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October 5, 1979

2-109-5

Mr. H. R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Arkansas Nuclear One-Unit 2
Docket No. 50-368
License No. NPF-6
Interaction Between Non-Safety Grade Systems and Safety Grade Systems
(File: 2-1510)

Gentlemen:

This letter responds to your September 17, 1979, letter on the subject of a "potential unreviewed safety question on interaction between non-safety grade systems and safety grade systems". This potential problem was further addressed in IE Information Notice 79-22, issued September 14, 1979.

In conjunction with Combustion Engineering we have reviewed the specific non-safety grade systems listed in IE Information Notice 79-22, as well as others, for potential interactions that could constitute a substantial safety hazard. We have not been able to identify such an interaction. While, in some cases, we have identified potential variations from the FSAR Licensing bases, the basic conclusion of the FSAR, that these events do no constitute an undue risk to the health and safety of the public, remains unchanged.

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In our preliminary screening for potential adverse environmentally-induced failures of non-safety grade equipment, it appears that potential problems arise only when such failures are combined with other failures or operator errors. The probability of the type of high energy pipe breaks we are considering is small. Such breaks are also more likely to be small cracks rather than abrupt failures so that the resulting adverse environment builds up over a period of time providing the potential for detection prior to component failure. Additionally, our review recognized the

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difference between a demonstrated deficiency (e.g., determination that a control component would operate in a fashion not within the limits presented in the safety analysis) and a potential unreviewed question.

As previously stated, we have not identified any events that would change the conclusions of the FSAR; namely that these events do not constitute an undue risk to the safety and health of the public.

As you must recognize, our investigation within the limited time frame required by your September 17 letter must be considered preliminary and could not include detailed analyses. Based on our preliminary investigation we are convinced that continued operation is warranted.

Justification for continued operation of ANO-2 is provided in Attachments 1 through 4. Attachment 1 summarizes Combustion Engineering's efforts to date. Attachment 2 is the event/interaction matrix generated by Combustion Engineering. Attachment 3 details generic event/interaction scenarios identified by Combustion Engineering. Attachment 4 supplements Attachment 3 and provides further plant specific information.

In addition Nuclear Safety Analysis Center (NSAC) has determined that the probability of severe consequences resulting from one of these high energy pipe breaks is very low for a typical nuclear power plant. The probabilistic analysis of IE Information Notice scenarios was prepared by NSAC as a result of AIF's promotion of an industry wide generic response to this concern. We have participated with AIF to the extent possible. The above referenced probability analysis will be submitted to you by NSAC later this week.

As a result of the Three Mile Island accident, there are a significant number of industry, governmental and regulatory investigations under way examining that licensing bases and the operating procedures of nuclear generating facilities. These investigations are already identifying areas where studies may result in the consideration of new or revised events as part of the bases for assuring the continued safety of nuclear plants. NUREG-0578 outlines several such events and suggests remedies.

NUREG-0578 requirements for analyses of potential safety problems envision the kinds of scenarios identified by Westinghouse and made the subject of IE Information Notice 79-22. Section 3.2 of NUREG-0578, Page 17 states in part,

"...The NRC requirements for non-safety systems are generally limited to assuring that they do not adversely affect the operation of safety systems.."

Further, on Page A-45 of NUREG-0578,

"Consequential failures shall also be considered..."

We, therefore, believe that the scope of the action required by IE Information Notice 79-22 is consistent with the requirements of NUREG-0578 and should therefore be integrated with the planned response sequence for compliance with the NUREG.

We are aware of the need to establish a priority of consideration of new issues based upon their potential impact upon the health and safety of the public. Such a priority is required so that the resources of skilled engineers and analysts can be applied to the review of the most important concerns first.

Very truly yours,

W. Cavanaugh, III

WC/MOW/ew

Attachments

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STATE OF ARKANSAS)
) SS
COUNTY OF PULASKI)

William Cavanaugh III, being duly sworn, states that he is Vice President, Generation & Construction, for Arkansas Power & Light Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this Supplementary Information; that he has reviewed or caused to have reviewed all of the statements contained in such information, and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information and belief.

