS GLOBAL ENERGY MARKET.

CONCLUSION - WE MUST ADDRESS THE POSSIBILITY OF MUCH WIDER USE OF NUCLEAR ENERGY THAN IS EVIDENT TODAY AND IN A MUCH BROADER GEOGRAPHIC AND SOCIETAL SETTING

A RESULTANT FOUNDATION

- CONTAINMENT CRITERIA LINKED TO CLEAR PROTECTIVE (REGULATORY) OBJECTIVES FORMULATED ON THE BASIS OF WIDE APPLICATION OF NUCLEAR ENERGY WITHIN THE UNITED STATES AND IN THE WORLD AT LARGE

- CONTAINMENT CRITERIA THAT ALLOW PROGRESSIVE DESIGN INNOVATION IN MEETING THE PROTECTIVE OBJECTIVES

- AN APPROACH BASED ON RISING STANDARDS OF ADEQUACY FROM DESIGN GENERATION TO DESIGN GENERATION

- AN APPROACH AND RELATED METHODS THAT PROVIDE THE BASIS FOR STRONG SUPPORT OF NUCLEAR ENERGY BY LARGE NUMBERS OF PEOPLE

DEFINING THE APPROACH

RELATED CONDITIONS

- SOUND ENGINEERING APPROACH
  - BEST ESTIMATE, MECHANISTIC ANALYSES
  - SUPPORTED BY ADEQUATE PHYSICAL UNDERSTANDING
  - "FACTORS OF SAFETY" ADDED EXPLICITLY

- THIS APPROACH CAN BE UNDERSTANDABLE AND CONVINCING TO PEOPLE NOT INVOLVED IN THE WORK AND IS THEREFORE CONDUCIVE TO THE GENERATION OF SUPPORT

DEFINING THE APPROACH

RELATED CONDITIONS

- THE NEW SYSTEMS SHOULD DEMONSTRATE ROBUSTNESS IN ACHIEVING THE CONTAINMENT FUNCTION
  - USE OF BASIC PHYSICAL CHARACTERISTICS
  - FAULT TOLERANCE
  - CAREFUL IMPLEMENTATION OF DEFENSE IN DEPTH WITH INDEPENDENT MULTIPLE LAYERS, EFFECTIVE FOR THE ENTIRE ACCIDENT SPECTRUM
  - ABSENCE OF THE POSSIBILITY OF BYPASS
- BALANCE BETWEEN PREVENTION AND MITIGATION (THERE WILL ALWAYS BE RESIDUAL UNCERTAINTY IN PREVENTION)

DEFINING THE APPROACH

RELATED CONDITIONS

- POSSIBILITY OF LONGER PLANT LIFETIMES (80 TO 100 YEARS)
  - ORIGINALLY REMOTE LOCATIONS MAY BECOME MORE POPULATED
  - IT WILL NOT BE A SERVICE TO SOCIETY TO LIMIT LAND DEVELOPMENT POSSIBILITIES
- WITH INCREASING "NUCLEAR FLEET" APPROACHES THAT ALLOW EVEN THE REMOTE POSSIBILITY OF FARMLAND WITHDRAWAL AND CLOSURE OF NEIGHBORHOODS (CHERNOBYL) WILL BE INCREASINGLY UNACCEPTABLE TO SOCIETY
- BOTH OF THESE FACTORS MILITATE FOR AN APPROACH THAT CONCENTRATES ON THE CHARACTERISTICS OF THE PLANT ITSELF AND DOES NOT RELY ON EXTERNAL RESPONSES BY THE REST OF SOCIETY

APPROACH AND RELATED METHODS

FOUNDATION ELEMENT - RISING STANDARDS OF ADEQUACY

- CONSISTENT WITH THE ADVANCED REACTOR POLICY STATEMENT
- LEVELS OF "ADVANCED DESIGNS" SHOULD BE RECOGNIZED AND APPROACHES DEFINED ACCORDINGLY
  - DESIGNS THAT ARE A LOGICAL EVOLUTIONARY STEP FROM OPERATING LWRS - BUILD FROM EXISTING RULES; DEMONSTRATE COMPLIANCE WITH SEVERE ACCIDENT POLICY; DEMONSTRATE IMPROVED FISSION PRODUCT RETENTION; COUPLE WITH FEATURES SUCH AS LONG TRANSIENT TIME; DESIGN TO TIGHTER PROTECTIVE OBJECTIVES
  - DESIGNS THAT REPRESENT A GREATER DEVELOPMENT STEP AND ARE AIMED AT LATER DEPLOYMENT - USE MORE PERFORMANCE RELATED CRITERIA TO ALLOW DESIGN INNOVATION; ESTABLISH EVEN TIGHTER PROTECTIVE OBJECTIVES

