bcc: w/encl.
JRS
JDL
DMC
RSS/V
SHHanauer, DSS
MAycock, NRR
RBrady, SEC
JBecker, ELD
GGower, IE

JAN 11 1979

MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

FROM: Joseph D. Lafleur, Jr., Deputy Director
Office of International Programs

SUBJECT: UK VISIT REQUEST (JANUARY 30, 1979)

The U.K. Nuclear Installations Inspectorate (NII) has requested NRC appointments for Dr. D. L. Reed of the NII during the week beginning January 29 of February 12, 1979, to discuss the following aspects of ATWS in PWRs and BWRs:

1. The basic requirements for ATWT studies in the licensing procedure in the U.S. (Dr. Reed will discuss the U.K. requirements).

2. NUREG-0460 in detail, referring in particular to comments made in separate correspondence (see attached).

3. Areas of greatest uncertainty in the methods of analysis and data used for ATWT studies.

4. The possibility of setting up benchmark calculations which compare methods of analysis used in the U.S. and Europe for ATWT.

If acceptable, I suggest scheduling a one-day session on Tuesday, January 30, 1979, beginning at 9:00 a.m. in IP Conference Room 8313 (MNBB), which has been reserved for your use.

Please advise Bob Senseney (492-7788) of this office whether NRR can accommodate this request and, if so, of the name(s) of the staff member(s) who will be involved. You should also indicate if you believe a second day will be necessary to adequately cover the topics. The February date may, of course, be chosen if preferred.

/S/
Joseph D. Lafleur, Jr.
Deputy Director
Office of International Programs

8104170315

Enclosure:
As stated

OFFICE	IP	IP	IP
SURNAME	RSenseney:ecb	DMChenier	JDLafleur
DATE	1/11/79	1/11/79	1/11/79

(1)

THANK YOU FOR PROVIDING US WITH A COPY OF NUREG - 0460 ''ANTICIPATED TRANSIENTS WITHOUT SCRAM FOR LIGHT WATER REACTORS''. WE HAVE READ THE DOCUMENT AND FIND IT USEFUL IN FORMING OUR OWN VIEW ON THESE TYPES OF FAULT IN LIGHT WATER REACTORS. OUR EXPERIENCE IN THIS FIELD HAS BEEN CONFINED TO REVIEWING THE GENERIC SAFETY OF A WESTINGHOUSE AND KWU PWR DESIGNS. HENCE OUR REVIEW OF THESE DESIGNS IS NOT AS WIDE AS PRESENTED IN NUREG - 0460 BUT OUR CONCLUSIONS ARE SIMILAR. HOWEVER THERE ARE A NUMBER OF POINTS DETAILED BELOW ON WHICH WE DIFFER AND WISH TO DISCUSS WITH YOU CONCERNED WITH THE PHILOSOPHICAL APPROACH TO THE ASSESSMENTS AND THE ANALYSIS OF ATWS FAULTS. BEFORE DISCUSSING NUREG - 0460 DETAILS OF OUR CONCLUSIONS IN OUR ASSESSMENT OF THE WESTINGHOUSE TROJAN PWR DESIGN WILL BE GIVEN.

NII REQUIREMENTS BASED ON THE ASSESSMENT OF THE WESTINGHOUSE ATWS

. WITH NO CHANGES TO THE PLANT (SUCH AS INCREASE IN THE NUMBER OF SAFETY VALVES) WE REQUIRE THAT THE BEST ESTIMATE TEMPERATURE COEFFICIENT IS MORE NEGATIVE THAN -7 PCM/O F WHEN THE REACTOR IS AT FULL POWER. FROM THE EVIDENCE PROVIDED THE TEMPERATURE COEFFICIENT WAS SMALLER THAN -7 PCM/O F AT THE BEGINNING OF LIFE. WE NOTED HOWEVER THAT OPERATION AT LOW POWER PRODUCES A COEFFICIENT OF -5 PCM/O F AFTER A FEW HOURS WHICH GOES PART THE WAY TO ACHIEVING OUR REQUIREMENTS. OPERATION AT LOW POWER WOULD REQUIRE ADMINISTRATIVE PROCEDURES AND THEREFORE IS NOT AN ENTIRELY SATISFACTORY WAY OF ACHIEVING OUR REQUIREMENTS.

ALTERNATIVELY INCLUSION OF MORE SAFETY VALVES ON A PRESSURIZER WOULD REDUCE THE PEAK PRESSURE IN AN ATWS AND WOULD OFF-SET THE ABOVE REQUIREMENTS. INSERTION OF BURNABLE POISON IN THE FUEL AT THE BEGINNING OF LIFE WOULD REDUCE THE BORON REQUIREMENT IN THE MODERATOR AND HENCE INCREASE THE SIZE OF THE MODERATOR TEMPERATURE COEFFICIENT SO AGAIN REDUCING THE ABOVE REQUIREMENTS, BUT THIS APPROACH REDUCES THE ECONOMIC PERFORMANCE OF THE PLANT. DESPITE THESE PROBLEMS WE BELIEVE THAT STEPS SUCH AS ABOVE CAN BE TAKEN TO PROVIDE AN ADEQUATE MODERATOR TEMPERATURE COEFFICIENT AT ALL TIMES DURING FULL POWER OPERATION OF THE PLANT TO COMPENSATE FOR AN ATWS.

