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Docket Nos. 50-348
50-364

Director, Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. L. S. Rubenstein

Gentlemen:

Response to NRC Evaluation of
Safety Parameter Display System (SPDS) for
Joseph M. Farley Nuclear Plant - Units 1 and 2

By letter dated June 12, 1985, the NRC Staff transmitted the Safety Evaluation (SE) for SPDS, Item I.D.2 of NUREG-0737, Supplement 1. This transmittal requested additional information pertaining to Sections III.B and III.E of the SE. The requested information is attached for your use.

If you have any questions, please advise.

Respectfully submitted,

R.P. McDonald

R. P. McDonald

RPM/JLO:ddb-D36

Attachment

cc: Mr. L. B. Long
Dr. J. N. Grace
Mr. E. A. Reeves
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A005
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Attachment

Response to NRC Evaluation of SPDS for
Joseph M. Farley Nuclear Plant - Units 1 and 2

NRC Request

In its evaluation of the SPDS variables, the staff has considered the Westinghouse Owner's Group's "Westinghouse Emergency Response Guidelines (ERGs) Program," which was reviewed and approved by the Staff, as a principal technical source of variables important to safety. The SPDS variables selected by the licensee and their coordination with guideline safety functions are summarized in Table 1 grouping made by the licensee. While the Staff finds that the variables selected comprise a general comprehensive list, it is noted that the status of the following variables is not discussed in the licensee's submittal:

1. Containment Hydrogen Concentration

   Containment hydrogen concentration is a key parameter used in the emergency guidelines to monitor combustible gas control and to indicate a compromise of the "Containment Conditions" safety function.

2. RCS Level

   The SPDS status of RCS level is not discussed per se in the licensee's submittals; however, an inventory critical safety function (with supporting variables) is discussed. The SPDS status of RCS level at Farley should be clarified to demonstrate that either RCS level or its functional equivalent is available.

3. Source Range Neutron Flux

4. Intermediate Range Neutron Flux

5. RHR Flow

6. Steam Generator (or Steamline) Radiation

7. Stack Radiation

8. Containment Isolation

The above variables do, for given scenarios, provide unique inputs to determinations of status for their respective CSFs, which have not been discussed by the applicant as being satisfied by other variables in the proposed Farley SPDS list. The Staff recommends that the licensee address these variables and and their function by : (1) adding these variables to the Farley SPDS, (2) providing alternate added variables along with justifications that these alternates accomplish the same safety functions for all scenarios, (3) providing justification that variables currently on the Farley SPDS do in fact accomplish the same safety functions for all scenarios, or (4) identifying that these variables are in fact available from the SPDS console.

Additionally, for a rapid assessment of Radioactivity Control, the licensee has not demonstrated how radiation in the secondary system (steam generators and steamlines) is monitored by SPDS when the steam generators and/or their steamlines are isolated. The licensee should discuss this capability.

APCo Response

1. Containment Hydrogen Concentration

Containment hydrogen concentration was not selected as a variable for the SPDS in order to ensure consistency with the emergency operating procedures. A flammable hydrogen concentration exists only following the metal-water reaction during an inadequate core cooling condition. To arrive at an inadequate core cooling condition a significant quantity of reactor coolant inventory must be released to the containment, most likely a reactor coolant system rupture.

A high containment pressure, rather than a high hydrogen concentration was selected as the indication of the potential for breach of the containment since high pressure will always exist when a high hydrogen concentration exists. However, the reverse is not true (e.g., following a secondary system rupture inside of the containment). Containment pressure is, therefore, a better indication of the potential for containment breach.

In development of the Westinghouse Owners Group Emergency Response Guidelines (ERGs), containment hydrogen was addressed in those areas in which it is potentially present and where containment integrity is challenged. These areas include plant conditions associated with loss of reactor coolant transients, inadequate core cooling transients and transients with an associated containment pressure increase above the value (high-2 pressure) at which hydrogen ignition poses a containment integrity concern. The hydrogen generation associated with a loss of reactor coolant is addressed in Farley Nuclear Plant (FNP) Emergency Event Procedure (EEP), EEP-1, Loss of Reactor or Secondary Coolant, through sampling and operation of hydrogen recombiners, if appropriate. This treatment serves to initiate early actions to address hydrogen, if required. The Status Trees are used to diagnose inadequate core cooling conditions (the precursor of significant hydrogen generation and high containment hydrogen concentrations) and containment high pressure conditions (which in combination with high hydrogen concentrations pose a potential integrity concern). Associated Functional Restoration Procedures (FRPs) FRP-C.1 and FRP-Z.1, respectively, then direct the operator to sample containment hydrogen concentration and initiate appropriate actions based on hydrogen concentration. This treatment of the hydrogen concern is considered an acceptable alternative to including hydrogen concentration in the Containment Status Tree.

