UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 7, 1977

Carl Stahle
LPM Pebble Springs Nuclear Plant

SUBJECT: ACRS QUESTIONS RE PEBBLE SPRINGS REVIEW

Attached are questions raised by an ACRS member, to which the Pebble Springs Subcommittee would like written responses prior to ACRS full Committee review of that project.

At this moment it is not planned to schedule another Subcommittee meeting prior to full Committee review, therefore it is requested that responses be provided as early as possible.

Ragnwald Muller
Senior Staff Engineer

ATTACHMENT
Questions raised by ACRS
Member

cc: R. Boyd
L. Crocker
S. Varga
T.H. Cox (2 copies)
J.C. McKinley
M.W. Libarkin
J.C. Ebersole
S.H. Bush
M.S. Plesset
H.S. Isbin
D. Okrent

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TOPICS ON PEBBLE SPRINGS (related to BSAR-205)

1. Provide the intrepretation used in design, of GDC 19 and Reg. Guide 1.75 (IEEE 334).

   The less conservative interpretation of GDC 19 does not allow common damage in control room.

   RG 1.75 permits convergence of total plant shutdown capability down to spacing measured in inches (with some form of panel or plate type of barrier) to a few feet of open space.

   More conservative interpretation of GDC 19 would require (as IAEA does) that safe shutdown can be accomplished if the control room (and presumably any other given safety "space") is subject to common damage within that space.

   Use of the less conservative interpretation of these criteria results as a "soft" design with extremely heavy requirements on "administrative control". If the design is "soft" describe the correspondingly "hard" administrative controls.

2. Clarify the rationale used for location of straight sections of main steam and feedwater lines in respect to potential damage to safety equipment. Is it assumed that such pipe sections are infallible?

3. Does the design accommodate potential for inadvertent flooding from vessel and piping failures within "safety" structures or in such areas where safe-shutdown equipment is located?

4. What is stress-level and maximum local deformation in steam-generator tubes and tube sheet as result of Post-LOCA flooding of tube-side of superheat section of steam-generators? Would some tube failures at this point in time seriously affect core cooling?

5. What is the maximum secondary system pressure developed after turbine trip with first subsequent random failure being loss of main feedwater flow control leading to flooding of superheat section of steam generators. Assume turbine trip without bypass (loss of condenser vacuum).

6. Does applicant know that time-dependent levels will occur in pressurizer, steam generator and reactor vessel after a relatively small primary coolant break which causes coolant to approach or even partly uncover fuel pins? What does operator do in respect to interpreting level in pressurizer?

During primary system refill from high pressure injection pumps there is some period when neither condensation nor natural convection is present to effect heat transport to secondary side. How is transition to natural convection <u>without assistance from primary coolant pumps</u> obtained.

7. What is the particular design of the start-up piping and pumping system for Pebble Springs? Does it involve operating with a liquid-solid secondary system? Has the Staff performed a safety analysis of this system?

8. Can the plant obtain access to the low-pressure RHR system from the high-pressure condition <u>using only safety grade</u> equipment?

9. Defend the rationale of having only two "active" service systems which perform continuing or long-term safety functions. The first "accident" is the failure of one train thus destroying "normal" redundancy. Dependence on a single system in terms of consequence of failure of that remaining system is essential to understanding intrinsic risks of such designs.

Describe each such system and consequence of total failure of services provided by that system as a function of time. Only "active" failures beyond first failure need be considered.

Possible examples of such systems are:

1. Battery (DC power system) (consider parasitic loads)
2. On-site AC power system - assuming prior loss of off-site AC system
3. Service water system
4. Component cooling system
5. Environmental control (HVAC) systems

"Redundancy" may be expressed in terms of time to restore service by any means whatever before undue damage ensues.

10. What are off-site dose levels resulting from Steam-Generator tube failure, associated with loss of off-site AC power due to upset from turbine generator trip? What is probability of such a grid failure following turbine trip?

11. Are any special precautions taken for storage and handling of hydrazine?

12. What is status of investigation of merits of a primary vessel coolant level indication system for use in post LOCA cooling for small breaks?

13. The fire protection system may be characterized as a "hard" or "soft" system in respect to independence or dependence on fire detection and extinguishing systems.

    In a local sense, in what particular locations is this plant dependent on administrative protection and early detecting-extinguishing techniques to protect vital shutdown system from fire damage? Is complete burnout assumed for local plant space or area such as one spreading room?