100%

WATER RELIEF RATES FOR PRESSURIZER RELIEF AND SAFETY VALVES

DESPITE THE CONSERVATIVE DATA USED IN THE ANALYSIS I.E RELIEF RATES DEDUCED FROM THE HOMOGENEOUS EQUILIBRIUM MODEL MULTIPLIED BY A FACTOR OF 0.9, WE WISH TO SEE TESTS OF WATER RELIEF RATES WITH THESE VALVES OVER THE RANGE OF PRESSURES AND TEMPERATURES EXPECTED IN AN ATWS SITUATION. THESE TESTS WOULD NOT ONLY DETERMINE THE PERFORMANCE OF THE VALVES BUT WOULD ALSO PROVIDE EVIDENCE OF MALFUNCTION.

TURBINE TRIP

How?

WE WISH TO SEE THE RELIABILITY OF THE TURBINE TRIP IMPROVED SINCE FAILURE TO TRIP WOULD INCREASE THE PEAK PRESSURES BEYOND THE LIMIT OF 3,200 PSIG FOR A NUMBER OF ATWS FAULTS.

METHODS OF ANALYSIS

THE EFFECT OF DATA UNCERTAINTIES SHOULD BE INCLUDED IN THE ANALYSIS. COMPARISIONS WITH OTHER CODES ARE REQUIRED TO DETERMINE THE UNCERTAINTIES IN THE METHOD OF ANALYSIS AND PHYSICAL MODELS. IN PARTICULAR WE WERE CONCERNED AS TO THE ADEQUACY OF THE MODELS USED FOR THE HEAT EXCHANGER AND THE PRESSURIZER.

LONG-TERM SHUTDOWN EFFECTS

ACCORDING TO OUR CRITERIA NO OPERATOR INTERVENTION WILL BE MADE FOR 30 MINUTES. HENCE WE REQUIRE THAT THE ANALYSIS SHOULD BE EXTENDED TO 30 MINUTES FOR EACH TRANSIENT IN ORDER TO SHOW THAT THE PLANT REMAINS SHUTDOWN DURING THIS PERIOD OF TIME.

COMMENTS ON NUREG - 0460

THIS SECTION IS DIVIDED UP INTO TWO PARTS, THE FIRST DEALING WITH THE BASIC REQUIREMENTS AND THE SECOND WITH MORE DETAILS OF THE ASSESSMENT.

A. BASIC REQUIREMENTS

Agrees with US

1. THE EXISTING NON-DIVERSE TRIP SYSTEM CANNOT BE CLAIMED TO BE MUCH BETTER THAN 10-4 FAILURES PER DEMAND. ADDITIONAL PROTECTION IS THEREFORE REQUIRED TO DEAL WITH ALL FREQUENT FAULTS (THIS IS MORE DEMANDING THAN WCAP 8330).

VERY TOUGH

2. THE MAXIMUM CIRCUIT PRESSURE FOR ANY ATWS SEQUENCE WITH A POSTULATED FREQUENCY HIGHER THAN 10-7 PER ANNUM SHALL NOT EXCEED 3,200 PSIG CORRESPONDING TO A CHANCE FAILURE OF 10-3 PER EVENT. ADDITIONALLY FUEL DAMAGE OR OTHER EFFECTS SHALL NOT EXCEED THAT WHICH GIVE RISE TO A RELEASE EQUIVALENT TO 1 ERL. AT 10^{-7}!

3. THE ADDITIONAL PROTECTION IMPLIED IN WCAP 8330 IS ACCEPTABLE PROVIDING THE FOLLOWING ARE COVERED.

3.1 ALL SUCH EQUIPMENT SHALL BE CALSSIFIED AS PROTECTION EQUIPMENT.

3.2 EQUIPMENT SHALL MEET THE SINGLE FAILURE CRITERION AND HAVE A RELIABILITY OF THE ORDER OF 10-3 PER DEMAND ALLOWANCES BEING MADE FOR THE FREQUENCY OF EACH INITIATING EVENT.

3.3 THE EQUIPMENT SHALL BE INDEPENDENT AND DIVERSE FROM THAT PROVIDED IN THE EXISTING TRIP FUNCTION.

4. NO OPERATOR ACTION SHALL BE REQUIRED FOR RECOVERY WITHIN 30 MINUTES OF THE INITIAL EVENT. THIS IMPLIES A RESTRICTION ON MODERATOR TEMPERATURE COEFFICIENT.

Nearly a 100% MTC!

5. THE MODERATOR TEMPERATURE COEFFICIENT, EQUIPMENT ADDITIONAL TO 3.2 ABOVE AND REACTOR POWER RESTRICTION IN EARLY LIFE SHALL BE ADJUSTED SUCH THAT THE ABOVE REQUIREMENTS ARE MET AT ALL TIMES WITH AN ADEQUATE ALLOWANCE FOR CALCULATION CONFIDENCE (ABOUT 10-3).

B. ADDITIONAL DETAILED NII REQUIREMENTS

THESE REQUIREMENTS ARE AIMED TO OBTAIN AN ACCEPTABLE BALANCE BETWEEN MODERATOR TEMPERATURE COEFFICIENT/ SAFETY VALVE CAPACITY/TIME WHICH ALLOWS US TO ACCEPT THE ATWS ARGUMENT.

MODERATOR TEMPERATURE COEFFICIENT

1. WE DO NOT AGREE WITH YOUR TARGET OF 10-6/REACTOR YEAR FOR AN ATWS TO HAVE SEVERE CONSEQUENCES AND BELIEVE THAT OUR TARGET OF 10-7/REACTOR YEAR CAN BE ACHIEVED.

