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

APR 10 1978

MEMORANDUM FOR: Frank Schroeder, Deputy Director
Division of Systems Safety

FROM: R. L. Tedesco, Assistant Director
for Plant Systems, DSS

SUBJECT: ALAB-444 WRITEUPS FOR CATEGORY A TASK ACTION PLANS

We have reviewed the following subject Task Action Plan writeups as requested by your note dated April 3, 1978: A-7, A-8, A-17, A-19, A-21, A-22, A-23, A-24, A-25, A-28, A-30, A-32, A-35, A-36 and A-39. We concur with the writeups if the changes indicated on the attached writeups are made.

Writeups attached: A-17, A-22, A-25 and A-39.

R. L. Tedesco, Assistant Director
for Plant Systems
Division of Systems Safety

cc: R. Mattson
W. Minners
V. Benaroya
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G. Lainas
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3. Basis for Continued Plant Operation and Licensing Pending Completion of Task.

As discussed in Section 1, this task addresses the development of a systematic process to review plant systems to determine their impact on other plant systems. The purpose of the task is to improve the licensing process. As discussed in Section 2 it is anticipated that this task will confirm that current licensing requirements and procedures acceptably control the potential for adverse systems interactions, even though some modifications for improvement in the review procedures and licensing requirements may be made.

Current licensing requirements are founded on the principle of defense in depth against credible occurrences. Adherence to this principle results in requirements such as physical separation and independence of redundant safety systems, and protection against events such as high energy line ruptures, missiles, high winds, flooding, seismic events, fires, operator errors, and sabotage. These design provisions supplemented by the current review procedures of the Standard Review Plan (NUREG-75/087) which require interdisciplinary reviews and which account, to a large extent, for review of potential systems interactions provide, for an adequately safe situation with respect to such interactions. The quality assurance program which is followed during the design,

construction, and operational phases for each plant is expected to provide added assurance against the potential for adverse systems interactions.

Plant licensing can continue pending ultimate resolution of this task because current licensing requirements provide an acceptable level of assurance against potentially adverse systems interactions. Previous licensing procedures that were followed for those plants now operating also provided assurance against potentially adverse systems interactions, although perhaps to a lesser degree than current procedures. ~~However,~~ Experience to date has demonstrated that operating plants have been designed to provide reasonable assurance that adverse systems interactions will not occur. However, in those instances such as fire protection and high-energy line break outside containment where the potential for adverse systems interaction has been identified, corrective measures have been or are being taken on each plant to attain an acceptable level of assurance against adverse systems interaction.

In summary, the staff considers that present plant design and review procedures which have been developed and refined from those procedures followed for plants now in operation, provide reasonable assurance that unacceptable adverse systems interactions will no occur. The results of this task are expected to confirm this view, although some modifications for improvement in the ~~to~~ review procedures and licensing requirements may be recommended. Accordingly, we conclude that while this task is being performed, continued operation and plant

licensing can proceed with reasonable assurance of protection to the health and safety of the public.

Task A-22

7.0 Basis for Continued Plant Operation and Licensing Pending Completion of Task

As indicated in Section 1, several aspects of the Main Steam Line Break analyses for PWRs as currently provided by applicants and reviewed by the NRC staff have been questioned. This task is to evaluate these questions or concerns and confirm or modify the present positions. The concerns derive principally from Issues 1 and 15 of NUREG-0138.

Issue 1 questioned credit for the operation of nonsafety-grade equipment as a backup for assumed single active failures in safety-grade equipment following a main steam line break. This task will evaluate plant response sensitivity to the operation or nonoperability of various nonsafety-grade systems and components. It will also develop a reliability assessment of such equipment.

why?

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Issue 15 is concerned with the mechanical response of the pressure vessel following a main steam line break. This task will consider safety systems and operator actions requested to (a) maintain acceptable pressure vessel stress levels, and (b) achieve long-term cooling.

