September 27, 1983

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LS05-83-09038

Docket No. 50-244

Mr. John E. Maier, Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT NUREG-0737 ITEMS II.K.3.1-AUTOMATIC PORV ISOLATION AND II.K 3.2-REPORT ON PORVs

R. E. Ginna Nuclear Power Plant

Item II.K.3.2 of NUREG-0737 required licensees of pressurized water reactors to submit a report to the NRC staff documenting the various actions taken to decrease the probability of a small break loss of coolant accident (LOCA) caused by a stuck-open power operated relief valve (PORV) and show how these actions constitute sufficient improvements in reactor safety. Safety valve failure rates based on past history of the operating plants designed by the specific nuclear steam supply system (NSSS) vendor were to be included in the report. Licensees had the option of submitting either a plant specific report or a generic report. Where a generic report was submitted, each licensee was required to document the applicability of the generic report to its plant.

Based upon the results of the report submitted in response to item II.K.3.2, licensees were to assess whether an automatic PORV isolation system was required. If required, licensees were to submit a system design that uses the PORV block valve to automatically protect against a small break LOCA caused by a stuck open PORV. Documentation was to include piping, instrumentation diagrams, electrical schematics and be in conformance with IEEE 279-1971 requirements.

The Westinghouse Owners Group submitted a generic report to the NRC staff in response to Item II.K.3.2 titled "Probabilistic Analysis and Operational Data in Response to NUREG-0737, Item II.K.3.2, for Westinghouse NSSS Plants." Westinghouse Electric Corporation, February 1981 (WCAP-9804).

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Your response to the subject NUREG-0737 items dated July 1, 1981 adopted the conclusions reached in WCAP-9804 as applicable for your facility namely that "the concept of an automatic PORV block valve closure system, which closes the PORV isolation valves when lower pressure is sensed subsequent to a PORV failing to close, cannot be warranted on the basis of providing additional protection against a PORV LOCA."

On this basis you proposed no modifications to provide automatic isolation of the PORVs in response to Item II.K.3.1

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We have completed our review of your response to the subject NUREG-0737 items including the Westinghouse Owners Group Report WCAP-9804. Our findings are contained in the enclosed Safety Evaluation (SE), which includes our contractor's Franklin Research Center's Technical Evaluation Report (TER) that evaluates the data contained in WCAP-9804. Based upon our review, we find that the requirements of NUREG-0737 Item II.K.3.2 are met with existing PORV, safety valve and reactor high-pressure trip setpoints and that an automatic PORV isolation system is not required for Ginna. This completes the staff's review of the subject NUREG-0737 items for your facility.

Sincerely,

Original signed by

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
Safety Evaluation with attached
Technical Evaluation Report,
dated 2/15/83, as revised
4/21/83.

cc w/enclosure
See next page

OFFICE	DL:ORB #5	DL:ORB #5	DL:ORB #5				
SURNAME	GDick:jc	HSmith	DCrutchfield				
DATE	9/23/83	9/12/83	9/13/83				

NRC FORM 318 (10-80) NRCM 0240 **OFFICIAL RECORD COPY** USGPO: 1981—335-960

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Docket No. 50-244

Mr. John E. Maier, Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

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Based upon the results of the report submitted in response to item II.K.3.2, licensees were to assess whether an automatic PORV isolation system was required. If required, licensees were to submit a system design that uses the PORV block valve to automatically protect against a small break LOCA caused by a stuck open PORV. Documentation was to include, piping, instrumentation diagrams, electrical schematics and be in conformance with IEEE 279-1971 requirements.

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On this basis you proposed no modifications to provide automatic isolation of the PORVs in response to Item II.K.3.1.