2. WE DO NOT ENTIRELY ACCEPT THE PROBABILITY ANALYSIS OF APPENDIX VII, WHICH ASSESSES THE EFFECT OF PARAMETER VARIATIONS AND EQUIPMENT RELIABILITY, SINCE IT ASSUMES THAT EACH PARAMETER IS INDEPENDENT OF ONE ANOTHER. WE NOTE THAT YOU ARE PLANNING TO INVESTIGATE THIS ASSUMPTION. NO WORK IS SCHUD.

3. THE AREA OF DIFFICULTY IN THE REPORT WHICH WE ARE MOST CONCERNED IS WITH THE TREATMENT OF THE MODERATOR TEMPERATURE COEFFICIENT. IT DOES NOT APPEAR TO US NECESSARY TO STIPULATE THAT THE MODERATOR TEMPERATURE COEFFICIENT SHOULD BE -7 PCM/O F FOR 99$ OF THE TIME.

4. FAILURE TO TRIP THE TURBINE APPEARS TO HAVE BEEN NEGLECTED IN YOUR ANALYSIS. DO YOU REGARD THE RELIABILITY OF THIS TRIP SUFFICIENTLY GOOD SO THAT IT CAN BE NEGLECTED? YES IF AMSAC

5. WE REQUIRE THAT OPERATOR ACTION CAN TAKE PLACE AFTER 30 MINUTES COMPARED WITH 10 MINUTES IN YOUR SAFETY ASSESSMENT. OUR REQUIREMENT IS MORE SEVERE IN THAT THE MODERATOR TEMPERATURE COEFFICIENT MAY NEED TO BE LARGER TO MAINTAIN THE PLANT IN A SUBCRITICAL STATE UNTIL OPERATOR ACTION TAKES PLACE.

6. WE AGREE WITH THE REST OF YOUR REPORT AND CONCLUSIONS AS TO THE REQUIREMENTS FOR TESTS ON SAFETY VALVE WATER DISCHARGE RATES AND ON THE METHODS OF ANALYSIS USED.

YOURS SINCERELY

DR D L REED

Vince,

I have concurred. Brian indicated that he wished to take a look too.

Cheney 2753
f-424

2-2-79

BHL 2/2/79

B. B.,

We have revised this letter per our discussions of the other day. If in agreement please concur and have Vince do so also and mail back,

Thanks,

R. Cheney

Mary,

Send to Brian.

Vn

SHH
1-21-80

Early generic verification of Alternate 3 was not obtained. It might turn out all right but it hasn't yet.
Must quit this Q-and-A forever.
But need more information to determine adequacy.

1. Future plants must install Alternate 4.

2. Present plants must install Alternate 3A.
   Alt 3A: A list of hardware specified by NRC based on what we know today PLUS what we guess, subject to future verification

3. Define "Present Plants" and "Future Plants"

4. Define required information to be supplied

5. Implementation
   Design
   Hardware
   Information

6. If in the future the information does not verify the adequacy of the design as installed, will impose additional requirem

- 2 - To Advisory 1-30

9-3 Has any staff thought been seriously given to qualifying the HPCI pump to a radiation and steam environment such that in the event of pressure suppression pool loss core melting could still be averted?

a/ If so, has staff :

1- Asked G. E. to study and make a report on such modification

2- Taken a position that the HPCI pump must be qualified to tollerate the steam and radiation that would enter the auxillairy building in the event of loss of Suppression Pool integrity

Thadani
H Vander Molen

9-4 Has any thought by Staff been given to enlarging or in other ways altering the Condensate Storage Tank (CST) to avoid core melt occurs due to Suppression pool failure?

Thadani
H. Vander Molen

9-5 Are temperatures above 200°F. being considered as possibly acceptable in the event of ATWS with a G. E. BWR-III plant?

Thadani
Su

9-6 What other measures are planned to prevent utilities including Applicant from start ups when there is high xenon condition and low or no moderator void? These conditions are thought to cause unnecessary short period SCRAMS

Thadani
NA

9-7 Using only PWR rods, NUREG/CR-0582, "Evaluating Strength and Ductility of Irradiated Zircaloy, Task 5", states on Page 20,

> One test of the additional Lot 2 material again exhibited a larger hoop strain (60%) at a point other than the burst point; a definite reason for this observed effect has not been determined.

Does staff oppose this position, to wit, the burst of a fuel rod when subjected to a transient heating burst shows no relationship to locations on the road where hoop strain is demonstrable?

Dean
Houston

9-8 Does the NRC take the position that the most severe loading on the containment steel shell is the LOCA?

a/ Has the Commission studied the effects of ATWS on the steel containment shell, such as one proposed by Applicant?

Thadani

9-9 . Does Applicant's plan for its containment shell have unique or new features not covered in the Standard Review Plan'S

K. Wichman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JAN 17 1979

NOTE TO: R. J. Mattson

The newly formed task force on ATWS met for the first time on 1/16 to discuss the work required between now and May 1979 and to develop the questions/statements for generic ATWS analyses. The task force members were requested to provide their input to the generic set of questions/statements before 1/24/79.

During this useful period of briefing of the task members followed by exchange of viewpoints, the following important questions/comments were raised and discussed.

1. Since MTC value specified is based on estimates of future operation, what, if anything, can we do if the future operation is different than that assumed in the development of the specified value?

   My comment: The applicant should be required to recognize that if the plant design or operation changes appreciably in the future such that the plant does not fall within the generic envelope, he may be required to reconsider earlier ATWS conclusions. If this is reasonable, would the rule or the regulatory guide provide the necessary mechanism for accomplishing this objective.