APPROACH AND RELATED METHODS

FOUNDATION ELEMENT - CONTAINMENT CRITERIA RELATED TO PROTECTIVE OBJECTIVES

- NEAR TERM ADVANCED LIGHT WATER REACTORS

  CORE DAMAGE FREQUENCY $\leq 1 \times 10^{-5}$ PER YEAR SITE BOUNDARY
  WHOLE BODY DOSE LESS THAN 25 REM FOR ACCIDENTS WHOSE CUMULATIVE
  FREQUENCY EXCEEDS 1×10^{-6} PER YEAR

- LONGER TERM OBJECTIVES

  GO BEYOND CONSIDERATION OF NO OFFSITE EMERGENCY PLANNING
  REQUIREMENT AND MAKE THIS CONDITION A SPECIFIC DESIGN OBJECTIVE

SHOULD THE CONTAINMENT DESIGN CRITERIA BE ALTERED?

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ACRS SUBCOMMITTEE MEETING

CHICAGO, ILLINOIS
OCTOBER 17, 1989

BASIC CRITERIA FOR ACCIDENT CONDITIONS

1. CONTAIN FISSION PRODUCTS RELEASED FROM THE FUEL AND THE PRIMARY SYSTEM (FIRST AND SECOND BARRIERS).

2. PASSIVELY CONTAIN (ACCOMMODATE) THE ENERGY STORED IN THE PRIMARY SYSTEM COOLANT AND FUEL AT NORMAL OPERATING CONDITIONS. (LARGE LOCA IS A WAY OF CONCEPTUALIZING THIS REQUIREMENT.)

3. REMOVE DECAY HEAT OVER THE LONG TERM.

CRITERION: CONTAIN THE ENERGY STORED IN THE PRIMARY SYSTEM

. THE PREVIOUS CALCULATIONS ARE ONLY APPROXIMATE TO ILLUSTRATE THE SIZES NECESSARY TO SATISFY THE CRITERION.

. OTHER ASPECTS NEED TO BE CONSIDERED, PARTICULARLY THOSE ASSOCIATED WITH NORMAL OPERATION.

. CONCLUSION - THIS CRITERION FOR CURRENT PLANTS:
  - IS ENVELOPING.
  - IS WELL CONCEIVED.
  - SHOULD BE RETAINED FOR FUTURE PLANTS.

CRITERION: CONTAIN FISSION PRODUCTS RELEASED FROM THE FUEL AND PRIMARY SYSTEM

- FOR THE TWO CONCEPTS USED IN THE U.S., THE CONTAINMENT COULD POTENTIALLY PRESSURIZE FOR TENS OF MINUTES OR LONGER DURING A SEVERE ACCIDENT.

- TO SATISFY THE CRITERION, THE CONTAINMENT MUST HAVE AN INTEGRAL STEEL LINER.

CRITERION: CONTAIN FISSION PRODUCTS RELEASED FROM THE FUEL ON THE PRIMARY SYSTEM

. CONCLUSION: WHILE THE VALUES SHOWN IN THE PREVIOUS SLIDE ARE APPROXIMATE, IT IS CLEAR THAT THE CRITERION FOR CURRENT PLANTS IS:

- WELL CONCEIVED, AND
- SHOULD BE RETAINED FOR FUTURE PLANTS.

OTHER LESSONS FROM REACTOR ACCIDENTS

. The TMI accident was caused by a lack of water.

. The TMI accident was terminated by adding water.

. The damaged Chernobyl reactor was stabilized for several hours by water addition (fire fighters) but was haulted because the water was spilling into and contaminating Units 3, 2 and 1.

. CONCLUSION: Water would be very effective in recovering from an accident state and future designs, like the current plants, should focus on ways to supply water to the containment and to remove the decay heat.

CONTAINMENT LINER INTEGRITY IS IMPORTANT

FUTURE DESIGNS SHOULD FOCUS ON

- COOLING THE DEBRIS TO PROTECT THE LINER.
- IMBEDDING THE LINER IN CONCRETE TO MINIMIZE THERMAL LOADS FROM DEBRIS.
- OR BOTH.

