FPL Energy
Seabrook Station

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JAN 28 2005

Docket No. 50-443
SBK-L-05013

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Seabrook Station
Facility Operating License NPF-86
Supplemental Response to Request for Additional Information Regarding
License Amendment Request 04-03, Application for Stretch Power Uprate

References:

1. FPL Energy Seabrook, LLC letter NYN-04016, "LAR 04-03, Application for Stretch Power Uprate," dated March 17, 2004.

2. NRC letter to FPL Energy Seabrook, LLC, "Seabrook Station Unit 1 – Request for Additional Information for Proposed Amendment Request Regarding the Application for Stretch Power Uprate (TAC MC2364)," dated August 18, 2004.

3. FPL Energy Seabrook, LLC letter SBK-L-04072, "Response to Request for Additional Information Regarding License Amendment Request 04-03, Application for Stretch Power Uprate," dated October 12, 2004.

By letter dated March 17, 2004 (Reference 1), FPL Energy Seabrook, LLC (FPL Energy Seabrook) requested an amendment to facility operating license NPF-86 and the Technical Specifications for Seabrook Station. This amendment request (LAR) is an application for a stretch power uprate which will increase the Seabrook Station licensed reactor core power by 5.2% from 3411 megawatts thermal (MWt) to 3587 MWt.

In Reference 2, the NRC requested additional information to support its review of Seabrook Station LAR 04-03. By letter dated October 12, 2004 (Reference 3) FPL Energy Seabrook provided its responses to the requests for additional information (RAIs) provided in your correspondence.

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Based on requests from the NRC staff during teleconferences on December 7, 2004 and January 5, 11 and 25, 2005, FPL Energy Seabrook is providing additional information related to the inadvertent safety injection event. This supplemental response amends LAR Attachment 1 Table 6.3.1-1, "Non-LOCA Analysis Limits and Selected Analysis Results," and provides information regarding the ability to isolate a pressurizer power-operated relief valve during water relief conditions. Enclosure 2 contains a copy of Westinghouse Nuclear Safety Advisory Letter NSAL-93-013, Supplement 1, "Inadvertent ECCS Actuation at Power," dated October 28, 1994 which addresses inadvertent safety injection as a potential overpressure event. Enclosure 3 contains revised UFSAR Section 15.5.1, "Inadvertent Operation of Emergency Core Cooling System during Power Operation."

Should you have any questions concerning this LAR, please contact Mr. Stephen T. Hale, Power Uprate Project Manager, at (603) 773-7561.

Very truly yours,

FPL Energy Seabrook, LLC

Mark E. Warner
Site Vice President

Enclosures (3)

cc: S. J. Collins, NRC Region I Administrator
V. Nerses, NRC Project Manager, Project Directorate I-2
G. T. Dentel, NRC Resident Inspector

Mr. Bruce Cheney, Director
New Hampshire Bureau of Emergency Management
State Office Park South
107 Pleasant Street
Concord, NH 03301-3809

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Oath and Affirmation

I, Mark E. Warner, Site Vice President of FPL Energy Seabrook, LLC hereby affirm that the information and statements contained within the supplemental response to the request for additional information regarding license amendment request 04-03 are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Sworn and Subscribed
Before me this

28 day of January, 2005

Notary Public

Mark E. Warner
Site Vice President

MICHAEL D. O'KEEFE
MY COMMISSION EXPIRES FEB. 5, 2008
NOTARY PUBLIC
NEW HAMPSHIRE

**Enclosure 1 to Letter SBK-L-05013**

**Supplemental Information to NRC
Regarding Seabrook Station LAR 04-03
Application for Stretch Power Uprate**

**FPL Energy Seabrook Response to Inadvertent Safety Injection Question:**

Provided below is supplemental information requested by the NRC in teleconferences on December 7, 2004, and January 5, 11 and 25, 2005. The supplemental information amends LAR 04-03, Attachment 1, Table 6.3.1-1 (page 6-57), and provides additional information regarding the qualification of the pressurizer power-operated relief valves (PORVs) and PORV block valves to close during water relief.

Table 6.3.1-1 amendment

LAR Attachment 1 Table 6.3.1-1 "Non-LOCA Analysis Limits and Selected Analysis Results" (page 6-57) for events Inadvertent Operation of Emergency Core Cooling during Power Operation and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory, the Selected Result column is changed from "Operator action at 10 minutes to preclude filling," to "Operator action at 10.1 minutes to preclude opening the pressurizer safety valves."