2. If the plant were to be permitted to operate at its "stretch" rating, how would we treat such a large (in some cases) change in an important parameter?

   My comment: Same as under 1. above.

3. Some questioned the use of nominal values of parameters in generic analyses and recommended using bounding values. Also, what if the sensitivity to a parameter is very high?

   My comment: An objective is to determine, as well as we can, the realistic course of an ATWS and thus we should use nominal values. If there are small differences in the nominal parameter values for a class of plants, the sensitivity studies could and should be relied on to make judgments.

   Additionally, my judgment based on review of earlier ATWS analyses is that there is no threshold phenomenon (i.e., extreme consequence dependence on small variation in a parameter value); however, if there is a very important parameter whose initial value is not well understood, use of a conservative value could be required if the preverification approach is to be successful.

dupe of 8001110591

4. How should PCI failures be treated?

   My comment: Use NUREG-0460, Vol. 2 approach or specify what the penalty might be. Eliminate, if possible, vague guidelines.

5. Jim Norberg indicated he would need assistance from someone familiar with ATWS and developing rules and reg. guides. Perhaps Roger Mattson could ask for John Huang, an ex-member of Standards and now, I believe, in the Division of Operating Reactors.

6. Frank Cherny emphasized the need for DOR participation in the review of mechanical engineering aspects of operating reactors. He recommended that we request Keith Wichman of DOR to work with us on ATWS.

7. A difficult question, because of variety of subjects, was the required format for the task force members to prepare their questions and/or comments.

   My comment: I think the most straightforward approach is to state what we want.

   Examples: Identify approved models.
   Identify open areas and recommend a way to resolve open areas. Specify what kind of penalty may be imposed if the vendor does not provide acceptable response. (Note: No Q1s or Q2s.)
   Specify transients to be analyzed.
   Specify ICs and sensitivity studies.
   Specify assumptions for alt. #3 and alt. #4 analyses.
   Require list of plants which fall under each set of analyses.
   Require list of systems relied on.
   Specify requirements for these systems for different alternatives.
   Specify what the analysis must include as a minimum.
   Specify the constraints on future design or operational variations.
   Specify criteria under which dose calculations need not be performed.
   Specify limits and operability criteria and require vendors to show how each class of plants would meet these limits. Keep in mind different approaches in PWRs on alt. #3 and alt. #4.
   Require vendors to specify the necessary plant modifications to satisfy the criteria of Volume 3, NUREG-0460. Require vendors to provide sufficient detail to ascertain that the mitigating systems criteria of Vol. 3 of NUREG-0460 shall be satisfied.

8. Allotted time for this ambitious approach is too short.

I would appreciate (a) your comments, especially if you have any disagreements with the above approach and (b) your requesting DOR to add John Huang and Keith Wichman to the ATWS task force.

A C Thadani

Ashok C. Thadani
Reactor Systems Branch

cc: R. Tedesco
S. Hanauer
T. Novak
F. Cherny
T.M. Su
H. Richings
D. Thatcher
F. Odar
S. Salah
G. Kelly
M. Tokar
R. Woods
R. Lobel
V. Rooney
G. Chipman
E. Jakel
J. Norberg
S. Newberry

ENCLOSURE 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TAP A-9

JUL 12 1979

NOTE TO: Fuat Odar

FROM: Ashok Thadani

My recent work on W and CE plants has caused me to be concerned that we may have over emphasized our concerns with overpressure events in PWRs and not given adequate attention to the possibility of core uncovery from ATWS events. The following is a short summary statement of my concerns and recommendations for further audit calculations (I intend to discuss this possible problem with PWR vendors also).

CONCERN IN PWRS

A. ATWS Events With Proper Functioning of Pressurizer Valves

For events like Rod Withdrawal (RW) with Turbine Trip and LOFW* (or LOL**) what is the power profile for long term. Note pressure and L_{przr} may be such that the operator may not actuate HPSI (Borated Water) even after 10 minutes and further the system pressure may be way above HPSI shut off head for some plants (e.g. some HPSI shut off head is 1200 psi). If the power remains high enough and if the pressure remains high enough such that Borated Water is either not injected or injection is delayed because of high system pressure then a potential for core uncovery exists.

Further, if water relief thru safety/relief valves is the major way of removing energy, then the core uncovery could occur because $W_{leak}\ h_f$ has to match energy produced and since $h_f < h_g$, W_{leak} has to be pretty high.

h_f - Coolant water enthalpy at Pressure P
h_g - Coolant Steam enthalpy at Pressure P
W_{leak} - Flow out of Pressurizer safety/relief valves
L_{przr} - Pressure level

---

*LOFW - Loss of Feedwater Flow
**LOL - Loss of Load

Dupe
8001110529

Need for Calculations on PWR

LOFW (or LOL) and RW Event

Analyze event assuming:

Case 1

a) Secondary heat transfer loss at X = 0.9 in Steam Generator
b) All Aux Feed Available
c) 99% and 95% MTC

Case 2

a) heat transfer loss at X = 0.9
b) 1/2 Aux Feed Available
c) Same as 1c

Case 3

a) heat transfer loss at X = 0.95 (or even higher)
b) full Aux Feed Available
c) Same as 1c

Case 4

a) Same as 3a
b) 1/2 Aux Feed Available
c) Same as 1a

Factors: It is important to correctly model heat flux and primary system inventory (HPSI - important).

Carry the calculations far enough (say 20 minutes or longer) to determine if the core can be uncovered.