FUTURE DESIGNS CAN ADDRESS SEVERE ACCIDENT ISSUES

- LIKE THE CURRENT PLANTS, FUTURE DESIGNS SHOULD PROTECT AGAINST OVERPRESSURE DUE TO HYDROGEN COMBUSTION.

  - VOLUME AND ULTIMATE PRESSURE CAPABILITY TO ACCOMMODATE A COMPLETE BURN OF HYDROGEN GENERATED BY THE OXIDATION OF 75% OF THE ACTIVE CLADDING.

  - INERT THE CONTAINMENT.

  - INTENTIONAL IGNITION (IGNITERS).

- PROVIDING PROTECTION FOR THE LINER.

  - WATER.

  - IMBEDDED.

FUTURE DESIGNS CAN ADDRESS SEVERE ACCIDENT ISSUES BY

(CONTINUED)

- USE A REACTOR CAVITY/INSTRUMENT TUNNEL CONFIGURATION WHICH DRASTICALLY REDUCES OR ELIMINATES THE POTENTIAL FOR DEBRIS DISPERSAL GIVEN A HIGH PRESSURE MELT EJECTION CONDITION.

- MAXIMIZE THE CAPABILITY OF PUTTING WATER ON THE CONTAINMENT FLOOR.

- MAXIMIZE THE POTENTIAL FOR ACCIDENT RECOVERY BY MAXIMIZING THE FLOOR AREA FOR DEBRIS ACCUMULATION.

CONCLUSIONS

- THE GENERAL CRITERIA USED FOR DESIGNING THE CURRENT PLANTS ARE WELL CONCEIVED.

- THE PRUDENCE OF THE CRITERIA USED IN THE U.S. IS DEMONSTRATED BY THE EXPERIENCE FROM REACTOR ACCIDENTS.

- THE GENERAL CRITERIA USED FOR CURRENT PLANTS ARE APPLICABLE TO FUTURE DESIGNS.

- THE IMPLEMENTATION OF THE CRITERIA CAN BE STREAMLINED.

- FUTURE DESIGNS COULD ADDRESS SEVERE ACCIDENT ISSUES TO REDUCE THE INFLUENCE OF UNCERTAINTIES.

CONTAINMENT DISCUSSION

Presented to the NRC ACRS Subcommittees on Containment and Structures--Chicago, Illinois, October 17, 1989

Prepared by M. Bender, Querytech Associates, Inc.

- 0 DEFINITION OF CONTAINMENT, A SYSTEMS CONCEPT
- 0 REFERENCE EXPERIENCE
- 0 CURRENT UNDERSTANDING FROM NRC AND INDUSTRY SPONSORED RESEARCH
- 0 DEVELOPING A DESIGN BASIS

CONTAINMENT DEFINED:

A SYSTEM INTENDED TO PREVENT THE SPREAD OF RADIONUCLIDES, RELEASED IN BULK FROM THE REACTOR CORE, BEYOND SPECIFIED SITE LIMITS IN THE EVENT OF A NUCLEAR ACCIDENT.

ESSENTIAL SYSTEM PROPERTIES:

1 Boundary closure sufficient to limit dispersal of radionuclides postulated to be present during and subsequent to an accident,

2 An effective heat sink to absorb nuclide decay energy and stored energy in coolants and surrounding structure for the purpose of controlling temperature conditions to limit subsequent chemical, physical state, or fluid perturbations that would aggravate radionuclide dispersal conditions,

3 Radionuclide trapping or stabilizing capability to prevent further dispersal of all but the noble gases during and subsequent to an accident including those caused by transient effects. (Holdup to permit noble gas (xenon) decay can be a valuable capability, but the trapping mechanisms must be of high reliability; the physical flow path may be the most effective device for this purpose.)