14. As a general principle why is the design heavily dependent on the component cooling system for safe shutdown rather than using the presumably more reliable service water system? Both concepts are used in the industry.

15. As an example of equipment separation which may be overlooked, describe the separation of the compressors for safety grade air cooling systems.

16. Describe the inlet-air protection system for the main control room.

What dose level would be imposed on operators after a LOCA with "realistic" releases (Not TID) to containment but with a single failure being that of electrical blow-out of an intermediate size penetration (say 10" dia.)?

17. Describe electrical protection for power-carrying penetrations subject to in-containment faulting during LOCA. Include penetration for main coolant pumps. Describe protection in context of both overcurrent trip and ground fault (arcing) protection to prevent electrical burnout and thus loss of mechanical integrity of the penetration. Include penetrations handling non-safety grade power circuits.

18. Page 9.9 describes what is apparently an electrical cooling system for Auxiliary Feedwater Pump rooms. Diversity was the basis for requiring engine driven Aux. Feedwater pumps, yet apparently electrically powered room cooling is necessary to assure the engine-driven function. Please clarify.

19. In respect to the volcanic ash problem:

    a. Are the diesel-engine air filters designed to prevent disabling uptake of ash to the engine during this situation?

    b. What other air uptakes have been evaluated to insure continued safe operation to shutdown during this condition such as:

       Control room ventilation and cooling

       Diesel generator air cooling

       Aux feedwater engine air cooling

       Service water motor cooling

       Any other critical air cooling system

20. For a main steam line failure inside containment followed by the first random failure being that of the opposite main steam line isolation valve to close, describe how excess flow is prevented through "non-qualified" valve failures such as turbine by-pass valves.

    In this connection, clarify the rationale which, in some designs, assumes that the large LOCA is "coincident (!)" with an earthquake but, assuming no LOCA, the failure of <u>other kinds of "passive" elements</u> (such as main steam lines in containment) cannot be tolerated - since subsequent application of the <u>single random failure</u> criterion would destroy critical active services.

21. Are the main feedwater isolation valves designed to provide the closing function in a bi-directional flow sense? Is instrumentation diversified to assure main feedwater flow interruption when required? Does this include separate d-c or inverter powered systems?

What prevents spurious closure of main feedwater systems in the light of the critical need to stop such flow when necessary? What is the estimated frequency of such closures as the original accident?

22. The SER indicates that certain cables will be tested for water resistance by submergence.
How often will this be done and what is the probable frequency of exposure to this condition during operation?
Is this sort of testing program proposed for the electrical wiring and penetrations within containment. If not, why not?

23. In once-through steam-generator designs, the auxiliary feedwater system must respond very promptly after main feedwater is tripped. Furthermore, the main feedwater system is presumably assured to trip during any significant seismic event.

Against these conditions it appears to be poor practice not to seismically qualify the condensate storage tank as the viable "passive" source of critical feedwater following a post-earthquake trip and shutdown. The present design does not require this but, instead, depends on the electrically driven (stopped and restarted on diesel power) service water system to provide suction to the Auxiliary Feedwater pumps. For this particular condition, the advantage of the diverse engine driven Aux Feedwater pumps is lost since suction must be provided by the electrically powered service water pumps.

Why has the design evolved in this manner?

24. From the standpoint of finding the worst credible situation in the context of the maximum rate and degree of subcooling of the unbroken primary coolant system, it appears that main steam line failure within containment (which disables pressurizer heaters and provides ECCS trip signals) coupled with failure of main feedwater trip, is probably the worst configuration (It is also presumably intolerable, if persistent, from the standpoint of containment pressurization).

Discuss the consequences of this event in respect to:

a. Degree and rapidity of return of fission power after rod insertion.

b. Thermal gradients in most severely affected parts of reactor vessel and steam generators and subsequent sudden rise of primary coolant pressure to safety valve setpoints after chilling the interior face of the vessel.

c. Maximum containment pressure as function of time of continued run-on of main and/or auxiliary feedwater flow to the failed steam generator.

25. In the startup of newer design B&W systems, using comparatively large pumps and piping and using a water-solid secondary system, the temperature of the water in the secondary system is raised to 400-500° and subsequently the secondary is drained until normal level is obtained. Has the Staff examined the safety aspects of this system?

