July 11, 1988

Docket No. 50-416

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LICENSEE: System Energy Resources, Inc. (SERI)

FACILITY: Grand Gulf Nuclear Station (GGNS), Unit 1

SUBJECT: SUMMARY OF APRIL 25-29, 1988 AUDIT OF 10 CFR 50.59 SAFETY EVALUATIONS

Introduction

The purpose of this audit was to determine the adequacy of safety evaluations performed pursuant to 10 CFR 50.59. During the audit, SERI procedures and guidelines for performing these safety evaluations were reviewed and interviews were held with the licensee's personnel who performed and approved the safety evaluations. In addition, minutes of plant safety review committee (PSRC) meetings were reviewed and a PSRC meeting to consider current safety evaluations was observed.

The audit consisted of selecting 20 safety evaluations from the summaries of safety evaluations made during the interval June 1, 1986 to May 31, 1987 (submitted to the NRC May 29, 1987 and November 27, 1987). Of these 20 safety evaluations, six were selected for an indepth review of the 10 CFR 50.59 determinations including background material and interviews with the evaluators. The other 14 safety evaluations received a cursory review.

An entrance meeting was held on April 25, 1988 in the SERI Corporate Office, Jackson, Mississippi. Present for the licensee were J. G. Cesare, W. Eiff and J. Summers. An exit meeting was held on April 29, 1988 at GGNS. Attendees at the exit meeting are given in Enclosure 1.

Criteria for Audit

In the entrance meeting, the NRC Project Manager described the scope of the audit and criteria to be used in the audit. The Inspection and Enforcement Manual Part 9800 was used as guidance in the application of 10 CFR 50.59 rules. Answers to four questions were sought during the audit:

1. Does the safety evaluation clearly identify the function of the equipment or procedure that was changed and the effect of the change on that function?

2. Does the safety evaluation identify specifically or by reference to the FSAR or other licensing document the previously evaluated accident or malfunction of equipment affected by the change?

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3. Does the evaluation identify the margin of safety for the equipment or procedure and the effect of the change on the margin of safety?

4. Are the criteria of 10 CFR 50.59(a)(2) met in the determination that there is no unreviewed safety question?

In the evaluation of Question 4, the staff assumed that the safety analysis report referred to in 10 CFR 50.59 was the Updated Final Safety Analysis Report (UFSAR) which should include letters filed during the operating license review. Also, since the staff sometimes performed independent analyses (such as dose consequences for accidents) and conditionally accepted some design criteria on the basis of letters of commitment, the staff's Safety Evaluation Report and Supplements, were considered, together with the FSAR in determining whether a change constitutes an unreviewed safety question. In general, a 10 CFR 50.59 safety evaluation was considered to be required for a change from the description of the facility and procedures which was filed in the operating license application and was either accepted by the staff in the SER or was available for the staff's review during the operating license phase.

In using the criteria of 10 CFR 50.59(a)(2) in the audit, any increase in the probability of occurrence of an accident or any increase in the consequences of an accident or any reduction in the margin of safety as defined in the basis for any Technical Specification is considered by the staff to be an unreviewed safety question. The margin of safety was considered to be the difference between the value of a parameter at the component failure point and the peak value calculated in bounding safety analyses for a particular accident and reported in the UFSAR.

SERI Procedures

During the audit, the licensee provided and described SERI procedures for performing 10 CFR 50.59 safety evaluations, provided and described an ongoing training program for personnel performing 10 CFR 50.59 reviews and described 10 CFR 50.59 program improvements (Enclosure 2).

The procedures provided are listed in the Reference section of this report. Forms used to perform applicability reviews (or screening) for safety evaluations required by 10 CFR 50.59 and environmental evaluations required by Appendix B of the Operating License (Environmental Protection Plan) are given in tables at the end of this report.

Reference (1) is a policy statement from which the other departmental procedures are derived. Each procedure requires an applicability review or screening to determine if a 10 CFR 50.59 safety evaluation or an environmental evaluation is required. The safety evaluation flow chart (Table 1) is similar to that in I&E Manual Part 9800 (Table 2) with two exceptions:

1. The SERI procedure does not require all changes to be reviewed by the PSRC whereas the I&E Manual Part 9800 states that most technical specifications require the Onsite Review Group to review all proposed changes to the facility, all proposed changes to procedures, and all proposed tests and experiments. In a check of GGNS TS, it was found that TS Section 6.0

requires PSRC to review all procedures, but not all proposed changes to the facility or all proposed tests and experiments. The BWR Standard Technical Specifications (NUREG 0123), like the I&E Manual Part 9800, require that all proposed changes to equipment and procedures and all proposed tests be reviewed by the Onsite Review Group.

2. In the applicability review or screening process to determine if a safety evaluation is required (Table 3), the questions to be answered yes or no, are whether the proposed change makes changes in the facility or procedures as described in the FSAR or involves tests not described in the FSAR. The I&E Manual Part 9800 flow chart (Table 2) has an additional question - whether the proposed change could affect nuclear safety in a way not previously evaluated in the FSAR.

The narrow interpretation in the SERI procedures is reflected in the applicability review and screening forms used by the SERI departments. However, the Nuclear Plant Engineering (NPE) Department performs a safety evaluation for all proposed equipment changes. The large majority of changes to equipment are initiated by NPE. The Nuclear Licensing Department later screens these NPE safety evaluations to see which ones will be included in the 10 CFR 50.59 files and summarized for the annual report to the NRC. The other departments, however, do not perform a safety evaluation unless it is found to require a 10 CFR 50.59 safety evaluation in the screening process.

If the equipment or procedure to be changed is not described in the FSAR, a safety evaluation is not deemed to be needed. This is a weakness of the applicability review or screening process in that only changes to equipment and procedures explicitly described in the updated Final Safety Analysis Report are considered to require a safety evaluation. The policy statement defines "described" to mean "inclusion by explicit statement or by reference in the textual portions of a document or by depiction on drawings contained in a document." It would be reasonable to expect a safety analysis to be made for all changes to the facility and procedures whether or not they are filed as 10 CFR 50.59 evaluations. Most Technical Specifications require the PSRC (or equivalent) to review all changes.

The screening form also asks the questions as to whether a TS change is required. In this regard, the policy statement and procedures state that design change packages requiring an addition to TS are not considered to require a change in TS. Therefore, such changes may be made and placed into service prior to NRC approval of a TS change request, provided a 50.59 safety evaluation is completed. The licensee was advised that this is an incorrect definition of a "change in TS" because 10 CFR 50.59 requires prior Commission approval of changes to the facility if they involve a "change in TS."

The policy statement, Reference 1, also includes procedures for determining whether a change involves an unreviewed environmental question or a change in the Environmental Protection Plan as required by Appendix B to the Operating License. A flow chart illustrating the process is reproduced in Table 4. It is noted that the question is asked as to whether the proposed change, test or experiment "will or may affect the environment," which is a broad enough question that should identify all applicable environmental related changes.

However, a similar broad question regarding nuclear safety (e.g., could the proposal affect nuclear safety?) is not asked in the screening process (see Table 3). Therefore, changes that should receive a safety evaluation may not, but may receive an environmental evaluation. This is particularly relevant to changes in radiation monitoring of gaseous and liquid effluents and or radiological environmental effects (Part 20 limits). An example of this is discussed further in the next section of this report which discusses the review of specific evaluations.

The procedures do not require meaningful qualifications for the applicability reviewer or the originator of safety and environmental evaluations. Except for NPE, each individual departmental procedure states the reviewer or evaluator may be anyone who is familiar with 10 CFR 50.59, GGNS TS, the GGNS FSAR, and the SERI procedure. Each reviewer must initial a qualification sheet indicating they are familiar with these documents. The qualification sheet is signed by the department director. The NPE department requires screening forms to be signed by the assigned responsible Engineer, Group Supervisor and the Principal Engineer (Table 6). The NPE department requires safety evaluations to be signed by the same three positions, and to be reviewed, approved and signed by the Responsible Manager. The Plant Operations department has an additional qualification that an originator or approver of safety evaluations or screenings shall be cognizant of the subject of the safety evaluation.