William Cavanaugh III

SUBSCRIBED AND SWORN TO before me, a Notary Public in and for the County and State above named, this 8th day of October 1979.

Sharon Kaye Hendrix
Notary Public

My Commission Expires:

My Commission Expires 9/1/81

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DESCRIPTION OF C-E'S EFFORTS

At the request of the C-E Owners Group on Post-TMI Efforts, Combustion Engineering conducted a review of potential control systems interactions during high energy pipe break events. C-E initially established a matrix of high energy pipe break events and control functions within C-E's ability to properly evaluate. A list of separate systems that should also be considered was developed and forwarded to the utilities participating in this effort along with a list of control function and events under consideration by C-E. These lists are attached.

In the time available, C-E reduced this matrix to include only those systems and events which required further evaluation. Some of these events were further eliminated by individual utilities on a plant specific basis. A general description of the procedure used by C-E to reduce this matrix is listed below.

I. An initial review of each postulated Control Function failure for each pipe break was completed and served as the basis for consideration. Where a postulated failure could potentially increase the severity of a high energy pipe break, the following criteria were employed to resolve the concern:

1. Is the postulated Control Function failure mode credible?

2. Is the Control Function Equipment (Sensor, Cables, etc.) in a location which could be impacted by the environment?

3. Is the Control Function Equipment (Sensor, Cable, etc.) qualified to operate properly in the postulated environment?

4. Where the postulated Control Function failure is credible, could its impact potentially affect the conclusions presented in the SAR? Considerations such as Maximum Control Function capabilities, and delayed, but proper operator action were employed in this effort.

It should be noted that the limited time available did not allow for extensive analysis. Prudent engineering judgement was utilized to eliminate those events/interactions which did not change the conclusions of SAR analyses.

Extensive evaluations involving the Auxiliary Feedwater system and other long term cooling mechanisms have not been performed. Auxiliary feedwater is being evaluated under Bulletins and Orders and Lessons Learned (NUREG 0578). This decision was made in order to concentrate on those items felt to be of greater significance in the short time available for assessment of control system high energy pipe break interactions..

In several cases, most notibly the PORV failure in the open position, no specific failure mechanism has been identified. The only manner for such

a failure to occur would be for power to be inadvertently applied to the valve solenoid and not be removed. Part of C-E's short term recommendations are for utilities to evaluate whether or not a failure mechanism of this type is credible.

The potential adverse impact of high energy pipe breaks on reactor coolant pumps was considered. Both the seized shaft and the simultaneous three or four pump loss of flow were eliminated from consideration based on judgement that these failures are not considered credible within the time frame limited by operator action (30 minutes) due to environmental impact alone. The impact of other potential loss of flow events (e.g. one or two pump loss of flow) during high energy pipe breaks was reviewed and it was judged that the resulting rapid reactor trip was sufficient to ensure that the conclusions of the SAR would not change.

Each Utility further evaluated and eliminated items from the matrix (Attachment 2) based on current operating procedures or specific equipment configurations, locations, or levels of qualification.

Attachment 3 details potential event/interaction scenarios.

AP&L has not identified any changes that need to be made to emergency procedures as a result of our preliminary investigation.

MATRIX OF EVENTS/CONTROL FUNCTIONS
FOR FURTHER CONSIDERATION AND ACTION

Pipe Break / Control Function	SLB	FWLB	CEA Ejection	SBLOCA	LBLOCA
Pressurizer Level		X			
Pressurizer Pressure					
Pilot Operated Relief Valves	X	X			
CEA Position	X	X	X	X	
Feedwater Flow	X	X			
Boron Concentration					
Turbine Control	X				
Steam Bypass	X				
Steam Dump Upstream of MSIV	X	X			
Steam Dump Downstream of MSIV	X				
Steam Gen. Blowdown					
Condenser					
Reactor Coolant Flow					