In NUREG-0138, the staff provided the bases for its interim positions on these two issues. For Issue 1, the staff stated that breaks in the

secondary system are considered to have less potential for major release of fission products than breaks in the primary system. (Typically, doses calculated for main steamline break accidents are a small fraction of 10 CFR Part 100 limits.) Consequently, less stringent requirements are imposed on the quality and design of systems required to cope with secondary system ruptures. This approach, in the staff's judgment, results in a proper weighting of consequences and safety requirements in order to assure a balanced level of safety over the entire spectrum of postulated design basis accidents.

The bulk of the components in the secondary system are essential to plant operation or availability, and are in a state of continuous or frequent operation. The considerable experience from both fossil and nuclear plant operation has ~~demonstrated~~ indicated a high reliability of such components. Awareness of this reliability level led to the position of permitting credit for nonsafety-grade equipment even though documented evidence for all systems and components was not available. This [illegible] effort is expected to confirm this judgment through the sensitivity and reliability studies referred to above.

Relative to issue 15b, we noted that a potential safety problem related to reactor vessel integrity does not become important until the vessel has been subjected to extended neutron irradiation during plant operation. The irradiation effect is to reduce the allowable stress at reduced

temperatures late in the life of the vessel. There must also be a potential for return to high pressure at reduced vessel temperature. This task will provide a conclusion regarding this latter aspect (discussed below) and will provide input to Task A-11 which includes evaluation of the effects of a steamline break on vessel integrity.

When considering the sequence of conditions following a main steam line break, the primary system is first depressurized by overcooling through the secondary system. The reduction in primary system pressure causes a reactor trip and actuation of the emergency core cooling system (ECCS). Pressure reductions in the primary system are accompanied by temperature decrease with shrinkage of the liquid volume. Actuation of the ECCS replenishes the volume of liquid. Unless terminated or controlled by the operator, the ECCS could eventually refill and repressurize the primary system to the safety valve set point. This task will evaluate the timing requirements for operator actions, the nature of the actions, and the likelihood of accomplishment and develop a position relative to the requirements for such operator actions.

The interim positions described above with regard to credit for nonsafety-grade equipment are continuing to be used in the review of construction permit and operating license applications and in evaluating the continued operation of operating reactors and, for the reasons stated above, we have

concluded that operation does not present an undue risk to the health and safety of the public. The other issue addresses the need for operator action following a main steamline break late in reactor life to assure vessel integrity. The information in this regard derived from this task will provide input to Task A-11 which specifically addresses reactor vessel integrity.

TASK A-25

NON-SAFETY LOADS ON CLASS IE POWER SOURCES

3. <u>Basis for Continued Plant Operation and Licensing Pending Completion of Task.</u>

As discussed in Section 1, the safety concern addressed by this task is whether or not the reliability of Class IE power sources is significantly affected by allowing the sharing of these sources by loads that perform safety functions and loads that perform normal plant functions (non-safety loads). The Class IE power sources are those on-site sources that are called upon only in the event of a loss of the preferred off-site power.

Present regulatory practice allows the connection of non-safety loads in addition to the required safety loads to Class IE power sources by imposing some restrictions. The results of this task will be used by the regulatory staff in evaluating construction permit applications submitted after 1978.

It is anticipated that the results of this task will not significantly alter current licensing criteria set forth in Section 8.3.1 of the NRR ~~Commission's~~ Standard Review Plan. In fact, this task may yield relaxed licensing requirements that would permit circuit breakers and fuses as isolation devices, under certain conditions, between non-safety loads and Class IE power sources.

Prior to the completion of the task, current licensing requirements set forth in the Commission's Standard Review Plan will be

utilized to assure safe plant operation. These requirements include the automatic disconnection of non-safety loads from Class IE buses at the onset of emergency conditions such that the non-safety loads do not affect the ability of the Class IE power sources to supply the required safety loads.