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September 27, 1983

Docket No. 50-244
LS05-83-09-038

Mr. John E. Maier, Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT NUREG-0737 ITEMS II.K.3.1-AUTOMATIC PORV ISOLATION AND II.K 3.2-REPORT ON PORVs

R. E. Ginna Nuclear Power Plant

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Based upon the results of the report submitted in response to item II.K.3.2, licensees were to assess whether an automatic PORV isolation system was required. If required, licensees were to submit a system design that uses the PORV block valve to automatically protect against a small break LOCA caused by a stuck open PORV. Documentation was to include piping, instrumentation diagrams, electrical schematics and be in conformance with IEEE 279-1971 requirements.

The Westinghouse Owners Group submitted a generic report to the NRC staff in response to Item II.K.3.2 titled "Probabilistic Analysis and Operational Data in Response to NUREG-0737, Item II.K.3.2, for Westinghouse NSSS Plants." Westinghouse Electric Corporation, February 1981 (WCAP-9804).

Your response to the subject NUREG-0737 items dated July 1, 1981 adopted the conclusions reached in WCAP-9804 as applicable for your facility namely that "the concept of an automatic PORV block valve closure system, which closes the PORV isolation valves when lower pressure is sensed subsequent to a PORV failing to close, cannot be warranted on the basis of providing additional protection against a PORV LOCA."

On this basis you proposed no modifications to provide automatic isolation of the PORVs in response to Item II.K.3.1.

We have completed our review of your response to the subject NUREG-0737 items including the Westinghouse Owners Group Report WCAP-9804. Our findings are contained in the enclosed Safety Evaluation (SE), which includes our contractor's Franklin Research Center's Technical Evaluation Report (TER) that evaluates the data contained in WCAP-9804. Based upon our review, we find that the requirements of NUREG-0737 Item II.K.3.2 are met with existing PORV, safety valve and reactor high-pressure trip setpoints and that an automatic PORV isolation system is not required for Ginna. This completes the staff's review of the subject NUREG-0737 items for your facility.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
Safety Evaluation with attached
Technical Evaluation Report,
dated 2/15/83, as revised
4/21/83.

cc w/enclosure
See next page

Mr. John E. Maier

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

INTRODUCTION

In response to NUREG-0737 Item II.K.3.2, licensees were required to perform the following actions:

(1) The licensee should submit a report for staff review documenting the various actions taken to decrease the probability of a small-break loss-of-coolant accident (SBLOCA) caused by a stuck-open power-operated relief valve (PORV) and show how those actions constitute sufficient improvements in reactor safety.

(2) Safety valve (SV) failure rates based on past history of the operating plants designed by the specific nuclear steam supply system (NSSS) vendor should be included in the report submitted in response to (1) above.

The purpose of this Safety Evaluation is to evaluate the responses of Westinghouse licensees to the above requirements.

The requirements of NUREG-0737 allowed each licensee the option of preparing and submitting either a plant-specific or a generic report. If a generic report were submitted, each licensee was to have documented the applicability of the generic report to his plant. All Westinghouse licensees referenced a Westinghouse report (WCAP-9804) prepared by the Westinghouse Owners Group to address the staff's concerns. Licensees asserted that WCAP-9804 was applicable to their plants but did not, however, provide any supporting documentation. The Westinghouse report claims that the requirements of NUREG-0737 Item II.K.3.2 are met with the existing PORV, SV and high-pressure reactor trip setpoints, and that no automatic PORV isolation system is required for Westinghouse plants. Therefore, the staff's review, which was mainly based on the technical evlauation performed by our contractor, Franklin Research Center (FRC), was concentrated on two areas, namely the adequacy of the Westinghouse report, and its applicability to any Westinghouse plant. The results of the FRC review are contained in the attached Technical Evaluation Report (TER).

DISCUSSION AND EVALUATION

A. CONTENTS OF WCAP-9804

The Westinghouse report considered a spectrum of initiating events that may lead to PORV/SV opening. The event tree methodology was utilized to determine various possible outcomes due to the initiating events and to estimate the SBLOCA frequencies due to a stuck-open PORV/SV (SBLOCA-PORV/SV frequencies).