DESCRIPTION OF REMAINING EVENTS AND CONTROL FUNCTIONS

I. Assessment of Control System Failures on Steam Line Break Event

A. Sequence of Events for Generic SAR Steam Line Break at Full Power, Inside or Outside Containment

1. Double-ended steam line break occurs
2. Reactor trip on low steam generator pressure
3. MSIS initiates to isolate the steam generators
4. RCS temperature decreases due to excessive steam removal
5. Total reactivity increases due to moderator cooldown effect
6. MSIVs close
7. Pressurizer empties
8. Low pressurizer pressure initiates SIAS
9. MFIVs close
10. Safety injection boron reaches core
11. Affected steam generator empties, terminating cooldown effect, the transient reactivity reaches peak and decreases gradually due to boron injection
12. Limited or no post-trip return-to-power
13. No fuel in DNB

B. Steam Line Break With PORV Control System Failure

1. Significant Interaction Effects:
   a. Increased Containment Pressure
   b. A stuck open PORV in combination with a steam line break has not been analyzed.

2. Assumptions
   a. Steam line break (large break inside containment for Item 1.A above, any size or location for Item 1.B above).
   b. Inadvertently PORVs open and remain open
   c. PORV Block valve also fails to close when required
   d. Initial condition: full power

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3. It must be emphasized that no mechanism has been identified for the PORV to inadvertently open and remain open since its signal to open comes from safety grade equipment and the Dresser valves and solenoids are qualified for an environment in excess of 400°F.

4. Sequence of Events

a. Large steam line break occurs inside containment.

b. Reactor trip occurs on steam generator low pressure within 5 seconds.

c. Should the adverse environment cause the PORV to inadvertently open and then remain open, the following steps may also occur. It should be noted that no mechanism has been identified which would cause this to occur.

d. Steam from PORV fills quench tank and bursts rupture disk releasing steam to the containment and causing additional containment pressurization.

e. Mass removal via PORV causes additional void formation within the reactor coolant system.

C. Steam Line Break With Feedwater Flow Control System Failure

1. Significant Interaction Effects

   a. Steam generator filling - causing potential piping structural problems

2. Assumptions

   a. Small steam line break inside containment that does not cause an immediate reactor trip

   b. Feedwater flow exceeds steam flow due to failure of steam generator level instrument, indicating flow

   c. SAR conservatism

      i. no operator action within 30 minutes

3. Sequence of Events

   a. Small steam line break occurs which does not cause an immediate reactor trip

   b. Steam generator level instrument fails, causing an increase of feedwater flow in excess of steam flow

   c. Steam generator begins to fill causing increased moisture content of steam

   d. If no operator action occurs undefined piping structural problems could result

   e. It should be emphasized that this event can be prevented by prompt operator action. Safety grade steam generator level instrumentation exists, enabling comparison with control grade level instruments of the feed system

POOR ORIGINAL

D. Steam Line Break With Failure of Main Steam Paths Downstream of MSIV's

1. Significant Interaction Effects

   a. Increase post-trip return-to-power

2. Assumptions

   a. Large steam line break inside containment

   b. MSIV on unaffected steam generator fails to close. This sequence of events is pertinent only if this assumption is made.

   c. Downstream of MSIV's main steam paths fail open

   d. Initial condition: full power

   e. SAR conservatisms

      i. end of cycle core

      ii. the most reactive CEA stuck out

      iii. steam blowdown through steam line break

3. The number of failures which must occur during this event are significant. First there must be the large break. Then the MSIV on the opposite steam generator must fail to close. There is a stuck rod on reactor trip. Then steam paths downstream of the MSIV's must be affected. These include turbine control valves and steam dump and bypass valves. The probability of this event occurring is much less than 10^{-6} per reactor year.