REFERENCE REACTOR ACCIDENT EXPERIENCE

1. NO RADIONUCLIDE RELEASES AT HIGH POWER

2. OPERATOR ALERTNESS HAS PREVENTED FUEL FAILURE AT POWER (E.G. BROWNS FERRY ATWS, DAVIS BESSE FEEDWATER TRANSIENT)

3. PREVIOUS PRACTICE HAS EXCLUDED SEVERE EVENTS FROM CONTAINMENT REQUIREMENTS (BWR ATWS, CORE COOLANT BLOCKAGE)

4. EARLY ACCIDENTS IN SMALLER INSTALLATIONS HAVE GUIDED SAFETY REQUIREMENTS (SL-1, NR-X, WINDSCALE)

5. TARAPUR AND CHERNOBYL SHOWN POTENTIAL RISK (NOT AS EXTENSIVE AS "DOOMSDAY" PREDICTIONS BUT EXTENSIVE AND SERIOUS)

6. TMI-2 SHOWED THAT CORE MELTING DOES NOT NECESSARILY VIOLATE CONTAINMENT. WITH MINIMAL COOLING UNDER SHUTDOWN CONDITIONS LOW CONTAINMENT PRESSURES EASILY MAINTAINED. LOW LEAKAGE WASN'T HARMFUL.

WHAT ARE THE LESSONS FROM ACCIDENT RESEARCH?

1. ACCIDENT PROGRESSION

1.1 "Murphy's Law" logic does not give effective design guidance.

1.2 Unencumbered accident progression will inevitably lead to imponderable accident conclusion.

1.3 Time is available for control accident interdiction.

1.4 The operator is an important part of accident control and operator interdictive provisions should not involve complex logic based on accident progression analysis.

LESSONS FROM SEVERE ACCIDENT RESEARCH

2. CONTAINMENT STRUCTURAL RESPONSE

2.1 Containment structural behavior is predictable and reliable up to elastic response limits. Reinforced concrete appears to provide non-catastrophic failure capability beyond elastic response limits.

2.2 Liner reliability contingent on assuring controlled structural movement under accident loadings--discontinuities still the major uncertainty in liner response.

2.3 Closures sealed with elastomers are the main source of leakage vulnerability. Experimental testing suggests that up to the point of significant leakage (observable flow) gasket materials in current use are functionally effective over the anticipated times of active accident progression if protected from overtemperature and intense radiation.

DESIGN BASIS

"DESIGN BASIS ACCIDENT" DEFINITION

1. ACCIDENT INITIATORS NEED TO BE POSTULATED--LOCA'S, LOPA'S, STEAM GENERATOR RUPTURES, ETC.

2. SEVERITY OF THE CONDITION NEEDS BETTER RATIONALE I.E. WORST CONDITION LOCA'S DISTORT BEHAVIORAL CHARACTERISTICS AND MISUSE SAFETY RESOURCES--EXAMINE SYSTEM PROPERTIES FOR A REALISTIC ACCIDENT BASIS.

3. ATWS TYPE EVENTS NEED TO BE INCLUDED IN SOME FORM. ENOUGH EXAMPLES EXIST TO DEFEAT ANY PROBABILISTIC ARGUMENT THAT THEY ARE OUT OF THE REALM OF PROBABILITY.

4. RADIONUCLIDE RELEASES SHOULD BE BASED ON REAL TIME EVENTS--ARBITRARY RELEASES DO NOT PROPERLY CHARACTERIZE THE ACCIDENTS AND DO NOT EFFECTIVELY COMBINE RELATED CIRCUMSTANCES.

DESIGN BASIS

"DESIGN BASIS ACCIDENT" DEFINITION

5. ACCIDENTS SHOULD NOT BE ASSUMED TO PROCEED TO THEIR NATURAL ENDPOINT UNLESS THE INTERDICTIVE OPPORTUNITIES ARE BEYOND ACCESS. AN ATWS MIGHT NOT BE CONTROLLABLE; A SMALL LOCA HEAT SINK BYPASS COULD BE CORRECTED IF KNOWN TO EXIST. ACCIDENT SENSING NEEDS TO BE BUILT IN TO THE DBA ASSESSMENT.

6. DESIGN CONTAINMENT ENCLOSURE FOR CONTROLLED FAILURE; ALLOW CONDITIONS NEAR TO STRUCTURAL YIELDING AND PROVIDE RUPTURE RELIEF THROUGH A KNOWN TRAPPING PATH BEFORE BURSTING.

7. PROVIDE FOR EFFICIENT TRAPPING MEDIA SUCH AS CAUSTIC SPRAYS, CHEMICALLY ACTIVE TRAPPING PONDS, RUGGED AND ACCIDENT INSENSITIVE TRAPPING DEVICES LIKE "SAND FILTERS".

SELF-ACTUATED PRESSURE RELIEF DEVICE
FOR
REACTOR CONTAINMENTS

(CONCEIVED BY L. MINNICK;
INVESTIGATED FOR EPRI BY S. LEVY, INC.)

