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October 27, 1978

MEMORANDUM FOR: Edson G. Case, Deputy Director
Office of Nuclear Reactor Regulation

FROM: L. P. Crocker
Technical Assistant to the Director
Division of Project Management

SUBJECT: ACRS CONCERNS REGARDING DAVIS BESSE, UNIT 1

Background

During the 201st ACRS meeting in January 1977, the Committee reviewed the OL application for Davis Besse 1. The Committee issued a favorable letter on the application on January 14, 1977; however, in the letter, the Committee identified a number of matters which it felt warranted further effort. A copy of the letter is at Enclosure 2.

The staff met with the Committee during the 216th meeting in April 1978 to discuss the status of resolution of the matters mentioned in the ACRS letter. Following this status report, a Committee letter of August 25, 1978 (Enclosure 3) to the NRC Chairman reported on Committee activities during the preceding quarter and mentioned Davis Besse 1. The tone of the letter implies dissatisfaction with the staff efforts to clear up the Committee concerns.

This subject of staff response to ACRS recommendations (presumably to specifically include Davis Besse 1) is scheduled to be discussed by the Committee during its meeting with the Commissioners on November 2, 1978. In preparation for that meeting, following is the current status of staff actions regarding the ACRS recommendations.

STATUS

General - The plant still is assigned to DPM, although it now appears that transfer to DOR is imminent. A copy of the transfer package is attached as Enclosure 4.

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Committee Recommendations - A total of 9 recommendations were made by the Committee in its January 14, 1977, letter. These recommendations and the current status of resolution (as per the transfer package) are provided as Enclosure 1.

Original Signed by
Lawrence . . Crocker

L. P. Crocker
Technical Assistant to the Director
Division of Project Management

Enclosures:
1. Status
2. ACRS letter 1/14/78
3. ACRS letter 8/25/78
4. Transfer Package

cc: R. Boyd
D. Ross
D. Vassallo
J. Stolz
L. Engle

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STATUS OF STAFF ACTIONS

1. Seismic Reevaluation

Committee Recommendation

The structures and components of Davis-Besse, Unit 1, were designed for a Safe Shutdown Earthquake (SSE) acceleration of 0.15g at the foundation level. Because of changes in the regulatory approach to selection of seismic design bases, the Committee believes that an acceleration of 0.20g would be more appropriate for the SSE acceleration at a site such as this in the Central Stable Region. The Applicant presented the results of preliminary calculations concerning the safety margins of the plant for an SSE acceleration of 0.20g. The Committee recommends that the NRC Staff review this aspect of the design in detail and assure itself that significant margins exist in all systems required to accomplish safe shutdown of the reactor and continued shutdown heat removal, in the event of an SSE at this higher level. The Committee believes that such an evaluation need not delay the start of operation of Davis-Besse, Unit 1. The Committee wishes to be kept informed.

Status

Seismic Reanalysis

License condition 2.C.(3)(r) requires the licensee to submit a seismic reanalysis and evaluation to the Commission for its review and approval of the adequacy of the facility systems needed to accomplish safe shutdown of the reactor and continued shutdown heat removal prior to startup following the first scheduled refueling outage.

In performing the reanalysis, a safe shutdown earthquake acceleration of 0.20g shall be applied at the foundation level of the plant and the response spectra shall be used as specified in Regulatory Guide 1.60. Guidelines for the seismic reanalysis shall be specified by the staff.

The responsibility for specifying the guidelines for the seismic reanalysis to the licensee is the Division of Systems Safety. The assignment for preparing these guidelines is specified in TACS, #4925 (May 23, 1978). Evaluation of the seismic reanalysis will be by the Division of Systems Safety. Draft guidelines were issued on September 9, 1978 and the draft guidelines were discussed with the licensee on September 19, 1978. The schedule for completing this task is (1) Issuance of guidelines to licensee - November 1978, (2) Submittal by licensee to staff guidelines - March 1, 1979, (3) Staff's review of licensee's submittal complete - May 1, 1979, (4) Staff site visit for seismic audit - June 1, 1979, (5) Staff identification of items requiring follow-up action - July 1, 1979, (6) Resolution of follow-up items - October 1, 1979, and (6) Final Report to ACRS - November 1, 1979. Staff reviewer will be J. Rajan (MEB). Management responsibility will be carried out by Operating Reactors Branch No. 4. (Reference: SER Supplement No. 1, pgs. 2-3, 3-1, 18-1, 18-2, and E-2)

