Westinghouse Nuclear Safety Advisory Letter

NSAL-93-013
June 30, 1993
and
NSAL-93-013, Suppl 1
October 28, 1994

Westinghouse Energy Systems Business Unit

ESBU

NUCLEAR SAFETY ADVISORY LETTER

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
Basic Component: Transient Accident Analysis	Date: June 30, 1993
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

Westinghouse has discovered that potentially non-conservative assumptions were used in the licensing analysis of the Inadvertent Operation of the ECCS at Power accident. Based on preliminary sensitivity analyses, use of revised assumptions could cause a water solid condition in less than the 10 minutes assumed for operator action time. If the PORVs were blocked, the PSRVs would relieve water and potentially cause the accident to degrade from a Condition II incident to Condition III incident without other incidents occurring independently. Per ANS-051.1/N18.2-1973, a Condition II event cannot generate a more serious event of the Condition III or IV type without other incidents occurring independently.

Westinghouse is unable to determine whether a defect causing a substantial safety hazard or a failure to comply resulting in a substantial safety hazard exists because sufficient plant specific information is not available. Under 10 CFR 21.21(b), if Westinghouse determines that there is insufficient information available to provide the capability to perform an evaluation, then Westinghouse must inform affected licensees of this determination.

Additional information, if required, may be obtained from the originator. Telephone 412-374-4302.

Originator(s): *Gary G. Ament*
G. G. Ament

K. J. Vavrek
K. J. Vavrek

H. A. Sepp
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

The Inadvertent Actuation of the Emergency Core Cooling System (ECCS) accident (also referred to as the Spurious SI event) is 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.

Standard Review Plan NUREG-0800, Rev. 1, Section 15.5.1, "Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory," states that to meet the requirements of GDC 10, 15, and 26 for incidents of moderate frequency an incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. To address this, Westinghouse adopted the following criterion:

> 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.

The basis for demonstrating that the pressurizer will not become water solid is to preclude the possibility of discharging primary coolant through the Power Operated Relief Valves (PORVs) and/or the Pressurizer Safety Relief Valves (PSRVs), causing the incident to progress from one of moderate frequency to an infrequent small break LOCA incident. A small break LOCA condition could result from failure of the PSRVs to close after discharging water since the PSRVs were typically not designed for water relief.

Based on a review of the analysis methods used to evaluate this accident, it was discovered that these methods were developed with the primary emphasis on criteria for maintaining RCS pressure below the design value and ensuring that fuel cladding integrity is maintained. These methods did not emphasize the criterion for preventing the pressurizer from becoming water solid within the allowable operator action time. Sensitivity analyses performed for this accident have shown that some analysis assumptions are non-conservative with respect to maximizing the potential for pressurizer filling. Revised analysis assumptions that conservatively consider the potential for pressurizer filling for the Inadvertent Operation of the ECCS at Power accident have been found to have a significant effect on the rate at which the pressurizer water volume increases.

TECHNICAL EVALUATION

Westinghouse has performed preliminary sensitivity analyses that indicates for some plant specific applications using revised assumptions the pressurizer can become water solid in less than 10 minutes. To conclude that Standard Review Plan NUREG-0800 is met, it must be demonstrated that the pressurizer does not become water-solid in the minimum allowable operator action time, that the PSRVs do not open, or that the PSRVs are capable of successfully closing following water relief. If ECCS flow is not terminated before water is discharged through the PSRVs, it cannot be demonstrated without plant specific PSRV operability assessments that this accident does not lead to a more serious plant condition. Water relief through the PORVs is not a concern, because the PORV block valves can be used to isolate the PORVs if they fail to close. If ECCS flow is not terminated before the pressurizer becomes water solid and water is discharged through the PSRVs, it can not be demonstrated that this accident does not lead to a more serious Condition III LOCA event.

TECHNICAL EVALUATION (con't)
The analysis for licensing basis assumed maximum ECCS flow which typically includes an additional 5 to 10 percent margin on discharge pressure above the vendor's specified pump performance. The licensing basis analysis assumed that the PORVs, the pressurizer water level control system, the steam dump system and the steam generator PORVs were not available to help mitigate this accident since they are considered to be control grade functions. Also, no credit was taken for letdown since it is isolated following a safety injection signal for those plants which use charging pumps for high head safety injection pumps.

ASSESSMENT OF SAFETY SIGNIFICANCE
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 allowable operator action time. 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 failure of the PSRVs due to water relief through the valves.

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 Condition II events is supported by NUREG-0800 and ANS-051.1. 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 this is the cause of the water relief) 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 open or must be capable of closing after release of subcooled water.

NRC AWARENESS/REPORTING CONSIDERATIONS
Westinghouse is unable to determine if this issue would cause a substantial safety hazard or a failure to comply resulting in a substantial safety hazard because sufficient plant specific information is not available. This information is being transferred to the applicable plants pursuant to 10 CFR 21.21(b). The NRC has not been notified of this issue.

RECOMMENDED ACTIONS

1. Licensees should first determine if their current licensing basis requires them to analyze the Inadvertent Operation of the ECCS at Power accident. If this accident is not included within their current licensing basis, no additional action is required.

2. Licensees should determine if their Pressurizer Safety Relief Valves are capable of closing following discharge of subcooled water. If the PSRVs were designed or qualified to relieve subcooled water, the Inadvertent ECCS Actuation at Power accident will not degrade into a more serious Condition III event, since these valves will close once ECCS flow has been terminated. It should be noted that the licensees may have qualified these valves in compliance to NUREG-0737, Item II.D.1.

3. If the PSRVs are not designed or qualified for subcooled water relief, the licensees should re-evaluate the Inadvertent ECCS Actuation at Power accident using one or a combination of the following options.

RECOMMENDED ACTIONS (con't)

Option I: Reduce the maximum ECCS flow used in the safety analysis. Preliminary sensitivity analyses have shown that using less conservative flow may sufficiently delay filling the pressurizer such that the operator action to terminate the accident can be successfully credited.

Option II: Use a less restrictive operator response time. Per ANSI/ANS-58.8-1992, "Time response design criteria for safety-related operator actions," credit can be taken in the analysis for the operator to stop one pump at 7 minutes, a second pump at 8 minutes, and depending on the plant specific design, the third at 9 minutes. Preliminary sensitivity analyses have shown that using these less restrictive operator action times may sufficiently delay or prevent filling the pressurizer.

Option III: Credit the use of one or more PORVs to help mitigate the accident. Preliminary sensitivity analyses have shown that if a water solid pressurizer condition is reached, one PORV should be sufficient to maintain pressure below the PSRV setpoints and prevent discharge of water through the pressurizer safety relief valves. To credit this option, the licensee would have to ensure that at least one PORV is always available (PORV block valve is opened). This option could also be credited if the PORVs are blocked by ensuring that the Emergency Operating Procedures (EOPS) instruct the operators to open at least one PORV block valve before the PORV setpoint is reached. Use of this option may require a change to the plant EOPS and/or the plant technical specifications to ensure that at least one PORV is available since most technical specifications currently allow the PORVs to be isolated during power operation.

Westinghouse
Energy
Systems
Business
Unit

NUCLEAR SAFETY ADVISORY LETTER

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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

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Strategic Licensing Issues

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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
Maanshan 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.