4. Sequence of Events

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   a. Large steam line break inside containment

   b. Reactor trips on low steam generator pressure trip signal

   c. MSIV on unaffected steam generator fails to close on MSIS

   d. Main steam paths downstream of MSIV open or fail to close due to control system malfunction caused by adverse environment following large steam line break.

   e. Open main steam paths increase the steam blowdown and increase moderator cooldown effect which adds positive reactivity to core. A post-trip return-to-power is more severe under these conditions.

POOR ORIGINAL

E. Steam Line Break With Atmospheric Dump Valve Control System Failure

1. Significant Interaction

   a. Post-accident controlled cooldown

2. Assumptions

   a. Steam line break outside containment and upstream of MSIV

   b. Atmospheric dump valves on opposite steam line open and remain open*

   c. SAR conservatism

      i. no operator action within 30 minutes

3. Sequence of Events

   a. A steam line break outside of containment but upstream of the MSIV occurs

   b. Reactor trip on low steam generator pressure

   c. Atmospheric dump valves upstream of MSIV's open and remain open due to control system failure

* The failure mechanism identified is a failure of the input signals that would cause the valve to open if operating in the automatic mode. Although no operator action is assumed for 30 minutes prompt operator action to shut the open valve would mitigate any effects of this event.

d. If no operator action takes place there would be the potential for dry-out and depressurization of both steam generators

e. Failure to shut atmospheric dump valves could inhibit a controlled plant cooldown by limiting the ability of the auxiliary feed pumps to deliver to the steam generator(s)

POOR ORIGINAL

II. Assessment of Impact of Control System Failures on Feed Line Break Event and CEA Ejection

A. SAR Feed Line Break

1. Sequence of Events

a. Main feed line break occurs downstream of reverse flow check valve, discharging main feed and steam generator fluid

b. RCS heatup due to loss of subcooled feed flow

c. Reactor trip occurs on steam generator low water level or high pressurizer pressure. Turbine trip occurs on reactor trip

d. Rapid RCS heatup and pressurization due loss of heat transfer as the ruptured steam generator empties

e. Depressurization of the ruptured steam generator initiates MSIS and isolates the intact generator

f. RCS pressurization terminates with opening of primary relief/safety valves and decreasing core heat flux

g. RCS cooldown begins, controlled by the main steam safety valves

h. Auxiliary feed is initiated automatically or by operator action

B. Feed Line Break With RCS Inventory Control Failure

1. Significant Interaction Effect

   a. Increased RCS pressurization due to liquid filled pressurizer

2. Assumptions

   a. Small feed line break inside containment

   b. Adverse environment impacts pressurizer level instrument causing indication to fail low which causes the control system to increase inventory (and pressurizer level)

   c. Initial conditions

      i. 102% power

      ii. steam bypass control system in manual mode

      iii. beginning-of-cycle core parameters

   d. Analysis conservatisms

      i. no operator action for at least 30 minutes

      ii. no credit for steam generator low water level trip in ruptured unit until empty

      iii. heat transfer in ruptured steam generator instantaneously terminated on emptying

      iv. failure of the feed line reverse flow check valve, if the break occurs upstream of the valve

3. Sequence of Events

   a. Feed line break in containment

   b. Main feed spills from break

   c. Adverse containment environment causes pressurizer level indication to fail low causing RCS inventory to increase

   d. Reactor trip occurs on steam generator low water level on high pressurizer pressure. Turbine trips on reactor trip

   e. RCS heatup results from rapid decrease in SG heat transfer due to loss of fluid from the ruptured steam generator

   f. Pressurizer relief and/or safety valves open

g. Potential for pressurizer to fill with liquid exists due to high level in pressurizer prior to heatup. Relief/safety valve relief capacity reduced by liquid discharge

h. Extent of increased RCS pressurization is dependent on time of pressurizer filling relative to the rapid heatup

C. Feed Line Break With PORV Control Failure

1. Significant Interaction Effects

a. A failed open PORV in combination with a feed line break has not been analyzed