2. ECCS

Committee Recommendation

The performance of the Emergency Core Cooling System (ECCS) has been evaluated using a Babcock and Wilcox evaluation model applicable to the raised-loop configuration. The NRC Staff has reviewed these evaluations and has determined that certain assumptions regarding return to nucleate boiling do not comply strictly with the provisions of Appendix K to 10 CFR Part 50. The NRC Staff is also reviewing several other areas relating to ECCS performance. These matters should be resolved in a manner satisfactory to the NRC Staff.

Status

This matter has been resolved and documented in amendments to the license. See, also, SER Supplement No. 1, pp. 6-3 through 6-10.

3. State of Ohio

Committee Recommendation

In conjunction with the evaluation and assessment of the impact of routine waste releases from this plant, the Committee recommends that the NRC Staff provide leadership in encouraging the development of improved environmental radiation surveillance capabilities on the part of the State of Ohio and appropriate local regulatory agencies.

Status

During our April 1978 meeting with the ACRS, we stated that the State environmental radiological protection program was not sufficiently funded to warrant NRC funding support.

Since April, the staff has had continuing dialogue with the State representatives, and presently plans to send staff representatives to the State in the near future to discuss several alternatives which may allow some NRC funding support for the State program.

The problem has been and continues to be a lack of adequate funding by the State.

4. Hermetic Seals

Committee Recommendation

The Committee notes that post-accident operation of the plant to maintain safe shutdown conditions may be dependent on instrumentation and electrical equipment within containment which is susceptible to ingress of steam or water if the hermetic seals are either initially defective or should become defective as a result of damage or aging. The Committee believes that appropriate test and maintenance procedures should be developed to assure continuous long-term seal capability.

Status

This is an ACRS generic concern, number IId-2. It also is incorporated in the staff's Technical Activities Program as Task C-1.

At such time as results of Task C-1 indicate that some action is necessary for Davis Besse 1, we plan to react to this Committee recommendation.

5. Instrumentation to Follow the Course of an Accident

Committee Recommendation

The Committee recommends that, prior to commercial power operation of Davis-Besse, Unit 1, additional means for evaluating the cause and likely course of various accidents, including those of very low probability, should be in hand in order to provide improved bases for timely decisions concerning possible off-site emergency measures. The Committee wishes to be kept informed.

Status

This matter has to do with implementation of Regulatory Guide 1.97.

At such time as a decision is made regarding implementation of this Regulatory Guide on operating plants, we will implement it on Davis Besse 1.

6. ATWS

Committee Recommendation

The question of whether the design of this plant must be modified in order to comply with the requirements of WASH-1270, "Technical Report on Anticipated Transients Without Scram (ATWS) for Water-Cooled Reactors," remains an outstanding issue pending the NRC Staff completion of its review of the Babcock and Wilcox generic analyses of ATWS. The Committee recommends that the NRC Staff, the Applicant, and the Babcock and Wilcox Company continue to strive for an early resolution of this matter in a manner acceptable to the NRC Staff. The Committee wishes to be kept informed.

Status

This matter is addressed in the SER, p. 7-3, and in SER Supplement No. 1, p. 18-4. The licensee has made appropriate commitments regarding ATWS to allow licensing.

At such time as the ATWS issue is resolved for operating plants, we will require appropriate modifications for Davis Besse 1.