Additional Information Regarding Qualifications of the PORV and PORV Block Valves for Water Relief

The following information is provided to address technical staff reviewer questions with regard to PORVs and PORV block valve operation associated with an inadvertent safety injection event at power. The information provided below can be found in the Seabrook Station UFSAR, sections 1.9 and 8.3; Seabrook Station letters SBN-969 dated March 17, 1986, SBN-1082 dated June 1, 1986, NYN-87136 dated November 23, 1987 and NYN-89057 dated May 8, 1989 which address closure of NUREG-0737, Item II.D.1, and in FPL Energy Seabrook's previous response to RAI #81 (SBK-L-04072).

As required by NUREG 0737, Item II.D.1, Seabrook Station had to demonstrate that the PORVs and PORV block valves, and associated upstream and downstream piping and supports, were qualified for all potential design conditions, including water relief. A summary of the component design and qualifications is provided below.

Safety Qualifications

Mechanical Design:

The PORVs and PORV block valves are classified ANS Safety Class 1 and Seismic Category 1 and are designed, fabricated and installed to the requirements of ASME Section III, Code Class 1. The piping and fittings from the pressurizer through the outlet of the PORVs are also classified ANS Safety Class 1 and Seismic Category 1 and are designed, fabricated, and installed to the requirements of ASME Section III, Code Class 1. Piping downstream of the PORVs is non-nuclear safety class, however the piping and supports have been evaluated for all design loading combinations including loads associated with a safe shutdown earthquake and water relief through the valves (see further discussion below).

Power Supplies and Controls:

The PORVs are fail closed pilot-operated solenoid valves powered by independent Class 1E power supplies. With the exception of the automatic controls for the valves, the electrical components and manual controls are classified safety related, Class 1E, and environmentally qualified.

The motor-operated PORV block valve power and controls are safety related, Class 1E, and environmentally qualified. Each PORV block valve is powered by an independent Class 1E power supply. In addition, the PORV block valves are included in the Seabrook Station Motor Operated Valve (MOV) Program, which includes the activities required to address Generic Letters 89-10, "Safety-Related Motor-Operated Valve Testing And Surveillance", 95-07, "Pressure Locking and Thermal Binding of Safety- Related Power-Operated Gate Valves" and 96-05, "Periodic Verification of Design Basis Capability of Safety Related Motor Operated Valves". This program requires periodic diagnostic testing and preventive maintenance of motor-operated valves to ensure the motor-operated valves are able to perform their intended safety function under design basis conditions. The design differential pressure for PORV block valve motor actuator sizing is 2365 psid, which is the closure setpoint of the PORVs. Based on the actuator capability, there is an approximate 20% margin over and above the capability required for closure of the valves at the design differential pressure.

Qualifications for Water Relief:

There are two specific areas that need to be addressed to ensure the PORVs and PORV block valves can operate under water relief conditions. First, the capability of the upstream and downstream piping and supports to accommodate the stresses and loads needs to be demonstrated, and second, the plant specific water relief conditions need to be bounded by the conditions in the EPRI Pressurized Water Reactor Safety and Relief Valve Test Program.

Although water relief conditions were not specifically identified for the inadvertent safety injection event for Seabrook Station, water relief conditions were evaluated and addressed as part of Seabrook Station's response to NUREG-0737, Task II.D.2.

With regard to the capability of the piping and supports, as identified in the response to RAI 8 in letter NYN-87136, the transient, stress, and support analyses addressed the following transient cases:

Case 1: The PORVs and pressurizer safety valves open sequentially at their respective set points. Flow through the valves will be steam with the exception of the initial water seal discharge through the PORVs.

Case 2: The three pressurizer safety valves discharge saturated steam, while the PORVs remain closed.

Case 3: The two pressurizer power-operated relief valves initially discharge steam and experience a transition to saturated water release plus a subsequent actuation during which 567°F water is discharged. The pressurizer safety valves remain closed.

Case 4: The three pressurizer safety valves discharge 567°F water, while the PORVs are inoperable.

Case 5: The two PORVs discharge with pressurizer conditions of 2400 psia and 329°F, while the pressurizer safety valves remain closed.

Case 6: One PORV discharges with pressurizer conditions of 2400 psia and 329°F, while the pressurizer safety valves remain closed.