PATENT APPLIED FOR BY EPRI

FUNDAMENTAL PURPOSE

TO PREVENT OVER-PRESSURIZATION OF REACTOR CONTAINMENT DURING ANY POSTULATED ACCIDENT OTHER THAN INSTANTANEOUS RELEASE OF ENERGY

SELF-ACTUATED PRESSURE RELIEF DEVICE
FOR
REACTOR CONTAINMENTS

ADDITIONAL FUNCTIONS PERFORMED

- SCRUBS RELEASED GASES OF PARTICULATES AND ANY MATERIAL HAVING AN AFFINITY FOR WATER.

- PROVIDES DILUTED, ELEVATED AND HEATED RELEASE OF NOBLE GASES.

- CONDENSES ESSENTIALLY ALL STEAM AND RETURNS THE WATER FORMED TO THE CONTAINMENT.

- REESTABLISHES CONTAINMENT INTEGRITY WHENEVER CONTAINMENT OVER-PRESSURE IS TERMINATED.

- PROVIDES RELIEF OF POTENTIAL CONTAINMENT VACUUM FOLLOWING INCIDENT.

SELF-ACTUATED PRESSURE RELIEF DEVICE
FOR
REACTOR CONTAINMENTS

INHERENT CHARACTERISTICS

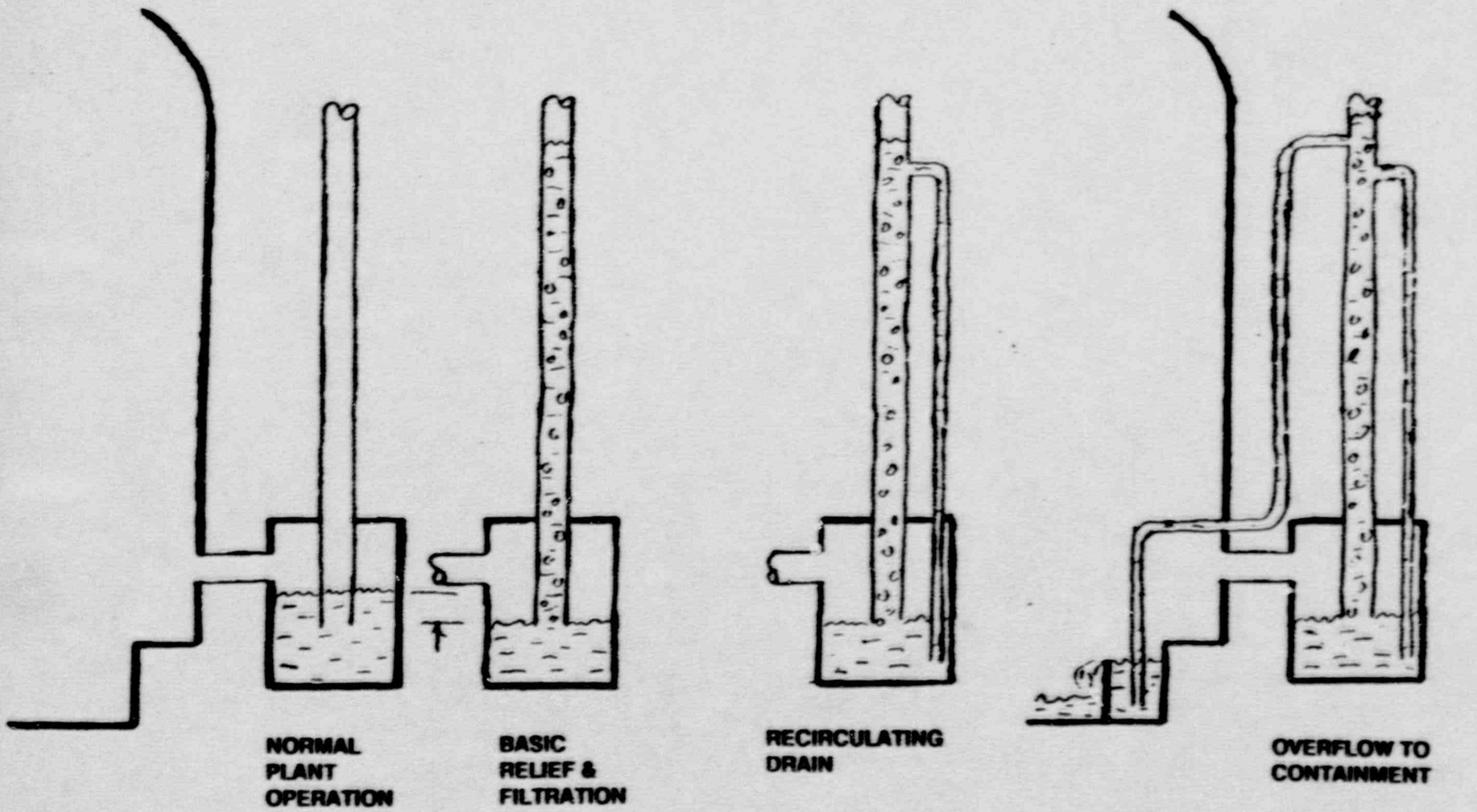
- TOTALLY PASSIVE ACTUATION, OPERATION AND RESET:

  - NO ACTIVE DEVICE OR MECHANISM,
  - NO OPERATOR ACTION,
  - NO POWER REQUIREMENT,
  - NO INSTRUMENTATION OR CONTROL, AND
  - NO MAKEUP WATER

  ARE REQUIRED THROUGHOUT THE COURSE OF THE TRANSIENT, REGARDLESS OF DURATION.

- SHIELDS ALL RADIOACTIVE MATERIAL COLLECTED AND, ULTIMATELY, CONTAINS WHATEVER HAS NOT BEEN RETURNED TO THE CONTAINMENT IN A SINGLE UNDERGROUND TANK.

SAPRD ~ IN PRINCIPLE



ENHANCES:
- STABILITY
- HEAT REMOVAL
- PRE-COOLING & MIXING

- MAINTAINS LEVEL IN STANDPIPE
- MAINTAINS POOL OF WATER UNDER REACTOR

Review of Present Structural Containment Design

- The LOCA load is well defined. The NSS supplier provides this load. It is coupled to the reactor's thermal capability.
- The ASME Containment Codes are complete. They are:

  Section III - Division 1 - Subsection MC,

  Section III - Division 2 - Subsection CC,

  and have been developed and are maintained by the Industry, Research, and Universities with participation by the NRC. These codes are based on LOCA loads.

Review of Present Structural Containment Design

- The containment capability of existing containments for an upper bound pressure load have been determined and safety margins compared to LOCA loads have been computed. The acceptance criteria in all these capability evaluations were beyond code allowables.
- Based on these studies, containments designed to current codes show considerable margins.
- Some of these results used in PRA have shown acceptable risk to public within current understanding of acceptable risk.
- Testing by Sandia of scaled containment models in steel and reinforced concrete have shown that in most cases, the scaled containments behave in a ductile manner. (leak before break)

Review of Present Structural Containment Design

- The work required to determine the containment capabilities was sponsored by:

  The Industry Degraded Core Rulemaking Program,

  Utilities commissioning plant unique probabilistic risk assessment studies,

  Sandia-NRC sponsored workshops,

  Sandia effort on NUREG 1150.

- The Advanced Light Water Reactor Study utilizes a containment designed for LOCA loads and using the ASME Code. System and layout provisions are made in consideration of severe accidents.

Review of Present Structural Containment Design

- Lessons learned from the Containment Capability Studies have highlighted that the containments must be ductile and must not have a weak link anywhere. Designs and care of details is of utmost importance and can be provided within current design basis.

Conclusion

- The present Structural Containment Design Criteria is adequate and should not be changed in the near future.

Recommendations for Future Development

- It is recommended that an industry effort, in participation with research, universities and the NRC, should be undertaken to develop loads and design criteria for containment based on severe accidents.
- The goals of this effort should be: Define severe accident loads in terms and ways that can be utilized in structural design without ambiguity.
- A consensus has to be reached regarding the events involved in a severe accident. Loads, in terms of time dependent pressures and temperatures and their probability of occurence have to be established.
- A consensus has to be reached regarding an acceptable probability of risk to the public in case of a severe accident.

Recommendations for Future Development

- Future structural designs will be based on probabilistic assessement of loads and resistance to achieve a safe structure. When this can be done appropriately, it is then the proper time to change the containment design basis.
- Revise present ASME design codes from deterministic to probabilistic in terms of load factors and allowables, and emphasize ductility.
- Based on the present work of the Advanced Light Water Reactor Industry Group, future containments may have only one of two configurations: the large dry containment for PWRs and a modified Mark II containment for the BWRs. Limiting consideration to these possibilities will facilitate the above tasks considerably.
- It is anticipated that such efforts will require a considerable amount of time.