The SERI policy statement and implementing procedures in each department include guidelines for performing the 10 CFR 50.59 safety evaluations, consisting of questions to focus the reviewer's attention on some change which could result in a requirement to change TS or an unreviewed safety question. It is noted that the reviewer is instructed to review FSAR Section 1.2, which contains generalized descriptions of the plant and plant drawings. For consideration of an increase in probability of an accident, the reviewer is referred to FSAR Section 15.0.3.1 "Identification of Causes and Frequency Classification," which defines four categories of events of varying frequencies - normal operations, incidents of moderate frequency, infre incidents, and limiting faults. For consideration of consequences, the reviewer is referred to Chapter 15 of the FSAR.

The current training program at SERI includes additional guidance regarding the determination of an increase in probability or consequences of accidents and of a reduction in margin of safety. A change from one of the four frequency classes to another or a significant increase in the probability of occurrence of the accident is defined as an increase in the probability of an accident. This is contrary to 10 CFR 50.59 wherein <u>any</u> increase would involve an unreviewed safety question. The safety margin refers to the margin between the component failure point and the safety limits (e.g., the safety limit for peak cladding temperature during a LOCA is 2200°F per 10 CFR 50.46). An equipment change which results in an increase in peak cladding temperature for the bounding LOCA analysis is not considered a reduction in margin unless 2200°F is exceeded. This is contrary to 10 CFR 50.59 where the margin of safety refers to the difference between the failure point of a component and the bounding value for previously analyzed accidents (e.g., for the LOCA, the

bounding peak cladding temperature is 2098°F). A "small" increase in consequences can be justified by engineering judgment to constitute no increase in consequences, which is contrary to 10 CFR 50.59 which states any increase in consequences involves an unreviewed safety question. The reviewer is guided to the "Basis for Technical Specifications" to evaluate the margin of safety, but the basis for a TS may not be discussed in the Bases section of the TS either qualitatively or quantitatively resulting in the reviewer's conclusion that there is no reduction in a margin of safety "as defined in the basis for any technical specification." In its comment on margin of safety, Reference (7), the staff makes this point and states that other licensing basis documents (SER, UFSAR) should be considered if the margin of safety is not discussed in Bases to TS. Many of the parameters considered to be consequences in the SERI guidelines are really parameters used to measure margins of safety (Table 7).

The guidelines used in the training program are less acceptable than the present procedures. These guidelines are based on the NUMARC draft 10 CFR 50.59 guidance document, which has been recently commented on by the NRC (Reference 7). Many of the comments noted above are similar to NRC comments in Reference 7.

REVIEW OF SELECTED SAFETY EVALUATIONS

Twenty safety evaluations were selected for the audit based on a prior reading of the summaries submitted to the staff for the period June 1, 1986 to May 31, 1987. More than 200 safety evaluations were reported during this interval, including about 50 evaluations of changes to the UFSAR. The selection was biased in favor of those changes which appeared to be significant with respect to safety and could involve an unreviewed safety question. The quality of the summaries varied from excellent descriptions of the change, reasons for the change, and the effect on safety (e.g., PLS-87-001, PLS-87-002, and NPE-87-113) to very poor (e.g., PLS-87-004, NPE-87-105, and NPE-87-179). In reviewing the summaries, it was not clear why some were reported as 10 CFR 50.59 safety evaluations. For example, NPE-86-218 concerned test plugs installed in cooling tower distribution pipes; NPE-86-219 concerned temporary removal of drift eliminators from the cooling towers (this is perhaps an environmental matter); NPE-87-019 concerned installation of a one-inch sample tap in the plant service water system; and NPE-87-023 concerned temporary installation of a public address desktop station in the drywell airlock.

Of these, 20 selected safety evaluations, six were reviewed in depth and 14 received a cursory review.

In addition, an environmental evaluation and three evaluations determined to be not reportable per 10 CFR 50.59 requirements were reviewed. These evaluations were selected by the licensee from its files at the PM's request.

Cursory Review

Of the 14 safety evaluations that received a cursory review, half of them had a satisfactory description of the function of the equipment being changed and the effect of the change on the function of that equipment or interfacing

equipment. Less than half identified the accident or anticipated operational occurrence affected by the change. Only one of the 14 identified the margin of safety relevant to the change.

An example of a safety evaluation not having a satisfactory description of the change is NPE-87-106 which stated that "recommendations 14 and 15, concerning the Power Driven Potentiometers" in the DR/QR Report were not required to be implemented. A TDI diesel generator DR/QR Report section was referenced and NPFI 86/00926 was attached as giving an evaluation, but the safety evaluation did not describe the function of the potentiometers, what the TDI Diesel Generator Owners Group recommendations were, and what the effect of not following them would be. In addition, no mention was made of the License Condition 2.C(25)(b) which gives TDI emergency diesel generator requirements and references the staff's Safety Evaluation Report of TDI diesel generators.

Other examples of inadequate description of the effect of the changes on accident probabilities, consequences or margins of safety are NPE-87-270, NPE-87-273 and NPE-87-277 with regard to extensive FSAR changes. The changes are identified by references, but the effects of the changes are not described. NPE-87-270 identifies modifications to feedwater pipe break restraints, revised pipe stress results, piping changes to reflect as-built conditions, changed locations of pipe break restraints, changed break locations of RWCU system, and added evaluation of high energy line breaks; however, the effect of revised pipe stress results on margin of safety is not discussed. NPE-87-273 describes changes to the FSAR description of a fuel handling accident inside containment to correspond to the FSAR analyses of that accident, with the equipment hatch open, but the effect on offsite dose consequences is not discussed. NPE-87-277 describes changes to UFSAR Section 3.10 to add seismic qualification criteria for replacement parts, but the effect of these new criteria on safety related equipment is not discussed.

Another example of an inadequate safety evaluation writeup is NPE-87-253 concerning degraded performance of engineered safety feature room coolers. Minimum design flows less than the nominal design flow in the FSAR were calculated as a new criterion for acceptable performance because nominal design flow could not be achieved. The basis given for no reduction in the margin of safety was that the minimum flows would maintain room temperatures within Technical Specification limits. However, it appears obvious that the margin of safety was reduced because design flows could not be achieved.

Another example of an inadequate writeup is NPE-87-295. An FSAR criterion stating that "RPS cabling was run separately from all other wiring" was changed to permit RPS cabling to be run with ESF wiring of the same power division. Although a GE analysis was attached, the safety evaluation writeup did not clearly demonstrate that the criteria of 10 CFR 50.59 were met for the changes not to be considered an unreviewed safety question.

The environmental evaluation, 031/87, described the change as the removal of the high radiation trip of the radwaste building exhaust fans, leaving the alarm intact. The evaluation also included a brief safety evaluation, but it was reported as an environmental evaluation and attached to the licensees latest Annual Environmental Report. The evaluations addressed the loss of the

ventilation system by relying on the staff's SER statement that the loss will not compromise the safety functions of essential systems or result in release of unacceptable amounts of radioactivity. The evaluations did not address the change which was deleting the capability to shutoff the fan automatically and instead rely on the operator seeing the alarm, following alarm instructions and having access to the fan control if high radiation were detected being discharged through the radiation building exhaust duct. Considerations of as low as reasonably achievable and Appendix I design objective guidelines as contained in the Technical Specifications, Appendix A, were not addressed. The reason for the change was not given. It is not clear why this was not reported as a safety evaluation.