Based on a review of the above in comparison to the water relief conditions for an inadvertent safety injection event, the above analyses remain bounding.

With regard to EPRI Pressurized Water Reactor Safety and Relief Valve Test Program, in addition to steam testing, the Garrett power-operated relief valve (the valve demonstrated is representative of the Seabrook Station PORVs) was subjected to one transition test, and three high pressure water tests. In the water tests, the pressure ranged from 2640 to 2760 psia and water temperatures ranged from 249°F to 648°F. The Westinghouse PORV block valve tested in the EPRI test program is the same design as the Seabrook Station PORV block valves. The block valves were tested at a design pressure of 2485 psia with steam. Note that steam testing was determined to be acceptable for demonstration of valve closure capability for both steam and water conditions. In addition, water relief through the pressurizer safety valves and PORVs was evaluated for the feedwater line break for Seabrook Station. These conditions included a maximum pressurizer pressure of 2505 psia and liquid temperatures of 603°F to 605°F. Again, these conditions bound the conditions for an inadvertent safety injection event.

Based on the above discussions, there is reasonable assurance that the power-operated relief valves and power-operated relief valve block valves can be closed under water relief conditions during an inadvertent safety injection event.

## Credit for Automatic Operation of the PORVs for UFSAR Chapter 15 Events

None of the current Seabrook Station UFSAR Chapter 15 safety analyses takes credit for automatic pressurizer PORV operation to mitigate the consequences of an event, including the inadvertent safety injection event described in Section 15.5.1, "Inadvertent Operation of Emergency Core Cooling System during Power Operation."

Because of confusing wording, UFSAR Section 15.5.1 has been revised to eliminate reference to automatic PORV operation. A copy of the revised wording is contained in Enclosure 3.

# Enclosure 2 to Letter SBK-L-05013

## Westinghouse Nuclear Safety Advisory Bulletin
## NSAL-93-013, Supplement 1
## Inadvertent ECCS Actuation at Power

Westinghouse
Energy
Systems
Business
Unit

ESBU

# NUCLEAR SAFETY ADVISORY LETTER

24/9522

THIS IS A NOTIFICATION OF A RECENTLY IDENTIFIED POTENTIAL SAFETY ISSUE PERTAINING TO BASIC COMPONENTS SUPPLIED BY WESTINGHOUSE. THIS INFORMATION IS BEING PROVIDED TO YOU SO THAT A REVIEW OF THIS ISSUE CAN BE CONDUCTED BY YOU TO DETERMINE IF ANY ACTION IS REQUIRED.

P.O. Box 355, Pittsburgh, PA 15230-0355

| | |
|---|---|
| **Subject:** Inadvertent ECCS Actuation at Power | Number: NSAL-93-013, Supplement 1 |
| **Basic Component:** Transient Accident Analysis | Date: Oct. 28, 1994 |
| **Plants:** See Page 2, Table 1 | |
| Substantial Safety Hazard or Failure to Comply Pursuant to 10 CFR 21.21(a) | Yes ☐ No ☒ |
| Transfer of Information Pursuant to 10 CFR 21.21(b) | Yes ☐ |
| Advisory Information Pursuant to 10 CFR 21.21(c)(2) | Yes ☐ |
| **Reference:** | |

## SUMMARY

As previously described in Nuclear Safety Advisory Letter (NSAL) 93-013, dated June 30, 1993, Westinghouse discovered that potentially non-conservative assumptions were used in the licensing analysis of the Inadvertent Operation of the ECCS at Power accident.

In addition to the information provided in the original NSAL, this supplement provides additional information related to this issue and specifically notes that Positive Displacement Pump (PDP) operation during normal operating conditions will tend to aggravate the event by reducing the time to reach a pressurizer water solid condition. If water relief from the pressurizer does occur, the piping downstream of the PSRVs/PORVs must be qualified for subcooled water relief. Normal operation of a PDP, concurrent with initiation of an Inadvertent ECCS Actuation event, will serve to increase the injection flow by approximately 100 gpm and, without operator action, shorten the time to reach a pressurizer water solid condition. Though not all plants listed in Table 1 necessarily operate a PDP during normal plant operations, this supplement to NSAL 93-013 will be transmitted to those plants identified in the original NSAL to ensure continuity.