THOUGHTS AND REFLECTIONS ON CONTAINMENT DESIGN CRITERIA

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PRESENTATION TO
ACRS JOINT SUBCOMMITTEE MEETING
CONTAINMENT SYSTEMS/STRUCTURAL ENGINEERING

OCTOBER 17, 1989

<u>SUMMARY</u>

- A DECADE OF KNOWLEDGE ON CONTAINMENT BEHAVIOR AND SEVERE ACCIDENTS HAS NOT BEEN FACTORED INTO THE ASME CODE

- RECOMMEND THAT A COMMITTEE (INDUSTRY, RESEARCHERS, REGULATORS) BE FORMED TO REWRITE THE CODE (DESIGN, FABRICATION, INSPECTION INCLUDING LEAK RATE MEASUREMENTS, SEVERE ACCIDENTS) CONSIDERING THE CONTAINMENT AS A <u>SYSTEM</u>

  FIRST STEP WOULD BE TO DETERMINE THE PHILOSOPHY

CONTAINMENT (cont'd)

- CONTAINMENT IS A SYSTEM--NOT AN ISOLATED COMPONENT (SHELL)

  I.E. SYSTEM CONSISTS OF

  Structure (Shell)
  Penetrations (Operable and Fixed)
  Bellows
  Drywell Head (BWR)
  Fuel Transfer Tubes
  Isolation Valves
  Basemat
  Instrumentation (Status of System)

  THE PERFORMANCE (BEHAVIOR) DEPENDS ON THE RESPONSE OF ALL OF THE PARTS AND ANY POSSIBLE INTERACTIONS; e.g., REACTOR VESSEL SUPPORT FAILURE WHICH THEN WILL LOAD CONTAINMENT THROUGH THE STEAM LINES.

LESSONS LEARNED

- Current Design Personnel Airlocks and Electrical Penetration Assemblies (Except for Electrical Peformance) Behaved Well (Leakage and Strength)

- Equipment Hatches
  Sleeve Ovalizes -- Leakage May Occur
  Pressure Unseating--Not Desirable

- Seals and Gaskets -- Performed Well Up to About 500°F

- Inflatable Seals -- Leakage will Occur at Overpressurization

- Basemats -- Data from a Recent Test Result has to be Interpreted; Additional Work may have to be Performed.

LESSONS LEARNED (cont'd)

- Stiffening Around Penetrations and 'Area Replacement' Rule Causes Strain Risers and May Lead to Early Failure

  In Particular, for Liners With Studs and (on Ring Stiffened) Steel Cylinders

- Basemat -- Cylinder Intersection in Reinforced Concrete Containments is Overdesigned

- Tori-spherical Heads do Buckle but do not Fail (i.e. Leak) till the Pressure is Several Times the Buckling Pressure

- Consequences of a Core/Concrete Interaction Depend on the Chemical Composition of Concrete

LESSONS LEARNED (cont'd)

- Substantial Corrosion of the Steel (Where it Enters the Concrete) May Occur

- Aerosol Retention in Concrete has not been Quantified

- Retention in Secondary Buildings has not been Quantified

- Containments have had Isolation Valves Left Open for Extended Periods

GOALS FOR THE NEW REQUIREMENTS

- Benign failure modes
- Long Life
- Simple Inspection, Including On-Line Monitoring
- Construction Ease
- Designers must become Familiar with Severe Accidents and Loads Beyond the Design Basis and the Fact that some Loads are not well Defined; i.e., Mind Set must Change

GOALS FOR THE NEW REQUIREMENTS (cont'd)

- Internal Structure (Compartments, Rooms) should be Designed to Minimize Effects of Fire, Flooding and Hydrogen Combustion.

- Realistic Leakage Requirements

- Realization that Buckling, per se, is not Necessarily Failure

POTENTIAL DIFFICULT POINTS

Definition of Loads

Design Criteria vs. Performance Requirements

Overpressure Protection

Leak Rate Testing

Current Licensing is done on a Prescriptive
Basis--Difficult to Accommodate Guidelines

Probabilistic Design Beyond the Current
State-of-the-Art