Note: In these calculations 0.9 X HEM model may be non-conservative.

B. ATWS Events With One or More Valves Stuck Open

Concerns: Are the vendor codes for this scenario adequate? Could the codes used be incapable of estimating void generation in the primary? Do we have audit capability for this scenario?

We need an early discussion of the above concerns. If the concerns are real, we need to perform some calculations soon.

A. Thadani

Ashok Thadani

cc: S. Hanauer
M. Aycock
ATWS Distribution

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 8 1979

NOTE TO: M. B. Aycock, Deputy Director
Unresolved Safety Issues Program

FROM: A. Thadani, Unresolved Safety Issues Program

Enclosures 1 and 2 describe some of my concerns on the incompleteness of our audit calculations on BWRs and PWRs respectively. I see a need for a short term (1 ~ 2 months) as well as long term (3 ~ 4 months) effort to conduct some audit calculations to confirm our past judgments on ATWS. The manpower needs and the computer time estimates are preliminary and were provided by M. Levine of BNL.

BWRs

All ATWS calculations to date performed by GE have utilized "REDY" code. Some audit calculations were performed by BNL in 1973, 1974. Subsequent tests at the Peach Bottom reactor indicated some inadequacies of the REDY code. Currently GE uses a 1.D "ODYN" code for all overpressure transient events. The staff is adamant that Turbine Trip Without Bypass (TTWOBP) ATWS overpressure event be analyzed using "ODYN" code. As discussed in Enclosure 1, two types of audit calculations should be performed.

Type 1: Short Term Plant Response

Analyze two ATWS transients, TTWOBP and MSIV closure
Carry calculations up to 1 minute real time
Codes: TWIGL - RELAP-3B
Manpower: 2 men - 4 to 6 weeks
Computer Time: 5 hours

Type 2: Long Term Plant Response (~ 10 min.)

Analyze effects of Boron injection on plant response for TTWOBP and MSIV closure ATWS events.
Codes: RELAP-3B
Manpower: 1 man - 2 months
Computer Time: 4 hours

dupe of
8001110658

PWRs

As explained in Enclosure 2, the staff audit calculations addressed only over-pressure concern and not the potential for core uncovering for some ATWS events. Thus there is a need for the following audit calculations (20 ∿ 30 minutes real time) for each vendor design.

Transients: Loss of feedwater with stuck open valve
Loss of offsite power with stuck open valve

a. base case
95% MTC, HEM
b. 99% MTC
c. Time delay in aux feed and 1/2 aux feed
d. 0.9 HEM
e. HPSI Design effects

Codes: IRT - RELAP-3B
Manpower: 2 man - 4 months
Computer Time: 80 hours

Comment: If we expect to go to Commission by December 1979, then we need as a minimum short term BWR audit calculation as well as the PWR calculations. If the PWR calculations show serious core uncovering problem, then I think we should discuss that with the Commission before December 1979 because our perception of higher risk from BWRs may not be quite correct. Because BNL staff is committed on other tasks (some physics type manpower may be available), I spoke with Richard Denning and Bob Collier of BCL and they indicated their knowledge of RELAP and their willingness to provide personnel to go to BNL and use BNL facilities to perform these tasks. BNL is receptive to the idea of getting this help from BCL. Thus with coordinated effort between NRC/BNL/BCL we may be able to meet the following schedule if we begin work by 9/1/79.

BWRs

Type 1: Complete by 10/30/79
Type 2: Complete by 11/30/79

PWRs

Preliminary Assessment 11/30/79
Studies Complete 2/28/80

Total Manpower ∿ 13 Man Months
Total Computer Time ∿ 90 hours

The need to perform BWR ATWS analysis:

Short Term:

Previous audit calculations were performed using point kinetics. The model did not include steam line dynamics. The previous GE code used for ATWS analysis is the REDY code. The REDY code did not predict Peach Bottom test results where steam line dynamics and space kinetics were important. The REDY code predicted neutron flux peak nonconservatively by a factor of 2 to 3. The need for fairly accurate heat flux cannot be overemphasized because of the resultant effects on containment and other structures. This is particularly important for plants with alternative #3 fix. On a best estimate basis the REDY code is not acceptable for sudden overpressurization transients. General Electric submitted the ODYN code for the analysis of these transient and they do not seek the approval of REDY for sudden overpressurization transients. Most ATWS events result in rapid overpressure condition. Hence, previous analyses performed by GE should be reperformed using the ODYN code at least to verify previous analyses. The staff should reperform the audit calculations using steam line dynamics and space kinetics models.

Long Term:

Audit calculations were not performed to verify GE calculations for long term behavior. The effectiveness of the fixes "Boron reactivity·feedback" was never evaluated. Both short and long term energy releases are important to evaluate torus behavior. The frequency and duration of the opening of the valves are governed by the effectiveness of the fix and the dynamics of the steamline. Because of the criticality of alternative #3 fix (small margin to limit) and its impact on consequences, it is necessary to perform some audit calculations.

The need to perform PWR ATWS analyses:

TMI-2 event showed that some transients may lead to boiling in the primary system loop and eventually to core uncovering. The previous ATWS analyses were performed evaluating the overpressurization effects which occur for a short time in the beginning of the transient. The aspects of core uncovering and boiling in the primary system were overlooked. It is necessary to establish: 1) the validity of the vendor codes used in the ATWS analysis if there is some boiling and 2) if there is boiling, does the core uncover. We need audit calculations to answer these questions.