Three safety evaluations determined to be "not reportable" were reviewed (NPE-87-058, NPE-87-198, and PLS-87-006). PLS-87-006 is a further evaluation of the installation of temporary air conditioning units in the computer room, reported in NPE-87-040, and it is not clear why PLS-87-006 should not have been reported together with NPE-87-040. The other two safety evaluations in this group contain changes at a level of detail not considered in the FSAR. NPE-87-198 concerned a change to panel internal wiring and grounding and NPE-87-058 concerned a brazing compound different from that recommended by the vendor for replacement of condenser coils in the computer room air conditioners. The staff agrees that these two safety evaluations are "not reportable."

Indepth Review

Each of the six indepth reviews was conducted by obtaining and reviewing material referenced in the safety evaluation, interviewing the evaluator and an approver and reviewing written answers to questions raised during the audit. The six safety evaluations were selected for an indepth review because the determination that the change was not an unreviewed safety question appeared questionable based on a reading of the 20 safety evaluations. It is not surprising, therefore, that it was found that five of the six are indeed questionable and need further review by staff technical personnel. The safety evaluations and referenced material, when supplemented by verbal and written responses to questions, provided an adequate understanding of the change. The licensee provided personnel and information requested in a timely manner.

NPE-86-241

This safety evaluation concerns the disposition of a materials nonconformance report (MNCR 1136-86-5th) for the safety grade backup air coolers in the fuel pool cooling and cleanup (FPCCU) pump room when the normal auxiliary building ventilation system fails. Attempts were made to increase air flow and water flow of the coolers to maintain the pump room at or below 104°F. When this air temperature could not be achieved, the coolers were accepted with the maximum attainable air flow and water flow, resulting in a calculated FPCCU pump room air temperature of 107°F when the normal auxiliary building ventilation system is lost. Technical Specification (TS) 3/4.7.8 requires the auxiliary building air temperature to be equal to or less than 104°F.

The safety evaluation concludes that no TS change is required on the basis that the Bases for TS 3/4 7.8 states "the area temperature limitations ensure that safety related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures." The safety evaluation states that since the room contains no equipment subject to 10 CFR 50.49, the TS does not apply to the FPCCU pump room. During the audit, NPE personnel stated further that the TS temperature limit of 104°F is for normal operation and the calculated 107°F is based on accident heat loads, presumably heat loads following a LOCA. The safety analysis states only that the 107°F temperature would result following loss of the auxiliary building ventilation system. The staff does not agree with the licensee's conclusion that the TS does not apply to the FPCCU pump room because it does not contain environmentally qualified equipment. The staff does agree that the TS does not apply to conditions following a LOCA; however, it does apply to normal conditions, including anticipated operational occurrences such as loss of the auxiliary building ventilation system. If the 107°F is calculated for a loss of auxiliary building ventilation systems, as stated in the safety evaluation, then the determination that no TS change is required is questionable.

The safety evaluation clearly identifies the change in equipment performance and the effect of the change on safety significant equipment. The safety evaluation does not adequately identify the previously evaluated accidents that are relevant to this change. The safety evaluation states that Class 1E electrical equipment in the room has been shown to operate in temperatures up to 104°F with a 10% margin and uses this as the basis for concluding there is no increase in the probability of occurrence or consequences of an accident and no increase in the probability of occurrence or consequences of a malfunction of equipment important to safety. The basis for the licensee's conclusion that there is no reduction in the margin of safety is that there are no environmentally qualified components in this room and the safety function of other safety related equipment will not be impacted by the higher air temperature. The staff does not agree with these conclusions because the design margin of 10% for Class 1E electrical equipment is clearly reduced by operation at 107°F instead of the design temperature of 104°F. Operation at a higher temperature would decrease the reliability of these components and increase the probability of their malfunction during an accident. Therefore, the conclusion that the change does not involve an unreviewed safety question is questionable.

NPE-86-279

This safety evaluation concerns the installation and operation of a new crane installed inside the containment to be used during outages to handle some of the loads formerly handled by the polar crane, including safety relief valves, control rod drives and LPRM casks. The crane can operate over the spent fuel in the upper containment pool and over the reactor.

The safety evaluation and attached discussions clearly identified the function of the new crane and the previously evaluated accidents affected by the installation and operation of the crane. The safety evaluation did not describe the margin of safety for handling heavy loads with the new crane, compared with handling the same loads with the polar crane (e.g., crane,

maximum load, brakes, speed). The basis for stating there was no reduction in the margin of safety is that the TS load limit of 1140 pounds for handling loads over spent fuel would be applied to the new crane as well as the polar crane. Response to questions indicated that loads up to 8400 pounds could be handled and that some structural modifications to the crane were made and it was downrated from its standard load chart.

The safety evaluation referenced the licensee's response to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" but did not reference the staff's Supplemental Safety Evaluation Report No. 5 (SSER 5) which gave staff's evaluation of that response and identified approved characteristics and procedures for the polar crane and other cranes in the vicinity of safe shutdown equipment. In response to questions during the audit, the licensee provided a summary of the degree of conformance of the containment hatchway crane to NUREG-0612 criteria. The UFSAR contains a general reference to the licensee's response to NUREG-0612 (Section 9.1.4.3); and so it was not revised when the new crane was installed. However, the licensee was deficient in (1) not including the response to NUREG-0612 in the initial UFSAR and (2) not revising the response when the new crane was installed. NUREG-0612, Section 5.1.1(2) requires load handling procedures to be developed meeting certain criteria. Such a procedure has been developed and used for the polar crane (07-S-05-316), but not for the new crane. In response to questions, the licensee said a general procedure used for all other cranes (07-S-05-300) is used for operation of the containment hatch crane; however, this procedure does not require NUREG-0612, Section 5.1.1(2) to be met. Operation of the new crane without such a procedure appears to be a deficiency.

The conclusion that there is no unreviewed safety question involvement is questionable because the probability of a dropped load may be increased due to the lack of a load handling procedure meeting the criteria of NUREG-0612. Further, there may be a reduction in the margin of safety in handling heavy loads since the design requirements of ANSI B30.2-1976 and CMAA-70, as recommended in NUREG-0612, do not apply to the containment hatch crane.

NLS-87-003

This safety evaluation concerns a change in the UFSAR, Section 7.3, to manual actuation criteria for the hydrogen recombiners. The criteria for manually starting the hydrogen recombiner in Revision 0 of the UFSAR were (1) when the hydrogen concentration reached 3.5 volume percent and (2) when the reactor water level reached the top of the active fuel. In Revision 2 of the UFSAR, the second criterion was deleted. UFSAR Section 6.2.5.2, had only the first criterion and was not changed. The safety evaluation identifies the change and the accident involved - the LOCA with core uncovery and hydrogen generation. However, the effect of the change is not discussed, other than to imply that the change was non-conservative. The margin of safety is considered not to be reduced because the change does not affect the Technical Specifications or plant operation. The reliance on hydrogen concentration as the sole parameter to start the recombiners appears to increase the probability of a LOCA in which hydrogen burn rather than hydrogen recombination, occurs. Hydrogen concentration measurements may not be as reliable as water level measurements. The UFSAR and emergency procedures utilize both hydrogen concentration and

water level for turning igniters on. This was discussed in the staff's meeting with the Hydrogen Control Owner's Group, of which the licensee was a participant on December 18, 1985 (Meeting Summary dated January 17, 1986). Because the change may increase the probability for a LOCA without benefit of hydrogen recombination, it appears the change involves an unreviewed safety question.

PLS-86-123

This safety evaluation concerns the installation of a jumper in the actuation circuitry for the standby gas treatment system (SGTS) to bypass relays which were designed to stop SGTS fans on a high-high temperature (310°F) in the filter train charcoal. The relays were bypassed because they were not environmentally qualified. The alarm in the control room was left operable so the operator could have the fans manually stopped. Alarm response instructions require the fans to be manually stopped by opening breakers in the auxiliary building.

