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JAN 8 1980

The Honorable Vic Fazio
United States House of Representatives
Washington, D.C. 20510

Dear Congressman Fazio:

We apologize for the delay in responding to your letter of April 2, 1979, regarding a number of problems associated with the design and operation of nuclear plants. Our specific response to each question is enclosed.

In your letter you also expressed a concern regarding all Babcock & Wilcox facilities. With regard to these plants, full-time inspectors have been assigned to each operating plant utilizing Babcock & Wilcox pressurized water reactors, like those at Three Mile Island. In addition, the licensees of all these plants which were not shut down at the time of the Three Mile Island accident, except for Oconee, shut down their plants. (Oconee was permitted to make required NRC modifications to its three units on a staggered schedule because of the severe local impact on power availability that would have occurred had all three units been shut down simultaneously.) We then issued confirmatory orders to the licensees of all Babcock & Wilcox reactors to assure that necessary plant modifications, additional personnel training, and revised operating procedures were put into effect before these plants resumed operation.

I hope the information provided is responsive to your needs.

Sincerely,

ORIGINAL SIGNED BY R. G. SMITH

Lee V. Gossick
Executive Director for Operations

Enclosure:
Responses to Questions

OFFICE	TMI-2 TF	D/TMI Support	NRR	NRR	EDO	
SURNAME	SMiner/mc	RVollmer	ECase	HDenton	LGossick	
DATE	12/26/79	12/27/79	12/ /79	12/ /79	12/ /79	

QUESTION 1. Are the pressurizer, the pressurizer relief tank and the pressurizer relief valves in the Babcock and Wilcox systems large enough and well-enough designed? Are they the same at Rancho Seco as they are at Three Mile Island?

The pressurizer and pressurizer safety relief valves in the Rancho Seco reactor coolant system are the same as other Babcock and Wilcox 177-Fuel-Assembly pressurized water reactors, including those at Three Mile Island. A comparison of these and other components on B&W plants is listed in Tables 1 and 7 in the attached "Staff Report on the Generic Assessments on Feedwater Transients on B&W Plants" (NUREG-0560).

The determination as to whether B&W pressurizers are large enough is still under review. At this point, modifications in other areas have compensated somewhat for large variations in pressurizer levels following transients; however, this area of B&W design is still under study.

The capacity of power operated relief valves is based upon operational considerations only; it is not a factor in the evaluation of transients or accidents. Therefore, it has not been a subject of our review. We have required analyses for a stuck open valve with the given capacities which show acceptable results.

As to design, preliminary failure rate data indicate that the pressurizer relief valves on B&W reactors when opened, fail to reclose more frequently than relief valves on other pressurized water reactors (PWR). Design and reliability studies are still underway, however. We have required modifications to B&W reactors that have substantially reduced the number of challenges to the pressurizer relief valves while the design is being fully evaluated.

The size and design of the pressurizer relief tank are currently under review. The relief tank size and capacity did not play a significant role in the accident; therefore, short term modifications or studies were not deemed necessary.

As the foregoing information indicates, various B&W systems and components are currently undergoing review. Any changes that may result from these reviews would be required at the Rancho Seco facility. We believe that the modifications required and in place at the present time are adequate until NRC reviews and studies are complete. These modifications include:

a. a decrease in reactor high-pressure-trip set point and an increase in the power-operated relief valve open set point;
b. the addition of anticipatory reactor trip systems; and
c. additional transient and small-break analyses.

We also require that prior to January 1980 the relief valve indication and isolation capabilities be upgraded and that additional relief and safety-valve testing be conducted in the near future.

Our safety evaluations of Rancho Seco compliance with these requirements is attached. We have concluded that the licensee's actions in response to our requirements provide adequate assurance for the protection of the health and safety of the public during operation of Rancho Seco.

QUESTION 2. Are the design and reliability of the feed water and auxiliary feed water pumps at Rancho Seco adequate to avoid the failures which occurred at Three Mile Island?

The failure of the auxiliary feedwater (AFW) system at Three Mile Island Unit 2 to function properly on loss of main feedwater was not related to the design or reliability of the auxiliary feedwater pumps. The NRC recognizes that a loss of main feedwater transient is a relatively common occurrence in all nuclear power plants and our recent efforts as a result of lessons learned by the TMI-2 accident have been directed toward improving the reliability of auxiliary feedwater systems which serve as a backup to the feedwater system. Consequently, we have required that the licensee implement the following "short term" changes at Rancho Seco in order to accomplish this goal:

a. revise the operating, surveillance test, and emergency procedures in order to improve operator familiarity with and response to AFW system requirements and thereby improve system reliability;

b. verify that adequate auxiliary feedwater is supplied by system pumps for all postulated accident modes;

c. add flow-indicating instruments to the AFW system lines which readout in the control room in order to aid the operators in verify that flow is reaching the steam generators;

d. provide control room annunciation for all automatic-start conditions of the AFW system;

e. perform tests to verify that AFW system valves fail in the correct position upon loss of their power supply.