The initiating event frequencies were based on the generic estimates for PWRS given in NP-801[3], and the estimates of SBLOCA-PORV/SV frequencies were obtained from the frequencies of the initiating events, the probabilities of exceeding PORV/SV setpoints given the initiating events, the availabilities of the PORV block valves, the PORV/SV failure probabilities, and the probability of operator error.

In addition, the Westinghouse report considered the impact of post-TMI modifications on probability data and compared pre-TMI results with post-TMI results.

Finally, Westinghouse performed sensitivity analyses on post-TMI results to assess the impact of the following parameters:

(1) safety injection system difference (high-head vs. low-head)
(2) probability of PORVs being blocked off
(3) probability of operator error

B. ADEQUACY OF WCAP-9804

Based on our review, the staff finds that the event tree methodology used in the Westinghouse report is a valid approach to estimating the SBLOCA-PORV/SV frequencies. The staff finds that most of the probabilistic data in the Westinghouse event tree appear reasonable with a few exceptions, for example, the PORV/SV failure probabilities. The Westinghouse analysis also includes a few stuck-open PORV/SV scenarios due to a spurious actuation of a safety injection system.

The results of the Westinghouse analysis (with credit for operator action) indicate that the post-TMI SBLOCA-PORV frequency is about $2X10^{-6}$/reactor-year for a plant with a high-head safety injection system, and is about 10^{-6}/reactor-year for a low-head plant. In addition, the post-TMI SBLOCA-SV frequency is about $5X10^{-6}$/reactor-year for a high-head plant. As discussed in the TER, FRC has performed calculations and verified these estimates, given the validity of the Westinghouse data. However, the staff believes that the following considerations should be incorporated in the Westinghouse analysis:

(1) PORV/SV Failure Probability

Westinghouse uses 10^{-3}/demand as the PORV/SV failure probability to reseat. This is the failure probability per opening, not per transient. The staff believes that the PORV failure probability may be an order of magnitude higher if the PORV failure event at Ginna and the PORV failure event at North Anna 2 (Licensee Event Report 80-29) are also included in estimating the PORV failure probability. Therefore, the SBLOCA-PORV frequency may increase by an order of magnitude.

(2) PORV Block Valve Availability

The Westinghouse analysis assumes that 45% of the time PORVs are not blocked off. If a plant operates with PORVs not blocked off all the time, the SBLOCA-PORV frequency may increase by about a factor of two. By the same token, if a plant operates with PORVs blocked off all the time, the SBLOCA-SV frequency may also increase by about a factor of two.

(3). Multiple PORV Openings

The Westinghouse calculation of SBLOCA-PORV frequency assumes a PORV opens once per transient. Most Westinghouse plants have two PORVs, and a few even have three PORVs. Therefore, depending on the load rejection capabilities, it is not uncommon for a PORV to open several times or for multiple PORV openings during an overpressure transient. Therefore we believe that the effect of multiple PORV openings should be included in estimating the SBLOCA-PORV frequency.

C. APPLICABILITY OF WCAP-9804

Our contractor FRC identified one Westinghouse plant, Plant No. 7 in Table I.1, "PORV Openings," of WCAP-9804, as having had an excessive PORV frequency compared with the other 27 Westinghouse plants. Further, the staff's review of the PORV opening data given in the Westinghouse report indicates that there is a large variance in the PORV challenge frequencies among the Westinghouse plants, and there is no PORV challenge data for numerous Westinghouse plants in the Westinghouse report. The staff believes that post-TMI modifications would result in a significant reduction in the PORV challenge frequency for Plant No. 7 of Table 1.1. However, in order to ascertain whether the generic Westinghouse report applies to a specific Westinghouse plant, the staff reviewed plant-specific information such as the PORV/SV challenge frequencies, the fraction of time the PORV block valves are closed, and the various post-TMI modifications that may have reduced the PORV/SV challenge frequencies. Based on our review, the staff believes the Westinghouse Owners Group report to be applicable to all licensees identified in Table 1. The report is judged not applicable to McGuire Unit 1.