The safety evaluation clearly identifies the function of the equipment before and after the change. However, the effect of the change on probability and consequences of an accident was not adequately discussed. Substitution of a manual trip of the fans for an automatic trip may increase offsite dose consequences following a LOCA or radiation release in the primary or secondary containment because of the longer time required to shut off the fans. Therefore, it appears the change involves an unreviewed safety question.

PLS-86-132

This safety evaluation concerns removal of snubbers from operable systems for functional testing, provided the snubbers are replaced or reinstalled within 72 hours in accordance with the Action Statement in TS 3.7.4.

This safety evaluation and the referenced engineering evaluation clearly described the change and its effect on accident probabilities and consequences and on the margin of safety. Snubbers are designed to protect piping systems during dynamic events (e.g., earthquake, flow transients, relief valve discharges, and suppression pool dynamic loads during a LOCA). The design basis of piping systems is stated to be faulted load conditions at design pressure and temperature. Functional testing is done under reactor shutdown conditions.

Because the temperature and pressure loading is less during shutdown, the margin of safety is not reduced when a snubber is removed for testing. If a transient such as a pump start in a partially filled system were to occur, the affected piping would be analyzed and inspected and appropriate corrective action taken before returning it to service. Restrictions were imposed on removing adjacent snubbers in a system. Removal of a snubber for testing during shutdown has no effect on the probability of an accident occurring or on its consequences because the margin of safety to pipe failure is not reduced and there is no high energy piping. Further, dynamic loads primarily occur during plant operation. The PM agreed with the conclusion of this safety evaluation that an unreviewed safety question is not involved in the removal of snubbers for testing in the manner described.

PLS-86-136

This safety evaluation concerns the deletion of a sentence in the UFSAR Section 15.7.6.1.1 which states that during fuel handling operations within the containment, procedures require the equipment hatch and at least one door in each personnel lock to be closed. The UFSAR Section 15.7.6.2.2.1 gives the results of an analysis of offsite dose consequences for a dropped fuel assembly within containment with the equipment hatch open.

The safety evaluation was well prepared, and the staff may find fuel handling in the primary containment with the equipment hatch and personnel locks open is acceptable. However, deletion of the FSAR statement appears to be an unreviewed safety question. The staff's offsite dose consequences for a dropped fuel assembly within containment are based on the primary containment being closed (SER Section 15.3.3). The licensee's safety evaluation for this change follows its procedures for 50.59 evaluations by considering only what is explicitly stated in the UFSAR. However, the UFSAR change would increase the consequences of an accident, based on the staff's analysis. Therefore, the deletion of the UFSAR statement appears to involve an unreviewed safety question.

Plant Safety Review Committee Approval

A Plant Safety Review Committee (PSRC) meeting was held on April 27, 1988 to consider approval of four 10 CFR 50.59 safety evaluations (SE). The SE originators and supervisors were present in addition to the PSRC members. The NRC Senior Resident Inspector and Project Manager attended the meeting as observers. Copies of the four SEs were received and reviewed by the Project Manager prior to the meeting. Questions on the four SEs were noted by the Project Manager before the meeting, but not made available to SERI. All significant questions were raised by the PSRC and appropriately responded to by SERI personnel in the meeting.

One weakness in the approval procedure was that the written SE did not provide adequate information to document the equipment safety function, the effect of the change on safety functions, and the basis for the determination that there was no unreviewed safety question, without the supplemental verbal response to PSRC questions. Based on a review of PSRC meeting minutes, approval or rejection of the SEs are noted, together with action items (e.g., make a determination if an incident report is needed), but significant supplemental information is not documented. For example, two of the SEs concerned errors in drawings. In one SE (CFRMISC0054R00), an adequate basis was given for the determination that the installation cable to a breaker was correct because it was stated that design calculations were reviewed and it was found that the installed cable had been used in cable sizing calculations and circuit voltage drop calculations. The calculations were referenced in the SE. In the other SE regarding a drawing error (CFRMISC0056R00), the SE only stated that the Division I and II diesel generator drawings show incorrect circuit connections for the air start solenoid valves. Responses to PSRC questions brought out the basis for the conclusion - the installed connections were in accordance with vendor drawings based on an inspection of the system. However, both SEs were

approved by the PSRC. Sending the latter one back for additional information would have provided good training for the evaluator and a good record in the safety evaluation.

Other examples of inadequate documentation were noted in the other two SEs. In CFRMISC0055R00, the SE description simply states "that a line closed by a blind flange is attached to each of three valves (F001, F004A and F004B) at the valve body drain hole, but the P&IDs and FSAR figures do not show these flanged lines. PSRC questions brought out the following relevant significant information: the flange is bolted; it was designed for local leak rate testing of the valves but is not used; it is a part of the primary containment boundary which is tested in the Type A tests; they do not need to be included in TS but will be included in the licensee's controlled document showing all components of the primary containment boundary (AECM-85/0137). This safety evaluation was not approved pending action to determine if inclusion of these flanged lines in AECM-85/0137 is acceptable and whether an incident report is needed. In the other safety evaluation (FSAR 87/0091R00), it was proposed to change UFSAR Section 9.3.1.2 to add a requirement that filters for air supplied to instruments should remove particles larger than 50 microns and to delete the requirement for pressure regulators for each instrument or group of instruments. The change was based on a GE specification that all instruments except the ADS can be supplied with a minimum air pressure of 100 psig and that a final filter of 50 microns is required. This proposal was rejected on the basis that a determination should be made as to whether this should be added to Technical Specifications. The safety evaluation simply concluded that no TS change was needed because filters and regulators were not addressed in the GGNS TS.

SUMMARY

The scope of the audit included an indepth review of six safety evaluations and a cursory review of 14 additional safety evaluations (SE). These SEs were selected from the summaries of SEs submitted to the NRC for the period June 1, 1986 to May 31, 1988. Response to questions regarding the SEs and background information was provided by the SEs originators and other SERI personnel. The SERI procedures for 10 CFR 50.59 reviews and the recently developed program improvement and training program were also reviewed and discussed. In addition, a PSRC meeting to consider four SEs was observed.

Regarding procedures for SEs, there are differences between the interpretation of 10 CFR 50.59 in the SERI procedures and the guidance in NRC Part 9800 of the I&E Manual and other NRC guidance as described below.

- SERI procedures and GGNS Technical Specifications do not require that all changes to the facility be reviewed by the Plant Safety Review Committee. The Standard Technical Specifications and Part 9800 recommend this. It is noted, however, that Nuclear Plant Engineering does perform a safety evaluation for all design changes it initiates. These SEs are screened by Nuclear Licensing to determine which ones are reportable and are to be filed as 10 CFR 50.59 changes.

- SERI procedures for safety evaluations and applicability reviews (or screenings) are deficient in that if equipment or a procedure is not explicitly described in the UFSAR, a safety evaluation is not required.

- SERI procedures incorrectly state that equipment modifications requiring additions to the TS may be made and placed into service prior to NRC approval of a TS change, provided a 10 CFR 50.59 safety evaluation is made.

- The SERI procedures limit consideration of reduction in the margin of safety to those defined in the TS Bases, whereas NRC guidance is that the basis for a TS may be explicitly or implicitly in the FSAR.

- SERI guidelines and procedures define margin of safety as the margin between the failure point and the safety limit in the TS or Commission rules whereas NRC guidance is that the safety margin is the margin between the failure point and the bounding accident or peak calculated value in the FSAR which is the basis for a TS.

- SERI guidelines would require a significant increase in probability of occurrence of an accident, but NRC guidance would define any increase in probability as an unreviewed safety question.