This supplemental information does not pose a substantial safety hazard or failure to comply per the definitions provided in 10 CFR Part 21.21(a).

Additional information, if required, may be obtained from the originator. Telephone 412-374-5036.

Originator(s): J. S. Galembush

H. A. Sepp, Manager
Strategic Licensing Issues

## TABLE 1 PLANT APPLICABILITY LIST

Byron 1 & 2
Braidwood 1 & 2
Zion 1 & 2
V. C. Summer
D. C. Cook 1 & 2
Shearon Harris
W. B. McGuire 1 & 2
Catawba 1 & 2
Beaver Valley 1 & 2
J. M. Farley 1 & 2
Vogtle 1 & 2
Seabrook
Millstone 3
North Anna 1 & 2
Surry 1 & 2
Salem 1 & 2
Diablo Canyon 1 & 2
Wolf Creek
Callaway
Sequoyah 1 & 2
Watts Bar 1 & 2
Haddam Neck (note 1)

Almaraz 1 & 2
Doel 1, 2 & 4
Vandellos
Asco 1 & 2
Krsko
Beznau 1 & 2
Ringhals 2, 3 & 4
Tihange 1 & 3
Zorita
C. N. des Ardennes
C. N. BR3
Kori 3 & 4
Yonggwong 1 & 2
Maansham 1 & 2
Mihama 2
Ohi 1 & 2 (note 1)
Takahama 1 (note 1)

Notes: 1. Westinghouse is not cognizant of the current ECCS design for these plants.

## TECHNICAL DESCRIPTION

### ISSUE DESCRIPTION

In the original issue of NSAL 93-013, the Inadvertent Actuation of the Emergency Core Cooling System (ECCS) accident is noted as a Condition II incident as defined by ANS-051.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." A Condition II incident is defined as a fault of moderate frequency, which, at worst, should result in a reactor shutdown with the plant being capable of returning to operation. A Condition II event cannot generate a more serious event of the Condition III or IV type without other incidents occurring independently.

As described in NSAL 93-013, the historical analysis methodology for the "Inadvertent Operation of the ECCS at Power" event used assumptions to conservatively demonstrate that the DNBR safety analysis and RCS pressure limits are met and that these assumptions may not be conservative with respect to maximizing the RCS inventory increase.

Standard Review Plan NUREG-0800, Rev. 1, Section 15.5.1, "Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory," states that "specific criteria to meet the requirements of GDC 10, 15, and 26 for incidents of moderate frequency are:

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values,

b. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs, and,

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently."

To address criterion (c), Westinghouse has historically applied the following more restrictive criterion for ease in interpreting the transient results.

> The pressurizer shall not become water solid as a result of this Condition II transient within the minimum time required for the operator to identify the event and terminate the source of fluid increasing the RCS inventory. Typically a 10 minute operator action time has been assumed.

It is easy to conclude that criterion (c) is met if it can be demonstrated that the pressurizer does not become water-solid in the minimum allowable operator action time. However, if ECCS flow is not terminated before the pressurizer becomes water solid, it is more difficult to demonstrate that this Condition II event does not lead to a more serious plant condition. Note that no credit for automatic actuation of RCS coolant letdown (pressurizer level control), pressurizer pressure control (PORVs), steam generator PORVs, or steam dump is taken since these are considered control grade systems. Without these systems available, it is anticipated that an Inadvertent ECCS Actuation at Power event could potentially lead to a water-solid pressurizer condition and result in a Condition III LOCA event if corrective action is not taken in a timely manner. An increase in the assumed injected flow due to the potential for concurrent operation of a positive displacement pump at the time of event initiation would further reduce the time to reach a water solid pressurizer condition, and hence, reduce the time available for appropriate operator actions.

## TECHNICAL EVALUATION

The historic "Westinghouse" internal acceptance criterion of preventing the pressurizer from reaching a water-solid condition during Condition II events clearly eliminates any concerns of escalating a Condition II event to a Condition III or IV event. However, this criterion is overly conservative and due to changes in analysis modeling assumptions made to conservatively analyze this event for proper consideration of pressurizer water volume, this criterion is now being challenged within the minimum allowable operator action time of 10 minutes typically assumed.