By assessing and quantifying the reliability assured by current regulatory requirements, this task will establish firm bases for connecting non-safety loads to Class IE sources without degrading the emergency sources below an acceptable level. This task will also determine whether some of the current licensing criteria can be relaxed and still provide adequate protection of public health and safety.

In summary, it is the staff's judgment that current licensing requirements set forth in the NRR ~~Commission's~~ Standard Review Plan Provide ~~which~~ Sufficient assurance ~~require an assessment~~ of the reliability of Class IE power sources ~~are sufficient~~ to continue with plant licensing or operation pending the ultimate resolution of the issue addressed by the task action plan.

TASK A-39

DETERMINATION OF SAFETY REFIEF VALVE (SRV) POOL DYNAMIC LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT

3. Basis for Continued Plant Operation and Licensing Pending Completion of Task.

As discussed in Section 1, the safety concern addressed by this task is the possible damage to wetwell internal structures and the pool boundary that could occur due to air-clearing and steam quenching phenomena resulting from safety relief valve (SRV) discharge into the suppression pools of BWR plants. It is of concern to all BWR plants using the Mark I, Mark II, or Mark III pressure suppression type containments.

As discussed in Section 2, this task will provide the basis for establishing acceptance criteria for safety relief valve loads and for suppression pool temperature limits. In conjunction with Task A-7 (Mark I Long Term Program) and Task A-8 (Mark II Containment Pool Dynamic Loads), a complete evaluation will be provided of suppression pool dynamic loads for BWR containments.

For plants with Mark I containments either in operation or not yet licensed for operation, the justification for continued operation and licensing is based on our evaluation of operating experiences which demonstrates ~~and~~ the plant capability to tolerate SRV loads in the short term. SRV operating experience has shown that in all but a few instances,

SRV discharges have performed satisfactorily without any evidence of damage either due to the hydrodynamic loads or pool temperature effect. In those isolated cases where localized damage has been encountered, the damage did not result in a loss of the containment function, or release of radio-activity, or undue risk to the health and safety of the public. In those cases, repairs were made and additional margin was included in the structures. With respect to the plant capability, the staff has concluded that the plants have the capability to tolerate SRV loads because the loads are related to the structural fatigue life. However, all plants will be required to demonstrate the capability to meet the SRV loads criteria and pool temperature limit which will be established by this task.

Plants with Mark II containments, of which there are none in operation, will be required to demonstrate the capability of accommodating the criteria developed under this task program. The lead plant for an operating license is the Zimmer Plant, which has a projected fuel loading date of October, 1978. The completion date for the interim criteria is June, 1978. We will require the Zimmer Plant to meet the interim criteria before an operating license can be issued.

For Mark III containments, we have issued acceptance criteria for SRV with quencher device. Although we believe that the loads criteria are conservative, we will require in-plant tests for confirmation. It should be noted that the criteria were established

on the basis of information previously provided by GE. However, recent GE studies indicate that there are some changes in the previously provided information. These changes would affect the load criteria. GE, however, has proposed an approach to modify the SRV control logic such that the current load criteria can be maintained. We have included this concern in a revision to this task action plan and have actively reviewed the GE proposed approach. Although we have not completed the review, we believe that such an approach is technically feasible. Since all Mark III containments use quencher devices, the pool temperature limits will not be an area of concern on the basis of current Mark III design.

In summary, the staff concludes that the SRV loads are related to the structural fatigue life. Therefore, we feel that the plants with Mark I containment can be allowed to continue operation until completion of the Mark I Long Term program. Interim criteria will be developed for use on Mark II containments prior to the issuance of an operating license for the first U. S. plant (Zimmer) to use such a containment. For Mark III containment, we believe that the current criteria are adequately conservative if the GE proposed approach is found acceptable as a result of our evaluation. Accordingly, we conclude that while this task is being performed, continued operation and plant licensing can proceed with reasonable assurance of protection to the health and safety of the public.