- SERI procedures result in changes in radiation monitoring of gaseous and liquid effluents and radiological environmental effects being incorrectly considered as environmental evaluations rather than safety evaluations.

- The procedures do not require applicable qualifications for evaluators of safety evaluations and applicability reviews, although high level managers sign safety evaluations. Applicability review forms are signed only by the evaluator, except for NPE forms which are signed by the supervisor and section manager.

SERI has initiated a training program for evaluators of 10 CFR 50.59 safety evaluations, based on NUMARC guidelines. As noted in Reference (7), the staff does not agree with present NUMARC guidelines, so some retraining may be necessary.

The review of 10 CFR 50.59 safety evaluations by the Plant Safety Review Committee is thorough. However, safety evaluations and the evaluators' training could be significantly improved if the discussions in the PSRC were required to be incorporated into the safety evaluations by the evaluator.

More than 200 summaries of safety evaluations were reported by SERI during the year considered in the audit, most concerning equipment changes in the plant or changes in the UFSAR. The quality of the summaries varied widely. The summaries should be improved by summarizing the function of the equipment changed, the accidents or equipment malfunctions considered, the margin of safety and the effect of the change on the function, accidents and margin of safety. The 14 SEs selected for cursory review and the six SEs selected for indepth review were selected because they appeared to be questionable with respect to meeting 10 CFR 50.59 criteria. It is likely that a goodly percentage of those not audited would clearly meet the 10 CFR 50.59 criteria.

Regarding the 14 safety evaluations (SEs) that received a cursory review, the following conclusions were reached:

- Half of the SEs had a satisfactory description of the function of the equipment being changed and less than half identified the accident or anticipated operational occurrence considered. Only one of the 14 identified the margin of safety relevant to the change. These deficiencies can be traced in part to the procedural deficiencies identified above.

- Several of the SEs were made for changes in the UFSAR. Because the changes were extensive and the safety evaluations were deficient, it was not possible to assess from the brief review during the audit whether an unreviewed safety question is involved.

Regarding the indepth reviews of SEs, the following conclusions were reached:

- NPE-86-241. The fuel pool cooling and cleanup pump room is calculated to have an air temperature of 107°F upon a loss of auxiliary building ventilation system. Electrical equipment in the room has been shown to operate in temperatures up to 104°F with a 10% margin. TS 3/4 7.8 requires air temperature in the Auxiliary Building (general area) to be less than 104°F. This appears to require a TS change and because of the reduction in margin of safety, the degraded cooler performance appears to involve an unreviewed safety question.

- NPE-86-279. A containment hatchway crane was installed inside containment to be used only during outages to prevent handling of heavy loads by the polar crane from being a critical path for outages. The UFSAR references the licensee's response to NUREG-0612 only briefly and generally in UFSAR Section 9.1.4.3. This is a deficiency in updating the FSAR. The response showing how the safety significant cranes meet NUREG-0612 should have been included because it was a substantial part of the staff's OL review. When the new crane was added, the response to NUREG-0612 should have been revised. The licensee did provide information during the audit to show the degree of conformance to NUREG-0612 criteria. However, the new crane does not have operating procedures based on these criteria; and, therefore, the probability of dropping a heavy load is increased. Also, the margin of safety may be reduced because standards recommended by the criteria are not applicable to the new crane. Accordingly, it may involve an unreviewed safety question.

- NLS-87-003. The UFSAR was revised to delete water level below the top of the active fuel, as an emergency procedure action point to manually start hydrogen recombiners. A 3.5% hydrogen concentration as measured on the hydrogen analyzers is now the only measurement used to manually start recombiners. Igniters are manually initiated by both parameters. This appears to be an increase in the probability of malfunction of the initiation of the recombiners following a LOCA because of the reduction in diversity and the unreliability of the hydrogen analyzers compared to water level measurement. Therefore, it may involve an unreviewed safety question.

- PLS 86-123. The standby gas treatment (SGTS) automatic control circuit for stopping the fans if a high charcoal temperature is reached was modified to delete the automatic trip feature because the relays were not environmentally qualified. The concern was that a trip of the fans because of relay failure would result in failure of the SGTS. Alarm response instructions were modified to require manual fan shutdown in the event a high temperature alarm was received. Substitution of a manual trip of the fans for an automatic trip may increase offsite dose consequences because of the longer time required for shutting off fans. Therefore, this may involve an unreviewed safety question.

- PLS 86-132. This SE was prepared for removal of snubbers for maintenance and surveillance. This SE adequately demonstrated that the 10 CFR 50.59 criteria were met.

- PLS 86-136. This SE was prepared to delete a statement in the UFSAR that, during fuel handling operations within the containment, the equipment hatch and one door in the personnel locks will be closed at all times. The UFSAR dose analysis is based on the equipment hatch being open for a fuel assembly dropped inside containment. However, the staff's SER dose analysis for this accident is based on the equipment hatch and personnel lock door being closed. It appears that radiological dose consequences calculated by the licensee (in the FSAR) would not be increased, but that doses calculated by the staff (in the SER) would be increased. Accordingly, the change in the UFSAR should have been considered an unreviewed safety question and submitted to the staff for review.

The NRC technical staff will review the five safety evaluations which were assessed as possibly involving an unreviewed safety question as defined in 10 CFR 50.59. Conclusions of the review will be reported as a supplement to this audit report.

R. Jo for

Lester L. Kintner, Project Manager
Project Directorate II-1
Division of Reactor Projects I/II

OFC	:PM:PD21:DRPR	:D:PD21:DRPR	:	:	:	:
NAME	:LKintner:cn	: EAdensam	:	:	:	:
DATE	: 7/6/88	: 7/8/88	:	:	:	:

OFFICIAL RECORD COPY

References

1. "Safety and Environmental Review and Evaluation," SERI Operating Manual No. 7.205, Revision 1; November 17, 1987.

2. "Nuclear Licensing Administrative Procedure - Safety and Environmental Evaluations - Safety Related" Procedure No. 1.12, Revision No. 3; March 23, 1987.

3. "10 CFR 50.59 Safety Evaluations" NPE Administrative Procedure No. 316, Revision 5; August 15, 1987 as revised October 26, 1987.

4. "Administrative Procedure - Safety and Environmental Evaluations - Safety Related," 01-S-06-24, Revision 10, July 29, 1987.

5. "Preparation, Revision, Distribution, and Control of Procedures/ Instructions/Manuals," QAP 5.10, Revision 19; January 7, 1988.

6. "Nuclear Support - Administrative Procedure - Safety Related," NSAP 1.12; May 11, 1987.

7. Letter from C. E. Rossi (NRC) to Thomas E. Tipton (NUMARC), dated May 12, 1988.

DISTRIBUTION FOR MEETING SUMMARY DATED: July 11, 1988

Facility: Grand Gulf Nuclear Station

Docket File
NRC PDR
Local PDR
PDII-1 Reading
E. Adensam
P. Anderson
L. Kintner
OGC
E. Jordan (MNBB 3302)
J. Partlow (9A2)
ACRS (10)

cc: Licensee/Applicant Service List

DF01
1/1

Mr. Oliver D. Kingsley, Jr.
System Energy Resources, Inc.