In summary, because a high containment pressure is the most accurate indication of a potential containment integrity challenge, its use has been specified in the Containment Tree, and, therefore, the SPDS. Hydrogen concentration indications are available elsewhere for those actions requiring this indication.

2. <u>RCS Level</u>

Reactor vessel level indication is being installed at FNP and will be added to the SPDS during the Unit 2 Fifth and Unit 1 Eighth Refueling Outages. The critical safety function status trees are being modified to incorporate the reactor vessel instrumentation.

As an interim solution, the WOG status trees developed for implementation without vessel level indication will be used. When used in conjunction with pressurizer level, adequate indications are available to address the range of possible RCS levels from a normal hot shutdown condition to the generation of a void in the reactor vessel above the top of the core.

3. <u>Source Range Neutron Flux</u>

Source range startup rate was selected instead of source range neutron flux for display on the SPDS. Post-reactor trip neutron flux decreases into the source range and remains there. While the source range count rate fluctuates, no sustained trending is expected under normal post-trip conditions. The absolute value of the flux in the source range provides little useful information on which to base critical operator actions for reactivity control. A positive startup rate, however, indicates an eventual loss of subcriticality and is the most rapid and reliable indication of plant safety status potentially requiring an operator response.

Source range flux can be used in situations other than challenges to the fission product barrier (fuel clad) when time (before loss of subcriticality) is not a concern. Source range flux located on nuclear instrumentation panels in the control room can be used as a backup to the SPDS startup rate indication. This, however, requires continuous data manipulation by the operator during a potential challenge to a fission product barrier.

4. <u>Intermediate Range Neutron Flux</u>

Intermediate range startup rate has been chosen for presentation on the SPDS instead of intermediate range neutron flux for the same reasons as discussed above for source range startup rate and flux. In addition, a positive startup rate in the intermediate range is a more severe challenge to the critical safety function of reactivity control than is a positive startup rate in the source range since less time is allowed for the operating staff to determine that a potential challenge exists and to take corrective action before power is generated.

As was true for the source range, intermediate range neutron flux level is presented elsewhere in the control room and can be used by the operating staff in situations not related to challenges to the fission product barrier or as a backup to SPDS startup rate.

5. RHR Flow

RHR Flow is an indirect indication of reactor core cooling and heat removal from the primary system under very limited situations (i.e., it is a precursor to the potential loss of core cooling and not a symptom). Core exit temperature and core subcooling margin, both included in the core cooling tree, are the direct means for determining if adequate core cooling exists.

A sustained loss of core cooling does not result in any operator actions requiring use of RHR flow. The high pressure and low pressure safety injection subsystems are used to ensure reactor core cooling. This includes the backup method of feed and bleed cooling as specified in FRP-C.1, "Response to Inadequate Core Cooling."

The primary and preferred means of post-reactor trip heat removal from the primary system is through the steam generators. The parameters to monitor heat removal from this path are included in the heat sink tree. Neither the tree nor procedures to restore a secondary heat sink require RHR flow for their use. Following a sustained loss of steam generator heat sink the backup means to remove core heat is with feed and bleed actions.

In summary, based on the Farley emergency operating procedures and critical safety function status trees, the safety function identified in Supplement 1 to NUREG-0737 as "Reactor Core Cooling and Heat Removal from the Primary System" does not necessitate presentation of RHR flow on the Farley SPDS. RHR flow is readily available to the operating staff, however, in the control room.

6. Steam Generator (or Steamline) Radiation

The principal use of steam generator or steamline radiation is for the diagnosis and control of radiation following a steam generator tube rupture. Without a tube rupture the secondary system radioactivity is expected to be well within the limits specified in the Farley Technical Specifications. To diagnose a tube rupture in a particular steam generator, narrow range water level indications are available in each steam generator to detect an uncontrolled level increase. If the level is uncontrollable, a tube rupture has occurred. Appropriate actions are then taken to isolate this steam generator and minimize any primary-to-secondary leakage through the break

These actions, specified in EEP-3, "Steam Generator Tube Rupture," and related procedures, are intended to minimize and terminate primary-to-secondary leakage. In following these, concerns such as steam generator overfill and the concurrent transfer of contaminated primary water are optimally addressed. Control of radioactivity is an inherent requirement in the tube rupture-related procedures. Providing a specific radiation monitor for each loop is, therefore, not essential for the operating staff to control radioactivity and would not provide any information (in addition to steam generator water level) that results in different operator actions or better control of radioactivity.