26. Considering such matters as (1) off-site power failure, (2) condenser vacuum failure, (3) spurious main feedwater valve closure (see item 21 preceding) and recent incidents of failures in auxiliary feedwater systems it appears that, single failure criteria notwithstanding, at least short term failures of the auxiliary feedwater system must be considered to estimate the needed reliability of such system.

What, for instance, would be the peak primary system pressure, consequences to primary coolant system safety and relief valves and rate of primary coolant loss following failure of the Auxiliary feedwater pumps when needed?

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ASSIGNMENTS FOR ACRS QUESTIONS ON PEBBLE SPRINGS

Item	RSB	AP	MEB	RAB	EIC
1		X			X
2		X	X		
3	X	X			
4	X		X	X	
5	X				
6	X				
7		X	X		
8	X				
9	X	X			~~X~~
10				X	X
11		X		X	
12	X				
13		X			
14		X			
15		X			
16				X	
17					X
18		X			
19		X			
20	X				
21	X				X
22					X
23		X			
24	X				
25	X	X			
26	X X	X			

X = Prime Resp.
(X) = Secondary "

DPM

TR

RSB - Scott Newbury
AP - William LeFave
MEB - Kulin D. Desai
RAB - [Cimp... / Tom]
EIC - Hulbert Li

POOR ORIGINAL

Attachment 6

8/23/79

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

file copy

August 20, 1979

MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
Commissioner Ahearne

THRU: Lee V. Gossick
Executive Director for Operations

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: RESUMPTION OF LICENSING REVIEWS FOR NUCLEAR POWER PLANTS

In May of this year I described a realignment of current and near-term priority tasks within the Office of Nuclear Reactor Regulation (NRR) to deal with activities relating to the accident at Three Mile Island (see SECY-79-344). One consequence of the realignment was a temporary delay in the processing of operating license and construction permit applications for nuclear plants pending completion of certain TMI-2 related tasks.

The short-term TMI-2 tasks are essentially complete, as summarized below, and based on the results of these efforts I have decided to resume staff licensing activities on pending construction permit and operating license applications. It is my judgment that the TMI-2 related actions being taken by NRR on licensee emergency preparedness (see SECY-79-450), operator licensing (see SECY-79-33-E), bulletins and orders followup (primarily in the areas of auxiliary feedwater system reliability; loss of feedwater and small break loss-of-coolant accident analysis; emergency operating guidelines and procedures; and operator training), and short-term Lessons Learned, if accomplished generally on the schedule we have selected, are necessary and sufficient for the continued safe operation of operating plants and for the resumption of staff licensing activities on pending construction permit and operating license applications. It is my intent to bring the staff's first completed review of a pending operating license application to the Commission for review prior to staff issuance of the license. The Lessons Learned Task Force and I also have considered whether the actions associated with these activities would foreclose other actions that subsequently may be shown to be necessary by the Lessons Learned Task Force, the President's Commission or the NRC Special Inquiry. We have no indication that they will.

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The principal element of the composite of staff activities listed above is the completion of my review and the ACRS review of the first report of the TMI-2 Lessons Learned Task Force (NUREG-0578). The Task Force report contains a set of recommendations to be implemented in two stages over the next 16 months on operating plants, plants under construction, and pending construction permit applications. The Task Force recommended 20 licensing requirements and three rulemaking matters in 12 broad areas (nine in the area of design and analysis and three in the area of operations). All but one of the 23 recommendations had a majority concurrence by the Task Force. The Task Force concluded that implementing its recommendations would provide substantial, additional protection which is required for the public health and safety.

The Advisory Committee on Reactor Safeguards has completed its review of the Task Force report. The several public meetings of the ACRS subcommittee on TMI-2 and the public meeting of the full committee on August 9 provided an opportunity for the presentation and discussion of public comments on the report. The ACRS letter of August 13, 1979, to Chairman Hendrie states that the Committee agrees with the intent and substance of all the Task Force recommendations, except four upon which the Committee offered constructive comments to achieve the same objectives articulated by the Task Force. The Committee also noted that effective implementation will require a more flexible, perhaps extended, schedule than proposed by the Task Force. A copy of the ACRS letter is provided as Enclosure 1.