TMI-2 event also showed that ATWS audit calculations must cover some failures which impact consequences. These failures are 1) stuck open valve, 2) delay in auxiliary feedwater and 3) reduction in auxiliary feedwater. We need audit calculations to establish sensitivities of the ATWS events to these failures. Further an inadvertent opening of a safety or relief valve is an anticipated transient which may have significant consequences and audit calculations are necessary to confirm vendor analysis. (Note vendor analyses are probably inadequate). The potentially serious consequences for some design (different HPSI shut off head) should be carefully reviewed and audit calculations of such cases are warranted.

A. Thadani
Unresolved Safety Issues Program

cc: ATWS Distribution

Enclosure:
As stated

ATWS Distribution

A. Thadani
T. Su
L. Ruth
K. Parczewski
M. Srinivasan
H. Vander Molen
M. Tokar
W. Regan
S. Coplan
F. Akstulewicz
F. Cherny
M. Aycock
T. Novak
R. Tedesco
R. Denise
R. Mattson
K. Kniel
T. Speis
P. Check
D. Eisenhut
B. Grimes
V. Noonan
R. Bosnak
D. Muller
F. Schroeder
J. Norberg
E. Jakel
ACRS (21)
PDR
F. Odar
RSB File (Carol Jamerson)
S. Hanauer
D. Fieno
K. Kniel

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D.C. 20555

JUL 11 1979

Generic Task Action A-9

NOTE TO: S. H. Hanauer, Director
Unresolved Safety Issues Program

FROM: Ashok Thadani

SUBJECT: BWR CODE EVALUATION

Most of the ATWS analyses were performed by GE using their code called "REDY" as described in NEDO-10802. Our audit calculations using RELAP only cover several seconds (25 seconds) of an ATWS transient and the calculated peak pressures agreed well with the GE calculational result. However, GE is now using "ODYN" code for all overpressure events. This new code was developed by GE because the Peach Bottom turbine trip test results did not agree well with the results predicted by "REDY". The application of ODYN code to ATWS events has not yet been reviewed by the staff. I see a need for the following effort in code evaluation.

1. Review "ODYN" for ATWS application.
2. Perform Audit Calculations
   a) First several seconds of ATWS event - determine flux, pressure, S/R discharge (the model includes RPT).
   b) Several minutes of ATWS Event - Use different SLCS injection rates, injection time and Sodium Pentaborate Solution concentration. Look at power, pressure, discharge through S/R valves and estimate pool temperatures.

Currently BNL is planning (under Tech Assistance contract) to use RAMONA (a 3D code) for transient analyses. It would appear, on the basis of my discussions with Fuat Odar, that any ATWS audit calculations using RAMONA cannot be completed until some time next year (~ March).

Since we hope to propose to the Commission a recommended course of action on BWRs in the next few (3-4?) months, I see a need for the following:

dupe of

8001110521

- Complete Item 1 above in two months.] Acceptable Model By 8/31/79
- Complete Item 2a above in two months (this could be justification of why earlier RELAP studies provide sufficient basis for now but to be further confirmed later using RAMONA or other suitable code). also, 8/31/79
- Complete item 2b early next year but prior to ATWS rule being effective (guess - March '80). Completion of this task would require that a fairly simple containment model be incorporated in the code.] this would answer stability question

While I see a need for supporting ATWS audit calculations, I do not believe that a 3-D core model is needed to get more accurate reactivity feedback effects. Before we sign a Tech Assistance contract using RAMONA, I recommend that you, Mike, Fuat, Dan (Fieno), and I meet to discuss our needs and help Fuat prepare Tech Assistance request consistent with our ATWS plans for BWRs.

Ashok Thadani

cc: M. Aycock
F. Cherny
D. Fieno
RSB Files
ATWS Dist.
F. Odar
T. Speis

PACIFIC GAS AND ELECTRIC COMPANY

PG&E — 77 BEALE STREET, 31ST FLOOR • SAN FRANCISCO, CALIFORNIA 94106 • (415) 781-4211

July 10, 1980

Mr. A. Schwencer, Acting Chief
Licensing Branch No. 3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Re: Docket No. 50-275
Docket No. 50-323
Diablo Canyon Units 1 & 2

Dear Mr. Schwencer:

Enclosed for your review is a draft of Operating Procedure OP-38, entitled, "Anticipated Transients Without Trip." This procedure is being provided at the request of the Staff (Mr. A. C. Thadani) to assist in the review of the ATWS issue at Diablo Canyon. It is not to be considered part of the docket file supporting our operating license application, nor will future revisions be submitted to the Staff unless requested.

Approved copies of the procedure will be available to the Region V Inspectors.

Very truly yours,

Philip A. Crane, Jr.

Enclosure

B001
s
1/1

Dup of

8007160520

PG&E **Pacific Gas and Electric Company**

NUMBER OP-38
REVISION 0
DATE 6/7/80
PAGE 1 OF 5

DEPARTMENT OF NUCLEAR PLANT OPERATIONS

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

EMERGENCY OPERATING PROCEDURE

TITLE ANTICIPATED TRANSIENT WITHOUT TRIP (ATWT)

APPROVED ______________________
PLANT MANAGER

FOR INFORMATION ONLY

SCOPE

This procedure describes the steps to be taken in the event of an ATWT. An ATWT is a failure of the reactor protection system to trip the rods in when one or more reactor trip setpoints have been reached.

SYMPTOMS

1. Reactor trip point exceeded without a reactor trip.
2. Possible Reactor Protection System activated alarm.
3. Possibly the reactor trip alarm.
4. DPRI indicates no rods drop.
5. RCS Hi pressure and level alarm.
6. NIS continues to read upscale.