These actions were taken in response to our requirement and have been reviewed and found acceptable by the NRC staff. More specific details on the changes can be found in the attached Evaluation of the Licensee's compliance with the NRC Order of May 7, 1979. In addition, we are continuing our reviews of the Rancho Seco facility and will shortly develop additional "long term" requirements for the AFW system.

QUESTION 3. Should there not be a television camera and/or other surveillance devices inside the containment? Would not more remote sensing devices inside the pressure vessel itself give us a much better idea of what is happening during and after an accident?

The NRC Lessons Learned Task Force investigated the need for a television camera capability as well as many other interesting recommendations from many sources. Although a camera would yield some worthwhile information regarding conditions inside the containment, such information would not be necessary for plant safety. We think that other improvements to monitor conditions inside containment are necessary and these will be required on Rancho Seco. The attached letter from D. Eisenhut to all Operating Reactor Facilities of September 13, 1979, discusses some of these improvements, which include the following remote-sensing devices: containment water level, containment pressure, and hydrogen concentration.

As discussed in the attached "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations" (NUREG-0578), all operating reactor facilities are required within the next few months to provide analyses and to implement procedures that will enable them to recognize inadequate core cooling and to implement the corrective actions which must be taken to assure adequate core cooling. Over the longer term, an evaluation of the use of a water-level sensing device inside the reactor vessel must be conducted. Rancho Seco's response and commitments in these areas will be reviewed by our staff in the near future.

QUESTION 4. It appears that the monitoring system outside Three Mile Island does not generate comprehensive or useful data on the accumulation of low-level radiation. What is the NRC's requirement in this regard, and is there a need to upgrade it?

The radiological environmental monitoring system outside of Three Mile Island is specifically designed to detect low-level radiation. The TMI monitoring program therefore conforms with the program recommended in NRC's Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants." The radiological portion of this guide has been revised and temporarily issued as a Branch Technical Position (BTP) until it is again issued as a Regulatory

Guide. A draft of this BTP is attached. These standards are presently being updated to take into account accidents that may involve releases of higher-than-normal levels of radioactivity.

When the Three Mile Island incident occurred, we did increase the radiological environmental monitoring program to provide proper surveillance. Increased surveillance at Three Mile Island was provided by the utility, NRC, Department of Energy, the Environmental Protection Agency, and the State of Pennsylvania.

QUESTION 5. I am told that on the reactor head there is a vent line, and on the vent line there is a valve which could release the hydrogen bubble into the containment section and thereby resolve what has become the key difficulty in cooling the reactor down. Yet at Three Mile Island this valve is a manual valve. Why are automatic valves not required? In the case of Three Mile Island an automatic valve would enable us to avoid the risks we have taken thus far in our attempts to get the bubble out of the way.

Prior to the TMI-2 accident there was no requirement for automatic venting of the reactor coolant system high points. As you are aware, the noncondensible gases in the TMI-2 reactor coolant system were removed by degassification (spray and vent) in the pressurizer and makeup tank.

We have recently decided to require that an automatic vent be required in all pressurized water reactors. This is discussed in detail in the attached letter from D. Eisenhut to all Operating Reactor Facilities of September 13, 1979. Rancho Seco will be required to commit to and eventually install this venting capability.

QUESTION 6. There is a device at Three Mile Island which enables us to recombine hydrogen and oxygen into water, and avoid the possibility of hydrogen and oxygen building up to explosive proportions. However, this recombiner cannot be used because it is not within a shielded area, even though it is part of the safety system. Should not all recombiners be in an area equally as protected as the containment section. Are recombiners required everywhere?

(a) Prior to initiating operation of the existing recombiner it was considered prudent to install a redundant recombiner. Operation of the installed recombiner could not be initiated until the redundant recombiner and its piping were installed and tested. The additional shielding you refer to was added between the recombiner and its control panel to provide additional assurance that the operator would be protected from radiation. The shielding was not required to operate the recombiners.

In general, we require that all plant licensees provide for adequate access to vital areas (including hydrogen recombiners) and protection of safety equipment by design changes, increased permanent or temporary shielding or post-accident procedural controls.