However, merely reaching a water-solid pressurizer condition does not imply that the event will escalate into that of a Condition III or IV event. ANS 51.1/N18.2-1973, lists Example (15) of a Condition II event as a "minor reactor coolant system leak which would not prevent orderly reactor shutdown and cooldown assuming makeup is provided by normal makeup systems only." Here, "normal makeup systems" is defined as those systems normally used to maintain reactor coolant inventory under respective conditions of startup, hot standby, power operation, or cooldown; using on-site power.

Since the cause of the water relief is the ECCS flow, the magnitude of the leak will be less than or equivalent to that of the ECCS (i.e., operation of the ECCS maintains RCS inventory during the postulated event and establishes the magnitude of the subject leak). Therefore, the above example of a Condition II event is met provided "orderly reactor shutdown" is also met.

To ensure "orderly reactor shutdown" can occur, the RCS pressure boundary must ultimately be isolatable once the source of the ECCS flow is terminated. To ensure the RCS pressure boundary can be isolated, the Pressurizer Safety Relief Valves (PSRVs) must function as designed and the power-operated relief and/or block valves must be available to the operator (after the minimum allowable operator action time) to provide isolation functions.

For continued conservatism in the safety analysis methodology, it is assumed that PSRVs must not pass water in order to ensure their integrity and continued availability. Therefore, the Westinghouse internal event criterion for this Condition II event is revised such that subcooled water discharge through the Pressurizer Safety Relief Valves shall be precluded for a Condition II transient.

Hence, a water-solid pressurizer condition should be precluded when the pressurizer is at or above the set pressure of the PSRVs. An exception to this criterion can be made if the utility can support a position that their PSRVs are designed and qualified to relieve subcooled water.

The plant technical specifications generally require the PORVs and block valves to be operable. Their operability is determined, in part, on the basis of their capability to manually control reactor coolant pressure. With one or more PORVs available, the PSRV setpoint will not be reached during this event. Any water discharge from the RCS would be through the PORV(s). Furthermore, isolation of the RCS following operator action to terminate ECCS flow is obtainable by available block valve closure.

For the potential condition of the plant operating with all the PORVs blocked, RCS pressure would exceed the PORV settings and continue to increase towards the PSRV setpoint. To preclude water relief through the PSRVs, either action to terminate the ECCS flow to avert a water-solid condition or to confirm that at least one PORV is unblocked and available for water relief, prior to reaching water-solid condition, must be taken within the minimum operator action time.

The acceptability of water leakage from the RCS for the Inadvertent Operation of ECCS at Power Condition II event is further supported by statements contained in NUREG-0800, Section 15.5.1 - 15.5.2. Specifically, Section III Review Procedures indicate (first paragraph on page 15.5.1-6):

> "The results of the applicant's analysis are reviewed and compared to the acceptance criteria presented in subsection II regarding maximum pressure in the reactor coolant and main steam systems and the minimum critical heat flux ratio (MCHFR) or departure from nucleate boiling ratio (DNBR). The variations with time during the transient of the neutron power, ......... pressure relief valve flow rate, and flow rate from the reactor coolant system to the containment system (if applicable) are reviewed."

Therefore, based on the aforementioned information, it is interpreted that Condition II criteria can be met with some water relief from the RCS. To meet the applicable Condition II criteria, the magnitude of any water relief must not exceed that of the normal makeup systems (which it will not by definition since the ECCS flow is the cause of the water relief) and as long as orderly shutdown of the reactor can occur. The latter implies that the RCS must ultimately be isolatable. Hence, PSRVs must not be exposed to discharge of liquid as a result of the pressurizer reaching a water solid condition.

Option II of the original NSAL 93-013, which references ANSI/ANS-58.8-1984, "Time Response Design Criteria for Nuclear Safety-Related Operator Actions," recommends the use of a less restrictive operator response time. Per ANSI/ANS-58.8-1984, the operator action times for event indication are based on specific time tests. Time test 1 requires 5 minutes and time test 2 requires (1 + N*1) minutes where "N" signifies the number of discrete manipulations required. PORVs would be expected to be available unless they were blocked due to a leaking PORV condition. Operator action associated with assuring PORV availability generally consists of manually opening a block valve to allow it to actuate on demand. An acceptable minimum time to assume initial operator action would therefore be 7 minutes.