Grand Gulf Nuclear Station (GGNS)

cc:
Mr. Ted H. Cloninger
Vice President, Nuclear Engineering
and Support
System Energy Resources, Inc.
Post Office Box 23054
Jackson, Mississippi 39205

Robert B. McGehee, Esquire
Wise, Carter, Child, Steen and Caraway
P.O. Box 651
Jackson, Mississippi 39205

Nicholas S. Reynolds, Esquire
Bishop, Liberman, Cook, Purcell
and Reynolds
1400 L Street, N.W.
Washington, D. C. 20005-3502

Mr. Ralph T. Lally
Manager of Quality Assurance
Middle South Utilities System
Services, Inc.
639 Loyola Avenue, 3rd Floor
New Orleans, Louisiana 70113

Mr. John G. Cesare
Director, Nuclear Licensing
System Energy Resources, Inc.
P.O. Box 23054
Jackson, Mississippi 39205

Mr. C. B. Hogg, Project Manager
Bechtel Power Corporation
P. O. Box 2166
Houston, Texas 77252-2166

Mr. Ross C. Butcher
Senior Resident Inspector
U. S. Nuclear Regulatory Commission
101 Marietta Street, N.W., Suite 3100
Atlanta, Georgia 30323

Mr. James E. Cross
GGNS Site Director
Systems Energy Resources, Inc.
P.O. Box 756
Port Gibson, Mississippi 39150

Mr. C. R. Hutchinson
GGNS General Manager
System Energy Resources, Inc.
Post Office Box 756
Port Gibson, Mississippi 39150

The Honorable William J. Guste, Jr.
Attorney General
Department of Justice
State of Louisiana
Baton Rouge, Louisiana 70804

Office of the Governor
State of Mississippi
Jackson, Mississippi 39201

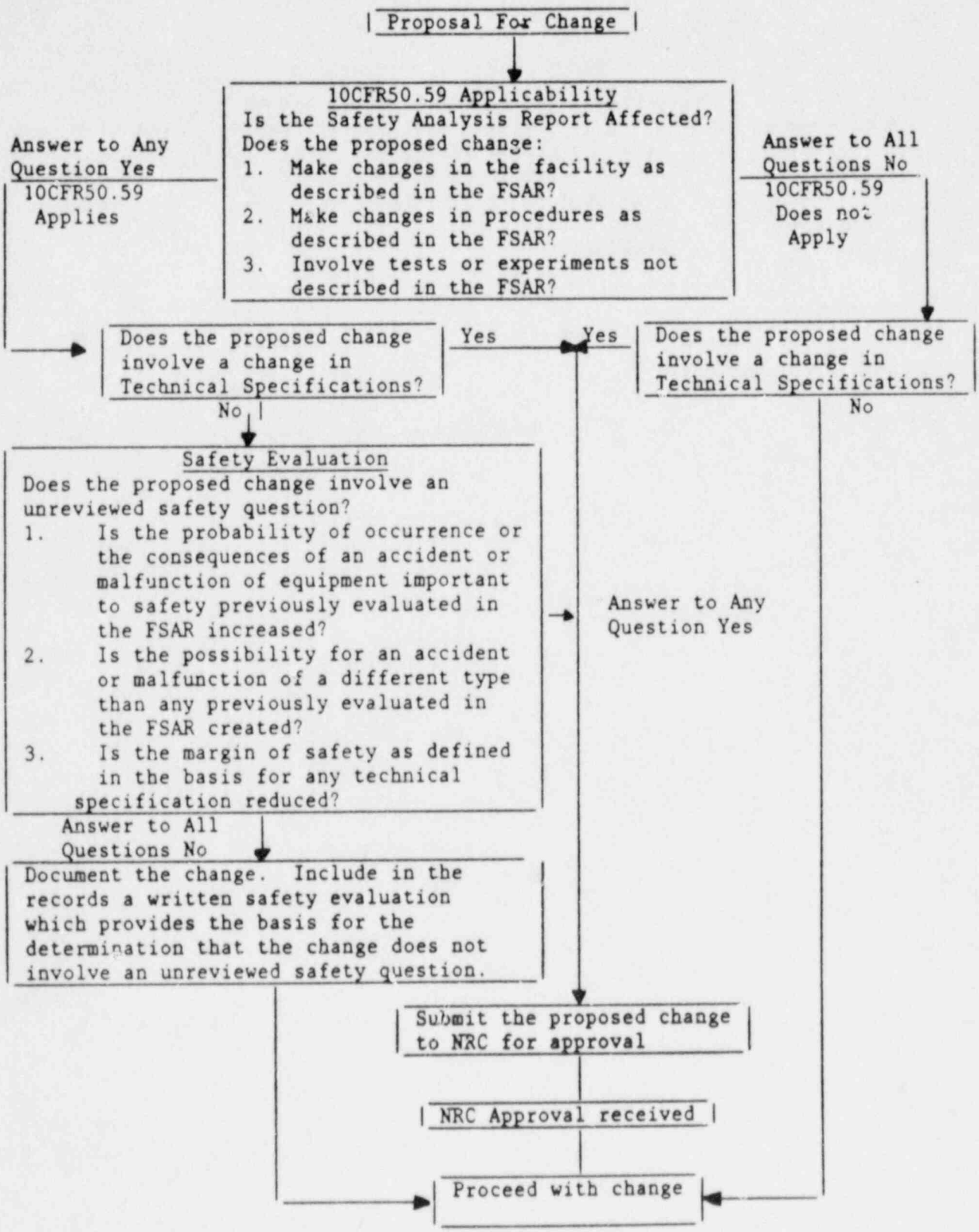
Attorney General
Gartin Building
Jackson, Mississippi 39205

Mr. Jack McMillan, Director
Division of Solid Waste Management
Mississippi Department of Natural
Resources
Post Office Box 10385
Jackson, Mississippi 39209

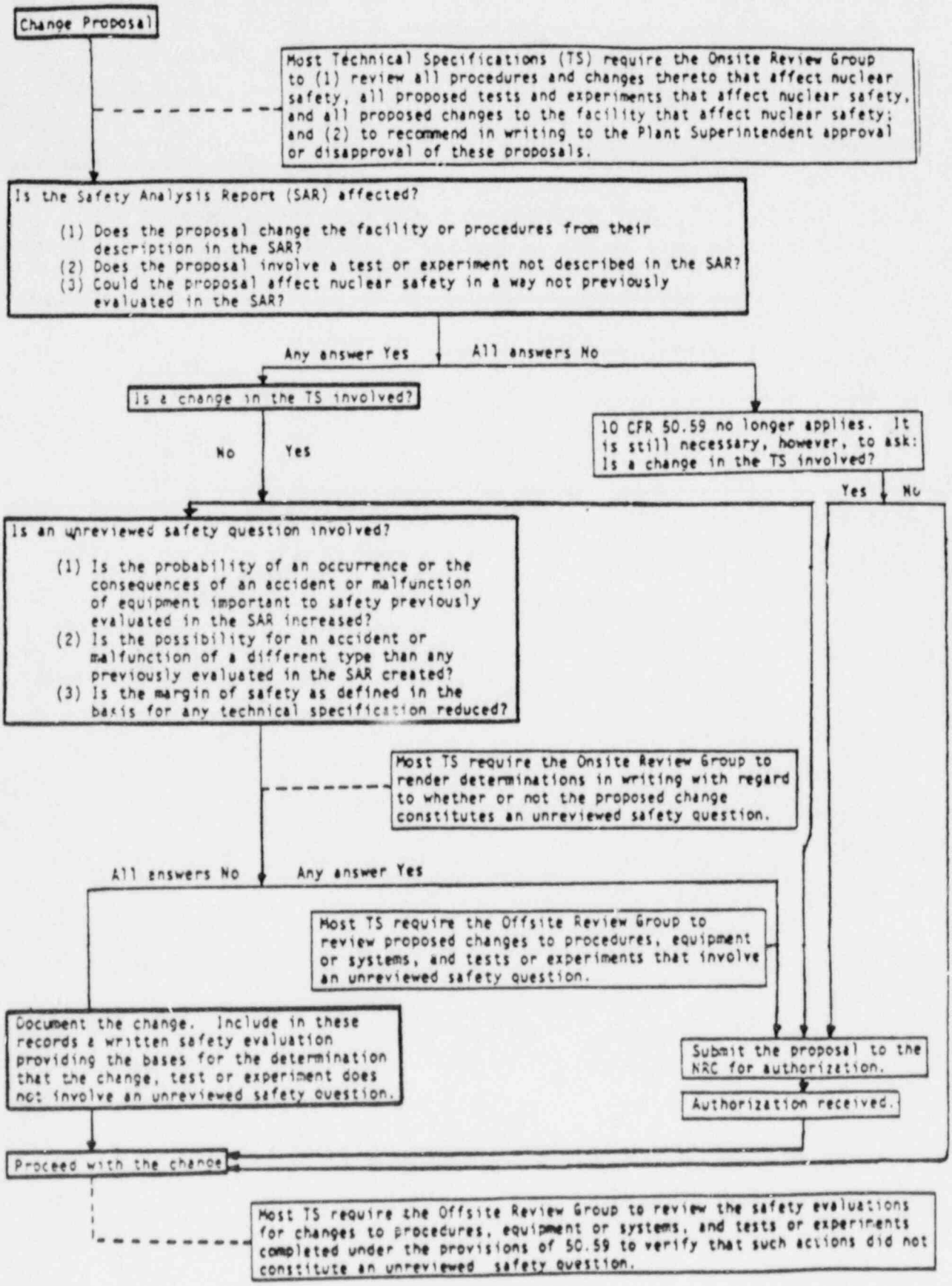
Alton B. Cobb, M.D.
State Health Officer
State Board of Health
P.O. Box 1700
Jackson, Mississippi 39205