AUTOMATIC ACTIONS

1. PZR PORVs open.
2. PZR spray valves open.
3. PZR safety valves open.
4. Steam dump activated.

OBJECTIVES

1. Ensure the reactor is shutdown.
2. Provide a heat sink for the reactor.

IMMEDIATE OPERATOR ACTIONS

ACTIONS	COMMENTS
1. Manually trip the reactor.	1. Use the red handle.
a. Verify rod bottom lights on DPRI.	
b. Verify NIS decreasing.	

8007160521

FOR INFORMATION ONLY

ACTIONS

2. If the rods fail to drop after Step 1 above, OPEN the 480 Volt LC 13 D and E breakers 52 HD 13 and 52 HE-4.

3. If the rods fail to drop after Step 2 above, open BIT inlet and outlet valves and start both centrifugal charging pumps.

4. Verify all three auxiliary feedwater pumps running.

5. Trip the turbine manually if required.

6. If the turbine fails to trip after Step 4 above, trip the turbine using the trip lever on the turbine pedestal.

7. Sound the site emergency alarm.

SUBSEQUENT OPERATOR ACTIONS

1. Verify steam dump operating to the condenser or 10% atmosphere steam dumps open. Transfer steam dump to the steam pressure mode with a 1005 psig setpoint.

2. Verify that at least one RCP is operating. If not, start as many as possible.

3. Check all rod bottom lights on, emergency borate 100 ppm for any stuck out rod.

4. Check the gross failed fuel detector for any signs of fuel damage.

5. Monitor steam generator water levels, air ejector off gas and steam generator blowdown radiation monitors for any indication of a steam generator tube rupture.

COMMENTS

2. This will deenergize the load centers supplying power to the rod control MG sets.

3. If the BIT is injected and the rods remain out of core, it is important to keep the RCP's in service and maintain hot standby conditions. A cooldown could allow the reactor to return to criticality.

5. With the reactor protection system failed, the P-4 signal is not present to trip the turbine.

Subsequent operator actions:

1. Monitor the heat sink (steam dump) closely after this transient.

2. RCP seals should be observed closely as the RCS Hi pressure may have affected them.

3. If no rods have inserted, emergency borate the RCS until 2000 ppm is achieved.

5. The leak may occur as a result of the RCS Hi pressure during the transient.

ACTIONS	COMMENTS
6. Monitor RCS parameters.	
a. Tavg should return to 547.	
b. PZR level should remain above 22%.	
c. RCS pressure should remain above 1950 psig.	
7. If pressurizer pressure decays below 1900 psig:	
a. Verify closed all PORV (close the backup valve if a PORV is found open.)	a. The Hi pressure transient may have stuck open a PORV.
b. Verify closed the PZR spray valves. (Close any valve found open.)	
8. Monitor all SI initiation parameters (PZR pressure, containment pressure, etc.) for SI conditions. If any parameter exceeds the SI initiation setpoint, manually initiate safety injection and proceed to OP-0.	8. With a failure in the Reactor Protection System, the automatic SI initiation is in doubt.
9. If manual initiation of SI fails, proceed to OP-0 and perform all Immediate Operator Actions Steps using manual control.	
10. If SI is not required, proceed as follows:	
a. Verify feedwater control valves close when Tavg reaches 554°F.	
b. Transfer the NIS recorder to monitor one IR and one SR channel.	
c. Declare this event a site emergency. Notify the appropriate outside agencies given in Emergency Procedures General Appendix 2 (Notification of Off-site Personnel in the Event of an Emergency).	

ACTIONS | COMMENTS

d. Check the turbine-generator coasting down properly.

   1) All turbine drain valves open.

   2) The AC bearing oil pump and the high pressure seal oil backup pump start automatically.

   3) The lift pump starts at about 600 RPM.

   4) The turning gear engages automatically at or near zero speed.

e. Maintain condenser vacuum; if vacuum is lost, use the 10% atmospheric dump valves to control steam generator pressure.

f. Establish and maintain hot standby operation, verify shutdown margin per STP R-19, and adjust RCS boron concentration if necessary.

g. If condenser vacuum is lost, check the level in the condensate storage tank to determine how long the unit can be maintained in hot standby prior to going to cold shutdown. Refer to Emergency Operating Procedure OP-7.

h. Prepare to take the plant to cold shutdown conditions.

FOR INFORMATION ONLY

APPENDIX Z
EMERGENCY PROCEDURE NOTIFICATION
INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the shift foreman proceed as follows.

   a. Notify the Plant Superintendent and Supervisor of Operations or their designated alternates.

   b. Designate this event a Site Emergency. Notify those agencies given in General Appendix 2 of the Emergency Procedures (Notification of Outside Agencies in the Event of an Emergency).

   c. Within one hour notify the NRC Operations Center using the red phone in the control room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 20 1980

MEMORANDUM FOR: Karl Kniel, Chief
Generic Issues Branch, DST

FROM: Ashok C. Thadani
Generic Issues Branch, DST

SUBJECT: NRC-EPRI ATWS MEETING SUMMARY

The staff met with the Electric Power Research Institute (EPRI) on May 5, 1980 to discuss the EPRI as well as the NRC considerations of the significant transients, the frequencies of these transients, and the testing frequencies of the electrical portions of the scram systems.