D. REVIEW OF RECENT PORV/SV DATA

(1) Estimate of SBLOCA-PORV Frequency

NUREG-0737 Item II.K.3.3, "Reporting SV and RV Failures and Challenges," requires that all PWR licensees promptly notify NRC of the PORV/SV failures and periodically report the PORV/SC challenges in annual or monthly reports beginning April 1, 1980. This requirement to report the PORV/SV operational data was imposed because, prior to the TMI accident, there was insufficient data to portray accurately the operational PORV/SC failures and challenges.

PORV/SV failure and challenge data from April 1, 1980 to March 31, 1983 was obtained from the above reports. There were no PORV/SV challenges during the 3-year period for many of the Westinghouse plants listed in Table 1; the maximum number of PORV challenges during the 3-year period was 4. If we use the 4 PORV challenges in 3 years, we estimate that the upper 95% confidence limit on the PORV challenge frequency is about 3.1/reactor-year. Moreover, assuming that (i) the PORVs are not isolated, (ii) the PORV failure probability is 10^{-2}/demand, and (iii) the operator error probability in not isolating a stuck-open PORV is $5X10^{-2}$/demand, we estimate that the SBLOCA-PORV frequency is about $1.5X10^{-3}$/reactor-year which still remains within the range of the SBLOCA frequency given in WASH-1400[4] (10^{-2} to 10^{-4} per reactor-year). The staff believes that the staff estimate of SBLOCA-PORV frequency is conservative because the operational data of 4 challenges in 3 years is bounding for all plants, and because the 95% confidence limit is used for estimating the PORV challenge frequency. Moreover, depending on the fraction of time that

PORVs are actually blocked off due to leakage, the PORV challenge frequency may be somewhat less. For example, plant specific data indicates that for about 33% of the time during the last 3 years PORVs have been blocked off. The staff estimates that the SBLOCA-PORV frequency would be about 1.0×10^{-3}/reactor-year, considering the fraction of time that the PORV block valves were closed for all plants.

(2) Estimate of SBLOCA-SV Frequency

The staff modifies the Westinghouse estimate of the SBLOCA-SV frequency with the assumptions below:

(i) There are in general 3 SVs in a Westinghouse plant.

(ii) PORVs are blocked off all the time due to leakage.

(iii) The SV failure probability is 10^{-2}/demand according to WASH-1400 and the recent IREP study on ANO-1.[5]

The staff estimates that the SBLOCA-SV frequency is about 3×10^{-4}/reactor-year which falls toward the lower end of the range of the SBLOCA frequency given in WASH-1400 (10^{-2} to 10^{-4} per reactor-year).

This estimate was obtained as follows. Westinghouse estimates the SBLOCA-SV frequency as 5×10^{-6}/reactor-year. However, their challenge frequency did not take account of the fact that all three SVs may be challenged. Moreover, the Westinghouse analysis assumed that PORVs are blocked about 50% of the time. If PORVs are blocked, the probability of a SV lifting is about ten times as high as if the PORVs are not blocked, according to Table 3-4 of WCAP-9804. Hence, for a plant with PORVs blocked all the time, it is appropriate to double the SV challenge frequency used by Westinghouse. In addition, Westinghouse assumed that the probability of a SV failing to close is 10^{-3}/demand, while the staff estimates this probability as 10^{-2}/demand. The net result is that the Westinghouse estimate of the SBLOCA-SV frequency should be multiplied by a factor of three because of the possibility of all three SVs being challenged, by a factor of two because PORVs at a given plant may be blocked all the time, and by a factor of 10 because of our estimate of the probability of a SV failing to close, per demand. This leads to an overall factor of 60, which leads to our SBLOCA-SV frequency of $5x10^{-6}$/reactor-year x 60 = $3x10^{-4}$/reactor-year. A similar estimate can be obtained from the fact there have been no challenges to SVs in Westinghouse plants in over 200 reactor-years of operation.