2. Assumptions

a. Feed line break inside containment

b. PORV's inadvertently open and remain open

c. PORV block valve also fails to close when required

d. No operator action until 20 minutes

3. PORV would not be expected to remain open due to actuation malfunction since Dresser valves and solenoids are qualified for temperatures in excess of 400°F

4. Sequence of Events

a. Feed line break occurs inside containment

b. Steam generator fluid and/or main feed spill from break

c. RCS heatup and pressurization results from loss of feed flow

d. PORV opens on high pressure and fails to reclose due to adverse environment

e. Reactor trip occurs on high pressurizer pressure. Turbine trips on reactor trip

f. RCS depressurization occurs if PORV's fail to reclose

g. Mass removal via PORV causes void formation within RCS

h. Feed line break in combination with a failed open PORV has not been analyzed

D. Feed Line Break With Feedwater Control Failure

1. Significant Interaction Effects

a. Overfilling of the steam generator(s) causing potential structural problems

2. Assumptions

a. Small feed line break inside containment

b. Feed control in automatic mode

c. Adverse environment causes steam generator level indication to fail low which causes the feed control system to increase feed flow above the steam flow

d. No operator action for 30 minutes

3. Sequence of Events

a. A small feed line break occurs inside containment

b. Main feed spills from break

c. Steam generator level instrument fails indicating low and causes increased feed flow in excess of steam flow

d. Steam generator begins to fill causing increased moisture content of steam

e. If no operator action occurs undefined structural problems could result

f. It should be emphasized that this event can be prevented by prompt operator action. Safety grade level instrumentation exists to compare to control grade instruments. The feed system can then be controlled manually

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E. Feed line Break with Atmospheric Steam Dump Control Failure

1. Significant Interaction Effects

a. Controlled plant cooldown

2. Assumptions

a. Feed line break outside containment and downstream of reverse flow check valve

b. Adverse environment impacts the atmospheric steam dump control on unaffected steam generator causing an uncontrolled steam release upstream of the MSIV's

c. No operator action until 30 minutes.*

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* The failure mechanism identified is a failure of the input signals that would cause the valve to open if operating in the automatic mode. Although no operator action is assumed for 30 minutes, prompt operator action to shut the open valve would mitigate any effects of this event.

3. Sequence of Events

a. Feed line break occurs outside of containment downstream of check valve

b. Steam generator fluid and/or main feed spill from break

c. Reactor trip occurs on steam generator low water level or high pressurizer pressure. Turbine trip occurs on reactor trip

d. Steam generator pressure increases following turbine trip

e. Environment could cause atmospheric dump valves upstream of MSIV in unaffected steam generator to open and remain open

f. If no operator action takes place there would be a potential for dry out and depressurization of both steam generators

g. Depressurization of both steam generators may limit the ability of the auxiliary feed pumps to deliver to the steam generator(s).

F. CEA Ejection With Failure of FWCS

A feedwater control system malfunction during a CEA ejection could produce effects similar to those described for the other events in this section.

III. Potential Effect of Reactor Regulating System During High Energy Pipe Break Events

A. CEA position malfunctions due to steam and feedline breaks and CEA ejection

1. Significant interaction effect:

a. Potentially higher reactor power levels prior to reactor trip than presently analyzed

2. Assumptions

a. Small high energy pipe break inside containment

b. Reactor regulating system in automatic mode

c. Adverse environment results in a low indicated power level from the ex-core sensor input to the Reactor Regulating System causing CEAs to be withdrawn

3. Sequence of events

a. High energy pipe break inside containment of a small enough size where immediate reactor trip does not occur

b. Control grade ex-core sensor indication fails low due to adverse environmental impact

c. Reactor regulating system causes CEAs to be withdrawn

d. Reactor power exceeds the power previously assumed during the transient

e. Reactor trip occurs due to high energy pipe break at conditions not considered in present analyses

B. Small Break LOCA With CEA Control System Malfunction.

1. Significant interaction effects

   a. Potential exists for increasing power. This would cause pressure to remain above low pressurizer pressure trip for a longer period than previously assumed