(b) The current design basis for combustible gas (hydrogen) control outlined by federal regulations does not require hydrogen recombination capability for older plants (those whose notice of hearing on applications for construction permits occurred prior to November 5, 1970). However, Rancho Seco is equipped with hydrogen recombiners. The recently completed "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979 (NUREG-0578), concluded that further evaluation of the immediate requirement for use of recombiners at every nuclear plant power is required in light of more general questions involving the degraded core consequences (hydrogen generation) experienced at TMI-2. This study is continuing.

QUESTION 7. I understand that several human errors contributed to the mishap at Three Mile Island. Might more automated equipment have avoided some of these errors? Further, since the training given nuclear plant workers is substantially the same--particularly at Babcock and Wilcox plants--shouldn't we conclude that there are some critical weaknesses in the curriculum?

Several equipment malfunctions and human errors have been identified as contributing to the mishap at Three Mile Island. Accordingly, steps have been taken to address design changes (including more automation of equipment) and procedure changes necessary to assist the operator. These changes have been considered on both a short- and long-term basis. Immediate changes were required of all Babcock and Wilcox facilities prior to restart after the Commission ordered shutdown. Also, the Lessons Learned and the Bulletins and Orders Task Force have recommended design and operating requirements on all facilities in operation or under construction to be implemented in the near future. Longer term modifications for all facilities are set forth in "TMI-2 Lessons Learned Task Force - Final Report" (NUREG-0585). This report suggests basic changes and goals in nuclear power plant safety policy, design and operations, and in the regulatory process. Specific recommendations for achieving the goals outlined are discussed in the report appendices. We are enclosing a copy of the report for your examination.

The human errors experienced at Three Mile Island stemmed in part from the incorrect interpretation of Reactor Coolant System instrumentation. Operator training had not previously addressed the system and operator response to the set of conditions experienced during the March 28, 1979 transient. As a result, all licensed operators at Babcock and Wilcox facilities received special training and a written examination on the transient. Additionally, these operators attended training sessions on the Babcock and Wilcox simulator to reinforce the operator response required during similar transient situations. All training was completed prior to an operator's reassumption of licensed duties during power operations. Long term commitments on operator training and licensing are being developed by the Operator Licensing Branch.

QUESTION 8. Contaminated water is being pumped into the Susquehanna. Should more storage be available on nuclear sites for such emergency waste?

In general, the radioactive water storage capacity at the nuclear sites is adequate to cover design basis accidents as well as normal operating conditions. For accident situations where contaminated water in excess of storage capacity is generated there is usually sufficient time to acquire temporary storage facilities for the excess water as was done at Three Mile Island.

With regard to your concern about the release of contaminated water, except for releases of liquids containing only low or nondetectable levels of radioactivity to the Susquehanna River, such releases are not currently permitted. Before such releases take place, the impact will be evaluated by the NRC and the evaluation will be made available to the public. By this course of action, we will assure that a thorough assessment is completed prior to release of the contaminated liquids and that the health and safety of the offsite population will be protected.

As a result of releases containing only low or nondetectable levels of radioactivity, the levels of radioactivity in the Susquehanna are indistinguishable from existing background levels at public water supply intakes from the river. These levels have been confirmed by independent measurements made by the NRC, the Environmental Protection Agency (EPA), and the Commonwealth of Pennsylvania.

The Commission has recently authorized use of EPICOR-II water treatment system for processing the waste water stored in tanks in the auxiliary building. We do not currently permit the discharge of water processed by the EPICOR-II system, even though the system has proven more effective in processing water than predicted in our environmental assessment. The disposal of the water processed by EPICOR-II will be addressed in a separate environmental assessment, as required by the Commission's Statement of May 25, 1979. Copies of the environmental assessment will be made available for public comment. Since Metropolitan Edison has not yet submitted a proposal on the disposal method of the decontaminated water, we have yet to prepare the environmental assessment. We anticipate that several months will pass before any decisions are made on disposal of the water.

QUESTION 9. In Babcock and Wilcox plants are the circulatory pumps adequately designed? I understand they are not all at Three Mile Island. Is there any indication that leakage of the steam generator heat exchanging tubes has complicated the control and mitigation of this accident? How is the design of the primary cooling loop different or similar at Rancho Seco?

There are many aspects to the design of the reactor coolant pumps in Babcock and Wilcox plants (we assumed that you are referring to these pumps). These pumps are designed to provide sufficient flow while the reactor is operating at power. They are not required for decay heat removal following a reactor trip. Following a reactor trip, or normal shutdown, natural circulation in the reactor coolant system will provide core cooling if adequate subcooling inventory and heat sink (steam generator) are maintained. The difficulties with obtaining natural circulation following the TMI-2 accident occurred because all of these conditions were not present. Restart and continued operation of one reactor coolant pump was used to provide core cooling until April 27, 1979, when the pump was stopped and natural circulation cooling commenced.