President
Claiborne County Board of Supervisors
Port Gibson, Mississippi 39150

Safety Evaluation Flow Chart



CHANGES TO FACILITIES, PROCEDURES AND TESTS (OR EXPERIMENTS) - 10 CFR 50.59



SAFETY AND ENVIRONMENTAL EVALUATION APPLICABILITY REVIEW FORM

PROCEDURE/DOC. NO. _____

REVISION NO. _____ ACN NO. _____

SAFETY EVALUATION APPLICABILITY REVIEW

		Yes	No
(1)	Change to Facility as Desc. in FSAR	___	___
(2)	Change to Procedure as Desc. in FSAR	___	___
(3)	Test or Experiment not Desc. in FSAR	___	___
(4)	Change to Tech. Specs.	___	___

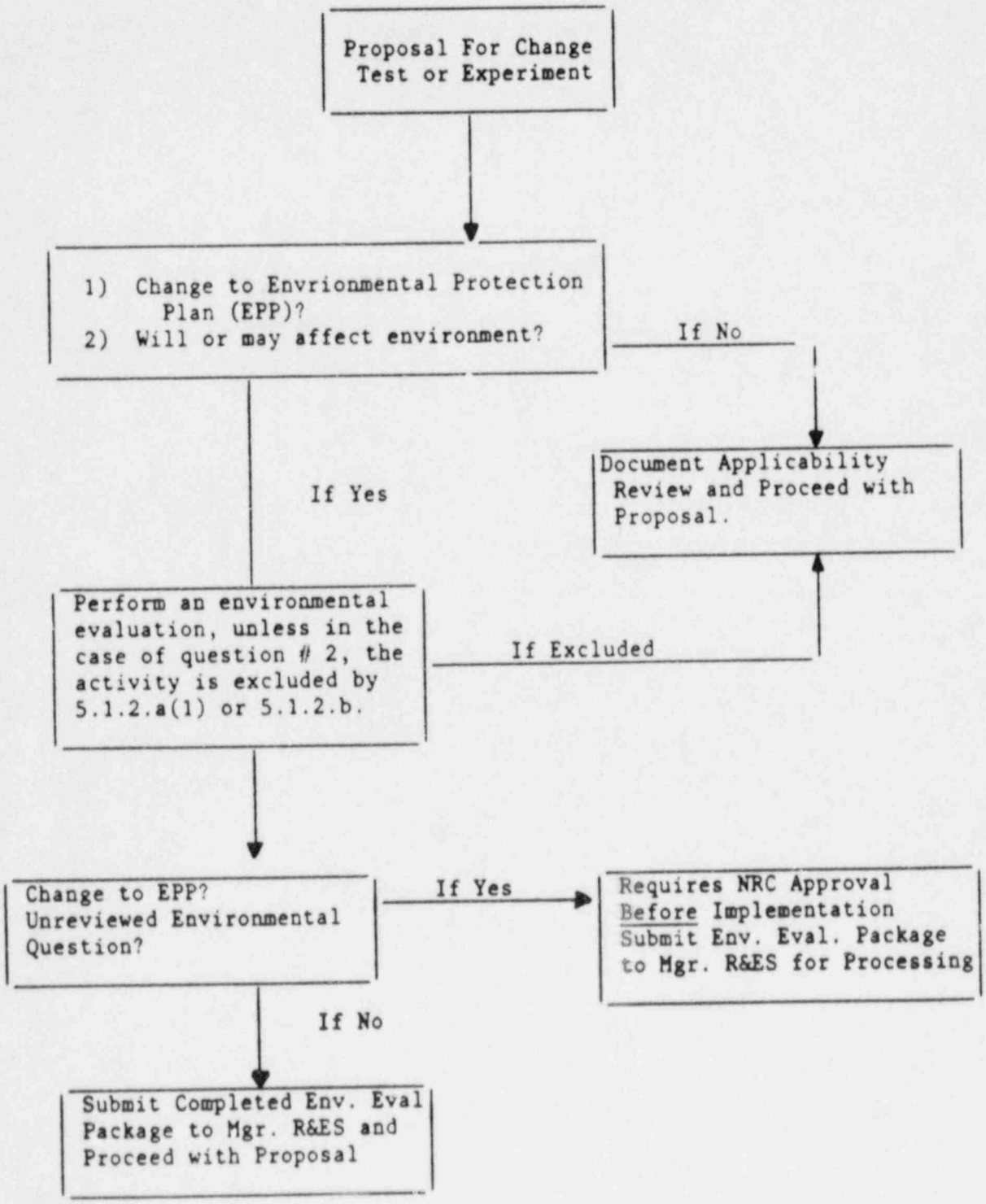
(If yes, perform 10CFR50.59 Safety Eval.)

ENVIRONMENTAL EVALUATION APPLICABILITY REVIEW

(1)	Change to Environmental Protection Plan (If yes, perform Environmental Eval.)	___	___
(2)	Will or may effect environment (If yes, perform an environmental evaluation unless excluded by 5.2.1.a [ ] or 5.2.1.c [ ]. (Check appropriate box.)	___	___

Signature _____ Date _____
Reviewer

ENVIRONMENTAL EVALUATION FLOW CHART



GRAND GULF NUCLEAR STATION UNIT 1
CHANGES, TESTS OR EXPERIMENTS
SAFETY AND ENVIRONMENTAL EVALUATION FORM

Originator ___1_____ Dept./Section _____1_____ Evaluation No._____5__
Document Evaluated ___2_____ Description_____6__
References _____3_____ _____ _____
_____ _____ _____
Attachments _____4_____ _____ _____
_____ _____ _____

FSAR Change Required? [ ] Yes [ ] No CR #_____ (If Yes) 7

Technical Specification Change Required? [ ] Yes [ ] No CR #_____ (If Yes)

I. SAFETY EVALUATION [ ] Not Applicable per Safety Evaluation Applicability Review 8

A. TECHNICAL SPECIFICATIONS

9 [ ] YES [ ] NO 1. Implementation or performance of the action described in the evaluated document will require a change to the GGNS Unit 1 Technical Specifications.
Basis:_____
_____

B. UNREVIEWED SAFETY QUESTION

10 Implementation or performance of the action described in the evaluated document will:

[ ] YES [ ] NO 1. increase the probability of occurrence of an accident previously evaluated in the FSAR.
Basis:_____
_____