Determination of the magnitude of an offsite dose rate is included in the FNP Emergency Implementing Procedures. The FNP Analytical Data Management System (ADMS) is utilized for emergency dose rate calculations. As a result, the radioactivity concentration has not been included in the SPDS.

7. Stack Radiation

Control of radioactivity, including that exhausted from the stack, is an inherent requirement of the emergency operating procedures. As such, a specific status tree has not been identified, but is implicit in the actions performed by the operating staff.

Specific equipment and actions are available if radioactivity is released within the containment. For example containment filters and spray can be used to control the amount of radioactivity released from containment. In the case of stack radiation releases, little or no control is available once radioactivity travels to the plant ventilation system. Presenting this information to the operating staff will not significantly change any actions that might otherwise be taken.

FNP utilizes a Vent Stack Radiation Monitoring System which interfaces with the ADMS to quantify the magnitude of the release. Actions taken as a result of stack radiation are defined in the FNP Emergency Plan. This information is therefore not presented on the SPDS.

8. Containment Isolation

Containment isolation was not included in the Farley SPDS since (1) containment isolation is verified by the operating staff very shortly after the post-accident response begins and (2) the emergency procedures and status trees are written to protect the fission product barriers thereby minimizing any radioactivity release.

Transients resulting in the potential for increase of radioactivity above the normal Technical Specification allowable limit will result in a safety injection signal. Phase A containment isolation occurs automatically on a safety injection signal. Shortly after the operating staff implements EEP-0, "Reactor Trip or Safety Injection," phase A containment isolation is verified if plant conditions require phase A isolation. If isolation has not occurred it is performed manually. Similarly, if the high containment pressure phase B isolation setpoint is exceeded, isolation occurs automatically and is verified by the operator. Again, it will be performed manually if it has not occurred automatically.

The emergency operating procedures, including the critical safety function status trees, are written and prioritized on the basis of protecting the integrity of the fission product barriers. For example, to protect the primary barrier, the fuel clad and matrix, the first status tree, "Subcriticality," and related procedures prevent and/or minimize the generation of power. Use of the second status tree "Core Cooling," ensures adequate decay heat removal. The third status tree, "Heat Sink," provides for the removal of the decay heat generated. The use of these three status trees protect the integrity of the fuel clad/matrix to prevent or minimize the release of radioactivity.

The second fission product barrier, the Reactor Coolant System, is protected by the first three status trees discussed above by minimizing the challenges from inadequate decay heat removal and the fourth status tree, "Reactor Coolant System Integrity."

The third barrier, the containment, is then addressed with the fifth status tree, "Containment."

Containment isolation, and similarly containment ventilation isolation, are verified to have occurred with use of symptom-based "containment" status trees and related procedures. Specifically, high containment pressure is indicative of an accident leading to the potential for release of radioactivty. Therefore, in the containment integrity function restoration procedure addressing high containment pressure, containment phase A isolation is verified and manually performed if necessary. Likewise, for indications of high radioactivity in the containment environment, the appropriate function restoration procedure ensures that the containment ventilation system is isolated and that the plant engineering staff is notified to obtain additional recommendations on operator actions. Even though the status tree does not specifically address containment isolation, the function is performed based on related symptoms in the existing [illegible]e.

In summary, upon entering the emergency procedures, following a safety injection signal, containment isolation is verified by the operating staff. While monitoring the status trees, symptoms indicative of situations where containment isolation may be beneficial are used to verify or initiate the containment isolation function and, therefore, are not presented on the status tree.

NRC Request

For the isolation devices between the core exit temperature monitoring system and the SPDS, the licensee shall submit the test results for confirmatory review to support the requirements (Item 3 a through f) identified in the NRC letter to the licensee dated April 4, 1984.

APCo Response

By letter dated May 11, 1984 Alabama Power Company informed the NRC that SPDS would be connected to qualified isolators for each thermocouple loop. Subsequent to this letter, Alabama Power Company has revised its design for the SPDS/Thermocouple interface. Thermocouple inputs for SPDS are now planned to be obtained via a fiber-optic data link between SPDS and a qualified Inadequate Core Cooling System (ICCS) processor. The physical characteristics of fiber-optics are such that no credible fault can be postulated for SPDS that will adversely affect the ICCS processor. Test results should therefore no longer be applicable.

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