2. Assumptions

   a. Small break LOCA inside containment

   b. CEA control system in automatic mode

   c. Adverse environment impacts CEA control system or related sensors resulting in consequential failure

   d. Control system causes CEA to withdraw

   e. Standard LOCA licensing assumptions

3. Sequence of events

   a. Small break LOCA occurs inside containment

   b. CEA control system in automatic mode

   c. Adverse environment caused by rupture potentially causes excore power indication to indicate low power level

   d. Should CEAs begin to withdraw, the magnitude of the overpower excursion prior to scram would be increased. This could produce a higher primary system pressure which could then delay reactor trip and SIAS and result in higher peak clad temperatures

Attachment 4

ARKANSAS NUCLEAR ONE-UNIT 2
PLANT SPECIFIC INFORMATION

The analyses provided by Combustion Engineering (Attachment 3) apply to ANO-2 except as noted in this Attachment. Due to plant specific operating procedures and/or equipment configurations we have been able to eliminate some items identified in Attachments 2 and 3. Item numbers in this Attachment match those in Attachment 3. All interactions identified in Attachment 2 will be addressed.

Item I.b - Steam Line Break With PORV Control System Failure

There is no PORV installed on the ANO-2 pressurizer. This precludes any interaction in this case.

ITEM I.c - Steam Line Break with Feedwater Control System Failure

The ANO-2 Reactor Protective System includes a high steam generator level trip and its accompanying alarm would alert the operator of a high steam generator level condition. He could then look at his safety grade steam generator level indication and manually control feedwater. Additionally, if steam generator level were allowed to increase beyond the point initiating a reactor trip, steam pressure could be expected to decrease as the generator approached a filled condition. The decreasing steam pressure would initiate a safety grade main steam isolation actuation signal which secures normal feedwater to the steam generators. The ANO-2 operators have traditionally been trained to react to feedwater control system malfunction such as this and, since warning is provided, could be expected to properly respond to this casualty.

Item I.d - Steam Line Break With Failure of Main Steam Paths Downstream of MSIV's

As stated in the assumptions listed in Item I.D. 2 of Attachment 3, this sequence of events is only a problem if it is assumed that the Main Steam Isolation Valve (MSIV) on the unaffected steam generator fails to close. The ANO-2 FSAR (Table 10.3-4) states that a failure of a MSIV to close is very improbable because:

(1) Redundant solenoid valves are provided on the air supply and exhaust lines to the air cylinders on each of the MSIV's. Redundant power supplies and

MSIS signals to these solenoid valves preclude the possibility of a single electrical failure resulting in the failure of the MSIV to close.

(2) The MSIV's are designed such that they will close with the air cylinder only or the springs only.

The reliability of these valves precludes this postulated interaction.

Item I.e - Steam Line Break With Atmospheric Dump Valve Control System Failure

As stated in the assumptions of Attachment 3 for this interaction, this sequence of events is only a problem if the atmospheric dump valves on the opposite steam line open and remain open.

In this event, a Main Steam Isolation Signal (MSIS) would be initiated. The atmospheric dump valves upstream of the MSIV's receive this signal (MSIS) and close. These valves are "Q" valves and their actuating devices are redundant, safety grade. The reliability of these valves precludes this postulated interaction.

Item II. B - Feed Line Break With RCS Inventory Control Failure

The ANO-2 Pressurizer level indication system is safety grade. This precludes this interaction.

Item II. C - Feed Line Break With PORV Control Failure

There is no PORV installed on the ANO-2 pressurizer. This precludes any interaction in this case.

Item II. D - Feed Line Break With Feedwater Control Failure

See response to item I.c.

Item II. E - Feed Line Break With Atmospheric Pump Control Failure

See response as Item I.E.

Item III. Potential Effect of Reactor Regulations System During High Energy Pipe Break Events.

The ANO-2 reactor regulating system does not have an automatic CEA withdrawal feature. This precludes any possible interaction.