[ ] YES [ ] NO 2. increase the consequences of an accident previously evaluated in the FSAR.
Basis:_____
_____

[ ] YES [ ] NO 3. increase the probability of a malfunction of equipment important to safety previously evaluated in the FSAR.
Basis:_____
_____

[ ] YES [ ] NO 4. increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR.
Basis:_____

[ ] YES [ ] NO 5. create the possibility of an accident of a different type than any previously evaluated in the FSAR.
Basis:_____

[ ] YES [ ] NO 6. create the possibility of a malfunction of equipment important to Safety different than any previously evaluated in the FSAR.
Basis:_____

[ ] YES [ ] NO 7. reduce the margin of safety as defined in the bases for any technical specification.
Basis:_____

II. ENVIRONMENTAL EVALUATION [ ] Not Applicable per Environmental Evaluation Applicability Review

A. ENVIRONMENTAL PROTECTION PLAN 11

12 [ ] YES [ ] NO 1. will require a change in the Environmental Protection Plan.
Basis:_____

B. UNREVIEWED ENVIRONMENTAL QUESTION

13 [ ] YES [ ] NO 1. concerns a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the NRC staff's testimony to the Atomic Safety and Licensing Board (ASLB), supplements to the FES, environmental impact appraisal, or in any decisions of the ASLB.
Basis:_____

[ ] YES [ ] NO 2. concerns a significant change in effluents or power level.
Basis:_____

[ ] YES [ ] NO 3. concerns a matter not previously reviewed and evaluated in the documents specified in II.B.1. above, which may have a significant adverse environmental impact.
Basis:_____

Evaluated: _____14_____
Originator/Date

Reviewed/Approved: _____15_____
Reviewer/Date

PLANT SAFETY REVIEW COMMITTEE REVIEW
(For Safety Evaluations Only)

Reviewed/Approved: _____16_____
Chairman, PSRC/Date

SAFETY EVALUATION APPLICABILITY REVIEW FORM
(Reduced Form 316.2)

SAFETY EVALUATION APPLICABILITY REVIEW FORM

DOCUMENT TYPE & NO. _____

REVISION NO. _____

SAFETY EVALUATION APPLICABILITY REVIEW		
	Yes	No
(1) Change to Facility as Desc. in FSAR	____	____
(2) Change to Procedure as Desc. in FSAR	____	____
(3) Test or Experiment not Desc. in FSAR	____	____
(4) Change to Tech. Specs.	____	____
(If Yes, perform 10CFR50.59 Safety Evaluation)		
Safety Evaluation No.* _____		
RE: _____ DATE: _____		
GS: _____ DATE: _____		
CPE: _____ DATE: _____		

*Insert N/A if the answers to all the above questions are marked "No.". Insert Safety Evaluation No. if any of the above questions are marked "Yes".

FORM 316.2, Rev. 0

TABLE II - 1

Accident Consequences

Fuel cladding -	
MCPR - Two Loop Operation	> 1.06 (TS 2.1.2)
- Single Loop Operation	> 1.07 (TS 2.1.2)
Peak Cladding Temperature (PCT)	< 2200°F (10CFR50.46(b)(1))
Increase in PCT	< 20°F (10CFR50 App. K II.1.b)
Fuel Enthalpy	< 280 cal/gm (TS Bases 3/4.1.4)
Reactor Coolant Boundary -	
Pressure (Operating Modes)	< 1325 psig (TS 2.1.3)
Water Inventory (Refueling Modes)	> Top of Active Fuel (TS 2.1.4)
Containment -	
Peak Pressure	< 11.5 psig (TS Base 3/4.6.1.6)
Internal Containment-to-Auxiliary Bldg and Enclosure Bldg Differential Pressure	> - 0.1 psid, < + 1.0 psid (TS Bases 3/4.6.1.7)
Radiological -	
Accident Doses	< 10CFR100 limits

ENCLOSURE 1

NRC EXIT MEETING ATTENDANCE SHEET

April 29, 1988

Attendees	Title	Organization
D. Cupstid	Tech. Support Supt.	SERI
W. Eiff	Prin. Quality Eng./NPE	SERI
G. Rogers	Engineer/Licensing	SERI
L. Daughtery	Compliance Supv.	SERI
J. Roberts	Manager, PM&C	SERI
S. Bennett	Licensing Proj. Sup.(NL)	SERI
M. Wright	Manager, Plant Support	SERI
A. McCurdy	Manager, Plant Operations	SERI
C. Hayes	Quality Programs Supv.	SERI
J. Czaiku	Nuc. Specialist	SMEPA
R. Moomaw	Tech Assist. to Mgr. Main.	SERI
R. Butcher	Sr. Resident Inspector	NRC
J. Summers	Compliance Coordinator	SERI
S. Fieth	Mgr. NPE	SERI
G. Ceasare	Director, Nuclear Licensing	SERI
J. Cross	Site Director	SERI
J. Yelverton	General Manager (Acting)	SERI
L. Kintner	Sr. Project Manager	NRC

10CFR50.59 PROGRAM IMPROVEMENT
4/25/88

OBJECTIVES

- o ADDRESS SRC/PSRC CONCERNS AND SALP COMMENTS
- o ACHIEVE CONSISTENT QUALITY
- o MEET INCREASING STANDARD IN INDUSTRY

SHORT TERM ACTIONS

- o OBTAINED/DISTRIBUTED SUMMARY OF PSRC OBSERVATIONS AND FREQUENTLY OBSERVED PROBLEMS
- o DISTRIBUTED LESSONS LEARNED FROM VARIOUS NRC INSPECTIONS AT OTHER UTILITIES

LONG TERM PROGRAM

- o RESEARCH
- o PUBLISH SERI "GUIDELINES"
- o REVIEW GGNS PROGRAM, PROCEDURES, PERSONNEL KNOWLEDGE, 50.59S AGAINST SERI "GUIDELINES"
- o ADDRESS FINDINGS

STATUS

- o SERI GUIDELINES PUBLISHED
- o REVIEW OF PROCEDURES AND SAMPLE 50.59S COMPLETE
- o PERSONNEL INTERVIEWS COMPLETE
- o ASSESSMENT REPORT ISSUED WITH FINDINGS & RECOMMENDATIONS
- o TRAINING PROGRAM ESTABLISHED & PRESENTED TO 85 EMPLOYEES TO DATE

FINDINGS

- o OVERALL, 50.59 QUALITY RELATIVELY GOOD WHEN COMPARED AGAINST SERI "GUIDELINES"
  - SRC/PSRC OBSERVATIONS: POSITIVE TREND
  - INCONSISTENT LEVEL OF DOCUMENTATION
- o TRAINING
  - INCONSISTENT TRAINING AND QUALIFICATION REQUIREMENTS FROM DEPARTMENT TO DEPARTMENT
- o PROCEDURES
  - NUMEROUS IMPROVEMENT AREAS TO ACHIEVE "GUIDELINES"
  - LEVEL OF REQUIREMENT DOCUMENTATION
  - STANDARD DEFINITIONS

ACTIONS OUTSTANDING

- o PROVIDE TRAINING TO APPROPRIATE PERSONNEL

  APPROPRIATE PERSONNEL INCLUDE:

  - 10CFR50.59 SCREENERS, ORIGINATORS, REVEIWERS/APPROVERS
  - SRC MEMBERS
  - PSRC MEMBERS

- o REVISE SERI POLICY & PROCEDURES TO ESTABLISH REQUIREMENTS TO ATTEND TRAINING

  - TO BE COMPLETED BY JULY 1988

- o EVALUATE EFFECTIVENESS OF IMPROVEMENTS TO BE COMPLETED BY OCTOBER 1988