I. EPRI Presentation on Frequency of Anticipated Transients

The EPRI analyses (Enclosure 2) concludes that:

- the total frequency of anticipated transients is 10.59 per reactor year for PWRs and 9.37 per reactor year for BWRs.
- the transients important for ATWS consideration have frequencies of 3.74/RY and 4.7/RY for PWRs and BWRs respectively.
- the ATWS events below 25% rated power level do not result in severe consequences and thus the frequencies of transients of significance is further reduced to 1.96/RY and 3.52/RY for PWRs and BWRs respectively.
- the extropolation of two transients using the learning curve (first year frequency + 39 x average frequency of years 2 through 8) /40 and individual plant design considerations would further reduce the significant transient frequencies to

  1.45/RY for B&W designed plants
  1.65/RY for CE designed plants
  1.18/RY for W designed plants
  3.52/RY for GE designed plants

Staff Comments:

The following Staff Comments were provided to EPRI concerning the frequency of significant transients in PWRs and BWRs.

dupe of
8006020127

DUPLICATE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 19 1980

Thanks
RH

NOTE TO: R. H. Vollmer, Director, Division of Engineering

THRU: R. J. Bosnak, Chief, Mechanical Engineering Branch, DE
J. P. Knight, Assistant Director for Components & Structures ~~Engineering, DE~~

FROM: F. C. Cherny, Section Leader, Mechanical Engineering Branch, DE

SUBJECT: SERVICE LEVEL C STRESS LIMIT - ATWS

Reference: Your recent question to J. Knight

There is, as you probably know, a long story that can be told of the entire period from the time when ATWS began as a generic unresolved issue, about 10 years ago, to where it is today, essentially almost resolved. The background as to how the ASME Service Level C has come to be part of the final acceptance criteria is in itself no less complex than most of that 10 year story. If you would like to discuss the background regarding the selection of the Service Level C criterion, I would be pleased to discuss it with you at any time.

However, in response to your immediate question to J. Knight regarding the meaning of the Service Level C Limit, I would briefly clarify as follows.

The ASME Boiler and Pressure Vessel Code specifies several stress limits of varying degrees of conservatism which can be used in the design of reactor coolant system components. These limits are termed in decreasing order of conservatism: Design, Service Level A, Service Level B, Service Level C, and Service Level D.

Without going into too much detail, the Design, Level A, and Level B limits are used to design components for normal steady state plant operation and for anticipated upset transient conditions. Compliance with the Design and Service Level A limits assures that reactor coolant pressure boundary component primary membrane stress levels are at or below the lower of 2/3 of the minimum yield strength or 1/3 of the tensile strength of the material for ferritic materials or 90% of the yield strength for stainless materials.

For the Level B limits, commonly accepted by NRC and throughout the industry for anticipated transient conditions the primary membrane stress level is permitted to rise as much as 10% over the Design or Level A limit, resulting only from loading associated with a pressure increase within the component.

dupe of
8007160597

Addressing your specific question, the Level C limit basically allows primary membrane stress levels up to the material yield strength for both stainless and ferritic materials.

The Level D limit is a great deal less conservative than Level C and has historically been used only for LOCA and SSE type loads where the stress levels can indeed be very high, but, unlike the ATWS pressure loads, tend to be quite localized in a given area of a component or support.

F. C. Cherny

F. C. Cherny, Section Leader
Mechanical Engineering Branch
Division of Engineering

cc: A. Thadani

ATWS Structural Integrity Comments
Re: Vol. 4 of NUREG-0460

A) SNUPPS Letters of 5-1-80 and 4-7-80

B) AIF Letter of 5-27-80

Summary of A) + B) comments:

Service Limit C criterion is too conservative.

Response — Basis for level C criterion is adequately documented in section 7.1.2 of Vol. 1, NUREG-0460 and Appendix D of Vol. 2, NUREG-0460

Industry has yet to make a substantive comment in this area.

c) BG&E Letter of 5-15-80

Summary of Comments: CENPD-263-P provided adequate demonstration of structural integrity

(B)

using "ground rules" provided in Vol. 3, NUREG-0460.

Response: No additional information provided. This submittal was discussed in depth in the C.E. Appendix in Vol. 4 of NUREG-0460.

III Joliet Edison	5/8	VanderMolen
IV EG&E	5/15	Thadani / Thatcher / Cherry / Akstulewicz
V WPPS	5/15	Thadani / VanderMolen / Thatcher / Pittiglio / Nash
VI EPCI	5/20	Thadani / Nash / Pittiglio
VII GE	5/23	VanderMolen (Help from Cherry / Sn).
VIII W	5/15	Thatcher
IX Midland	5/16	VanderMolen
X EG&E (to Ahearne)	5/15	VanderMolen
XI SNUPPS Letter		

Note SNUPPS Ltr. to AT ~~, EG&E Ltr to Chairman Ahearne~~ missing – Check with VanderMolen

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Copy for Cherry

May 19, 1978

Note to A. Buhl
P. Check
H. Denton
D. Eisenhut
B. Grimes
J. P. Knight
H. Ornstein
M. Malsch
R. Mattson
W. Minners
T. Novak
D. Ross
Z. Rosztoczy
V. Stello
M. Taylor
A. Thadani

SUBJECT: ATWS ANSWERS FOR ACRS

Enclosed you will find draft answers to ACRS questions as follows:

May 1 McCreless memo Item 6

May 26 " " Item 1

Since I am going on official travel on Tuesday, comments should reach me by cob Monday, May 22.

Stephen H. Hanauer
Technical Advisor to
Executive Director for Operations

Enclosures:
1. May 1 Item 6
2. May 26 Item 1