The ACRS comments on NUREG-0578 concentrate on four of the Task Force recommendations. These are: (a) the revision of limiting conditions of operation to require plant shutdown for certain human or procedural errors; (b) the inerting of MKI and II BWR containments; (c) the provision of recombiner capability at operating plants that do not already have it; and (d) the addition of a shift technical advisor at each operating plant. The first three of these matters require Commission rulemaking, and it is a straightforward matter for the staff to consider the comments in the process of developing the required Commission papers. I will assure that is done.

It is my intent to ask the Office of Standards Development (SD) to proceed expeditiously with a Commission paper proposing a new rule on limiting conditions of operation (item a, above). I will ask SD to include in the paper the alternative approach recommended by the ACRS, and one other approach that I think merits consideration. My alternative would amend the Task Force recommendation so as to differentiate between an isolated occurrence and a repetitive pattern. For example, the forced shutdown aspect of the Task Force recommendation could be reserved for a repeat violation within a relatively short time period, such as two years.

In the case of the two hydrogen control matters (items b and c, above), I intend to follow the advice of the ACRS by asking SD to delay completion of the required staff papers for proposed rulemaking until after receipt and review of the final report of the Lessons Learned Task Force, now scheduled for completion in mid-September. It is likely that the inerting and recombiner requirements recommended by the Task Force will be included in the eventual solution to the hydrogen control problems encountered in the TMI-2 accident. However, in view of the short time until the availability of the overall hydrogen control recommendations by the Task Force, I agree with the ACRS that it is best to not dilute staff effort in this area by prompt pursuit of the two short-term recommendations, one of which was a minority view of the Task Force for these same reasons.

The ACRS comments on the shift technical advisor (item d, above) have resulted in our reassessment of the possible means of achieving the two functions which the Task Force intended to provide by this requirement. The two functions are accident assessment and operating experience assessment by people onsite with engineering competence and certain other characteristics. I agree with the Task Force that the shift technical advisor concept is the preferable short-term method of supplying these functions. However, I have concluded that some flexibility in implementation may yield the desired results if there is management innovation by individual licensees. The Task Force has prepared a statement of functional characteristics for the shift technical advisor that will be used by the staff in the review of any alternatives proposed by licensees. It is provided here as Enclosure 2.

In addition to commenting on four of the Task Force recommendations, the ACRS letter of August 13 recommends three additional instrumentation requirements for short-term action. These are containment pressure, containment water level and containment hydrogen monitors designed to follow the course of an accident. I agree with these recommendations. The Task Force has prepared descriptions of these requirements in the same format as Appendix A of NUREG-0578. They are provided here in Enclosure 3.

I have also decided on one further licensing requirement for short-term action. It is a requirement for remotely operable high point venting of gas from the reactor coolant system. The Task Force has prepared a description of this requirement; it is provided here in Enclosure 4. The Task Force had previously deferred this item for further study, but it is my judgment that design efforts by licensees can and should be initiated now.

Finally, the Task Force has compiled a set of errata and clarifying comments for NUREG-0578. It is provided here as Enclosure 5.

In summary, the Task Force recommended prompt licensing action on 20 items (excluding the three rulemaking matters). I have added the three additional requirements recommended by the ACRS in its August 13 letter and one more on the basis of my own review. This Office will issue letters to all operating plant licensees and all construction permit and operating license applicants within the next two weeks requiring them to commit within 30 days to meet the total of 24 licensing requirements on the implementation schedule provided here in Enclosure 6. Another letter to be issued at approximately the same time will state the requirements flowing from the work by the Bulletins and Orders Task Force on operating plants which also need to be picked up on the license applications.

Several licensees have advised that some of the hardware changes required in NUREG-0578 can be accomplished at much lower cost during springtime refueling outages in 1980. For good cause shown, we intend to consider such flexibility in the implementation schedules. The end date for full implementation of all licensing requirements has not been changed from the January 1, 1981, date recommended by the Task Force. The implementation dates for the Commission rulemaking actions will be established in the course of rulemaking.

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. ACRS Ltr Carbon to Hendrie dtd 8/13/79
2. Alternatives to Shift Technical Advisors
3. Instrumentation to Monitor Containment Conditions
4. Installation of Remotely Operated High Point Vents in the Reactor Coolant System
5. NUREG-0578 Errata
6. Implementation of Requirements for Operating Plants and Plants in OL Review

cc: Mitchell Rogovin
Saul Levine
Robert Minogue
Victor Stello
William Dircks
Carlton Kammerer
ACRS