Two additional concerns must also be addressed in conjuction with potential water relief through either the PORVs or PSRVs (if qualified for such). The definition of a Condition II incident states that the event at worst "should result in a reactor shutdown with the plant being capable of returning to operation. In order to meet this condition, the piping downsteam of the PSRVs and/or PORVs must be qualified for water relief. Secondly, water relief may result in overpressurizing the Pressurizer Relief Tank (PRT), breaking the rupture disk, and spilling contaminated fluid into containment. Therefore, the radiological consequences of this occurrence must also be evaluated.

To conclude that Standard Review Plan NUREG-0800 is met, it must be demonstrated that 1) the pressurizer does not become water-solid within the minimum allowable operator action time, 2) the PSRVs do not relieve water or that the PSRVs are capable of successfully closing following subcooled water relief, 3) the downstream piping is capable of handling the water discharge flow, and 4) that the radiological consequences of breaking the PRT rupture disk do not violate the applicable offsite dose limits. Water relief through the PORVs is not a concern because the PORV block valves would be available to isolate the PORVs should they fail to close.

## ASSESSMENT OF SAFETY SIGNIFICANCE

The assessments provided in the original NSAL remain valid. Analyses of the Inadvertent ECCS Actuation at Power accident using revised analysis assumptions with the primary emphasis on conservatively demonstrating acceptability with respect to pressurizer filling have been performed. These analyses show a potential for reaching a water-solid condition before the ten (10) minute allowable operator action time typically assumed. Without the appropriate operator action to terminate the ECCS flow prior to reaching a water-solid pressurizer condition, the accident may progress from a Condition II to a more severe Condition III LOCA event as a result of the potential failure of the PSRVs to close after water relief.

Although Westinghouse previously adopted the conservative criterion of preventing the pressurizer from becoming water solid, the acceptability of water leakage from the RCS for Inadvertent Operation of ECCS at Power Condition II events is supported by NUREG-0800 and ANS-51.1/N18.2-1973. To meet the applicable Condition II criteria, the magnitude of any water relief must not exceed that of the normal makeup systems (e.g., operation of the ECCS) and the ability to orderly shutdown the reactor must be maintained. The latter implies that the RCS must ultimately be isolated. Hence, the PSRVs must either not relieve water or must be capable of closing after release of subcooled water.

Without appropriate operator action to terminate safety injection flow prior to reaching a water-solid pressurizer condition, the Inadvertent ECCS Actuation at Power event may progress from a Condition II to a more severe Condition III LOCA event as described above. While this occurrence may result in a violation of one of the applicable licensing basis criteria for a Condition II event it is not considered a significant safety concern. As a LOCA event, discharge of coolant out of the PSRVs and PORVs due to ECCS flow is not significantly adverse relative to other Condition III LOCA events currently analyzed. This is because the pressurizer is located on the hot leg (a hot leg LOCA being less severe than a cold leg LOCA) and because the Inadvertent ECCS Actuation at Power event typically models maximum ECCS flow (to maximize the effects of the initiating event) which is a benefit for LOCA. As such, the Inadvertent ECCS Actuation at Power induced LOCA is bounded by the existing small break LOCA analyses.

Relative to the qualification of the PSRV/PORV downstream piping, it has been demonstrated that the thermal hydraulic loads downstream of these valves, generated for water solid discharge, are bounded by the steam-slug discharge case which was used for the design of the pressure safety and relief system. Therefore, the downstream piping is qualified under the existing design criteria for the water solid discharge event. An evaluation of the radiological consequences has been performed which bounds the Table 1 plants for which the required analysis information is available (i.e., U.S. plants). The radiological releases (offsite doses) resulting from breaking the PRT rupture disk are within acceptable limits.

## NRC AWARENESS/REPORTING CONSIDERATIONS

Westinghouse has determined that this supplemental information does not represent a substantial safety hazard or a failure to comply resulting in a substantial safety hazard. The NRC has not been notified of this issue.

## RECOMMENDED ACTIONS

The recommendations provided in the original issue of NSAL 93-013 remain valid. The purpose of this supplement is to provide additional information related to this issue and specifically note that PDP operation during normal operating conditions will tend to accelerate the event by reducing the time to reaching a pressurizer water solid condition. If water relief from the pressurizer is predicted, the PSRVs and the piping downstream of the PSRVs and PORVs must be qualified for subcooled water relief. Normal operation of a PDP, concurrent with initiation of an Inadvertent ECCS Actuation event, will serve to increase the injection flow by approximately 100 gpm and, without operator action, shorten the time to reaching a pressurizer water solid condition.