PORV Leakage Problem

The Westinghouse report stated that PORVs in the Westinghouse plants are blocked off about 55% of the time. The intentional blocking of PORVs is done to eliminate PORV leakage to ensure that the reactor coolant system (RCS) leakage does not exceed the technical specification limit. Since there are many Westinghouse plants which have blocked off PORVs, it may imply either that PORVs need to be modified to correct the leakage problem or that there should be some maintenance or repair work on PORVs on a periodic basis. A plant that operates with PORVs blocked off may depend on SVs to relieve pressure. Considering the fact that the SV capacity is much larger than the PORV capacity and there is no block valve to terminate a SV release, the consequences of a stuck-open SV may be more severe than those of a stuck-open PORV. In addition, if PORVs are not blocked off, they supply additional pressure relieving capacity in an ATWS (anticipated transient without scram) event. It would appear prudent to limit the time that plants operate with PORVs blocked. The staff is considering the need for imposing a technical specification limit on the amount of time a plant can operate with PORVs blocked. The need for upgrading the reliability of PORVs is a proposed generic issue (see the memorandum from D. DiIanni on the subject, "Proposed Generic Issue - PORV and Block Valve Reliability."[6])

CONCLUSION

Based on the review of the licensees' responses, the staff concurs, for the licensees given in Table 1, with the licensees' conclusions that the requirements of NUREG-0737, II.K.3.2 are met with the existing PORV, SV and high-pressure reactor trip setpoints, and the the automatic PORV isolation system is not required.

Table 1

PORV CHALLENGES IN WESTINGHOUSE PLANTS FROM APRIL 1, 1980 TO MARCH 31, 1983

PLANT	Number of PORV Challenges [1]
Beaver Valley	4
D. C. Cook 1	0
D. C. Cook 2	1
Farley 1	3
Farley 2	0
Ginna	2
Haddam Neck	2
Indian Point 2 & 3	0
Kewaunee	0
North Anna 1	2
North Anna 2	4[2]
Point Beach 1	1
Point Beach 2	0
Prairie Island 1 & 2	0
Robinson	0
Salem 1	2
Salem 2	0
San Onofre	2
Surry 1	2
Surry 2	0
Trojan	0
Turkey Point 3	0
Turkey Point 4	4[3]
Yankee Rowe	0
Zion 1	2
Zion 2	2

Notes:
[1] No SV challenges were known for the plants.
[2] (i) A PORV lifted twice at zero power when a RCS loop stop valve was opened and a reactor coolant pump (RCP) was started.
(ii) A PORV lifted twice when a RCP was started to vent air from RCS.
[3] Two of the PORV challenges were manual actuations for low-temperature overpressure protection.
[4] McGuire 1 is not included in the above list of Westinghouse plants because our contractor, FRC, has determined that WCAP-9804 is not applicable to McGuire 1 which has a different PORV design than other Westinghouse plants.

REFERENCE

1. WCAP-9804, "Probabilistic Analysis and Operational Data in Response to NUREG-0737, Item II.K.3.2 for Westinghouse NSSS Plants," Westinghouse Electric Corporation, February 1981.

2. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3. EPRI NP-801, "ATWS: A Reappraisal, Part III, Frequency of Anticipated Transients," July 1978.

4. WASH-1400, "Reactor Safety Study," October 1975.

5. NUREG/CR-2787, "Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One-Unit 1 Nuclear Power Plant," June 1982.

6. Memorandum dated June 6, 1983 from D. DiIanni for W. Minners through R. Clark, "Proposed Generic Issue - PORV and Block Valve Reliability.

Principal Contributor:
E. Chow

Attachment:
FRC Technical Evaluation
Report