There has been no indication since the accident that tube leakage exists in the B steam generator. This steam generator was isolated and has remained isolated since the accident. The accident sequence of events (attached) indicates that the operators thought a steam generator was leaking.

QUESTION 10. How content is the NRC with status of evacuation planning and implementation as a result of its experience here?

The emergency plans of all power reactor licensees have been reviewed by the staff in the past for conformance to the general provisions of Appendix E to 10 CFR Part 50. However, the most recent guidance on emergency planning, primarily that given in Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants," has not yet been fully implemented by most reactor licensees. Further, there are some additional areas where improvements in emergency planning have been highlighted as particularly significant by the Three Mile Island accident.

The NRC staff has undertaken an extensive effort to improve licensee preparedness at all operating power reactors and those reactors scheduled for an operating license decision within the next year. Meetings were held at our regional offices to discuss the recent impacts on emergency planning and current regulations. In addition, a Nuclear Regulatory Commission Emergency Planning Task Force will visit every operating reactor site in the country to evaluate upgrading on the site's emergency planning since the Three Mile Island accident. So far 18 site visits have been completed. This effort will be closely coordinated with a similar effort by the Office of State Programs to improve State and local response plans through the concurrence process and Office of Inspection and Enforcement efforts to verify proper implementation of licensee emergency preparedness activities. A rulemaking action is currently available for public comment that would require State and local government emergency plans as a condition for issuance of an operating license and for continued operation of a nuclear power plant.

Enclosures:
Listed on following page

Enclosures:
1. NUREG-0560
2. Evaluation of Licensee's Compliance with NRC Order
3. Letter dated 9/13/79 to All Operating Nuclear Power Plants
4. NUREG-0578
5. Branch Technical Position
6. NUREG-0591
7. Operational Sequence of Events

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4TH DISTRICT, CALIFORNIA

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JUDY KERN

April 2, 1979

Mr. Joseph M. Hendrie, Chairman
Nuclear Regulatory Commission
1717 H Street, N.W.
Washington, D.C. 20555

Dear Mr. Hendrie:

The events at Three Mile Island have brought to our attention a number of problems with the design and operation of nuclear power plants particulary plants designed by Babcock and Wilcox. As you know, there are nine B&W facilities in the United States. One of them, Rancho Seco, serves a portion of my district and is operated by the Sacramento Municipal Utilities District. Therefore, my constituents are legitimately concerned that any design or operational problems inherent in B&W facilities be corrected immediately.

Following are a number of specific questions which the Three Mile Island experience has raised. I know the NRC is preoccupied with the immediate crisis in Harrisburg, but the answers to these questions should be clear already and I think it is incumbent upon the NRC to disseminate them as soon as possible to the other B&W plants. In addition, I think it is the NRC's obligation to determine immediately whether B&W facilities should be shut down to correct any design defects, or whether any new operating procedures must be developed to compensate for equipment weaknesses. The public around these plants need restored confidence that its officials are dealing with the situation quickly and firmly. Unfortunately, the current confusion at Harrisburg does not create that impression.

NRR Are the pressurizer, the pressurizer relief tank and the pressurizer relief valves in the Babcock and Wilcox systems large enough and well-enough designed? Are they the same at Rancho Seco as they are at Three Mile Island?

NRR Are the design and reliability of the feed water and auxiliary feed water pumps at Rancho Seco adequate to avoid the failures which occured at Three Mile Island?

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PLEASE RESPOND TO:
X WASHINGTON ☐ VALLEJO ☐ SACRAMENTO ☐ WOODLAND ☐ YUBA CITY

NRR Should there not be a television camera and/or other surveillance devices inside the containment? Would not more remote sensing devices inside the pressure vessel itself give us a much better idea of what is happening during and after an accident?

NRR It appears that the monitoring system outside Three Mile Island does not generate comprehensive or useful data on the accumulation of low-level radiation. What is the NRC's requirement in this regard, and is there a need to upgrade it?

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In Babcock and Wilcox plants are the circulatory pumps adequately designed? I understand they are not all at Three Mile Island. Is there any indication that leakage of the stream generator heat exchanging tubes has complicated the control and mitigation of this accident? How is the design of the primary cooling loop different or similar at Rancho Seco?

MCR

How content is the NRC with the status of evacuation planning and implementation as a result of its experience here?

SP

I am most hopeful that you can make the answers to these questions available to the management and the public of these plants in the very near future. The NRC's leadership is urgently needed. Thank you very much.

Best regards,

Sincerely,

Vic Fazio

VIC FAZIO
Member of Congress

VF:pfc