7. By-Pass Loop

Committee Recommendation

Davis-Besse, Unit 1, has installed a bypass loop containing two manually operated valves around the decay heat removal system suction line isolation valves. The normally closed bypass valves would be opened in the event of a spurious closure of one of the decay heat removal system suction line isolation valves during system operation. The Committee recommends that further attention be given to the means employed for isolation of the low pressure residual heat removal system from the primary system while the latter is pressurized, and that reliable means be developed to assure such isolation. This matter should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

Status

Design Modification Alternatives to the Present Key Lock Control in Manual Bypass Valves DH 21 and DH 23

Licensee condition 2.C.(3)(p) requires that the licensee submit an analysis of design modification alternatives for the present key lock control in the manual bypass valves DH 21 and DH 23 around the decay heat removal suction line valves to decrease the likelihood of the bypass path being opened inadvertently when isolation of the decay heat removal loop is required. The submitted analysis and installation of approved design modifications shall be completed prior to startup following the first scheduled refueling outage. Evaluation will be made by the Division of Operating Reactors. Management responsibility will be carried out by Operating Reactors Branch No. 4. (Reference: SER Supplement No. 1, p. 5-5 and p. E-3)

8. Fire Protection

Committee Recommendation

The Committee supports the NRC Staff program for evaluation of fire protection in accordance with Appendix A to Auxiliary and Power Conversion Systems Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." The Committee recommends that the NRC Staff give high priority to the completion of both owner and staff evaluations and to recommendations for Davis-Besse, Unit 1, and for other plants nearing completion of construction in order to maximize the opportunity for improving fire protection while areas are still accessible and changes are more feasible.

Status

Reevaluation of Fire Protection Program

License condition 2.C.(3)(h) of NPF-3 requires the licensee to increase the level of fire protection in the facility to the levels recommended in Appendix A to the Standard Review Plan 9.5.1, Revision 2, "Fire Protection System," or with alternatives acceptable to the staff. The level of facility fire protection as stipulated in item 2.C.(3)(h) shall be completed within three (3) years from the date of issuance of NPF-3.

License condition 2.C.(3)(h) also requires that the licensee implement Section B of Appendix A, "Administrative Procedures, Controls, and Fire Brigade," and Section C of Appendix A, "Quality Assurance Programs," prior to startup following the first regularly scheduled fueling outage.

By letter dated August 29, 1977, the licensee was provided a copy of NRC document, "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance," to be used as supplement guidance for the licensee's implementation of Sections B and C, Appendix A.

On October 13, 1977, a meeting was held with the licensee where the staff addressed the inadequacies of the licensee's Fire Hazard Analysis Report for Davis-Besse, Unit 1 submitted on Feburary 11, 1977. The licensee stated that they would resubmit an amended Fire Hazard Analysis Report in November 1977.

A meeting was held on December 6, 1977 where the licensee and the Auxiliary System Branch discussed the facility design for fire protection in the cable spreading room.

On January 11, 1978 the licensee submitted Revision No. 1 to the Fire Hazard Analysis Report for Davis Besse, Unit 1. Revision No. 1 was found to be acceptable and is presently under detailed review by the staff and staff's consultants. A fire protection site visit was completed on May 23-25, 1978, and staff requests for information were issued to the licensee on July 6, 1978. On August 1, 1978, a meeting was held with the licensee to clarify certain staff requests for information.

Major milestones scheduled for completing the Davis-Besse, Unit 1, Fire Protection Review are: (1) licensee's response to staff requests for information received on or before September 27, 1978; (2) meeting with licensee to resolve any remaining open items - October 25, 1978; (3) DSS and DPM SER input to ORB No. 4 - December 1, 1978; and (4) issuance of Davis-Besse, Unit 1, Fire Protection Evaluation Report - January 15, 1979.

Staff review and evaluation required to complete this task will be by Division of Systems Safety and Division of Project Management. DSS reviewer is V. Leung (ASB) and DPM reviewers are F. Allenspach and J. Conway (QAB) and J. Holman (OLB). (Note: The licensee states that they have committed to meeting BTP 9.5.1, Fire Brigade Training, in the Fire Hazards Analysis Report, Table 4.1, sheet 8.) Management responsibility will be carried out by Operating Reactors Branch No. 4. (Reference: SER, p. 9-14 and 9-15, SER Supplement No. 1, p. 9-1)

Interim Fire Protection Technical Specifications

The licensee was notified on November 21, 1977 by telephone and telecopy of our requirements for implementing interim fire protection Technical Specifications for plant systems and administrative procedures. A letter dated November 28, 1977 was sent to licensee providing the the sample standard Technical Specifications and requesting the licensee's response by December 7, 1977 for an application for license amendment with the plant specific interim technical specifications.

The licensee submitted proposed interim Technical Specifications on December 12, 1977 and on March 22, 1978 (Amendment No. 9) interim fire technical specification was issued for the presently installed fire protection equipment at the facility, as has been done for other operating facilities. The interim Technical Specifications deal only with administrative surveillance and corrective steps to reduce the likelihood of damaging fires pending our final review of the fire protection of Davis-Besse, Unit 1.

Resolution of this matter will be by Division of Systems Safety. DSS reviewer is V. Leung (ASB), and the schedule for completing this task is the same as specified for completing the Fire Protection Review (see Item 5, Enclosure 1). Management responsibility will be carried out by Operating Reactors Branch No. 4.

9. Sabotage

Committee Recommendation

The Committee believes that the Applicants and the NRC Staff should further review security provisions for Davis-Besse, Unit 1, for measures that could significantly reduce the possibility and consequences of sabotage, and that such measures should be implemented where practical.

Status

The Licensee's response to 10 CFR 73.55 was submitted on May 25, 1977. The staff review for Phase 1 was completed on September 8, 1977. A Modified Amended Security Plan was submitted in December 1977 and the review was completed in April 1978. A revised Modified Amended Security Plan was submitted in June 1978 and still is under review. In general, the licensee is in full compliance with staff requirements.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555
January 14, 1977

Enclosure 2

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: REPORT ON DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

Dear Mr. Rowden:

At its 201st meeting, January 6-8, 1977, the Advisory Committee on Reactor Safeguards completed its review of the application by the Toledo Edison Company and the Cleveland Electric Illuminating Company for a license to operate the Davis-Besse Nuclear Power Station, Unit 1. Members of the Committee visited the plant on May 18, 1976, and a subcommittee meeting was held in Washington, D.C. on December 21, 1976. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicant, the Babcock and Wilcox Company, the Bechtel Corporation, and the NRC Staff. The Committee also had the benefit of the documents listed. The Committee reported on the application for a construction permit for this unit on August 20, 1970.

The Davis-Besse Nuclear Power Station, Unit 1, is located on the southwestern shore of Lake Erie about midway between the cities of Toledo and Sandusky, Ohio. The minimum exclusion distance is 2400 ft. The low population zone, with a radius of two miles, included about 870 people in the 1970 census. The nearest population centers are Toledo (1970 population 383,818) and Sandusky (1970 population 32,674), both about 20 miles from the plant.

The nuclear steam supply system employs a Babcock and Wilcox pressurized water reactor similar in most respects to those first used in the Oconee Nuclear Station. This system differs from the Oconee units and several other similar units in that the steam generator loops are raised about 30 ft above the level in the original plant arrangement. Although this change was made to eliminate the need for internal vent valves, four such valves are provided because of their beneficial effect in reducing steam binding following a postulated loss-of-coolant accident.

The proposed power level for the unit is 2772 MWt, as compared to 2633 MWt proposed at the construction permit stage. This higher power level is the same as that proposed for the Rancho Seco and Three Mile Island, Unit 2 reactors, both of which have been reviewed by the NRC Staff and the Committee and found acceptable.

The structures and components of Davis-Besse, Unit 1, were designed for a Safe Shutdown Earthquake (SSE) acceleration of 0.15g at the foundation level. Because of changes in the regulatory approach to selection of seismic design bases, the Committee believes that an acceleration of 0.20g would be more appropriate for the SSE acceleration at a site such as this in the Central Stable Region. The Applicant presented the results of preliminary calculations concerning the safety margins of the plant for an SSE acceleration of 0.20g. The Committee recommends that the NRC Staff review this aspect of the design in detail and assure itself that significant margins exist in all systems required to accomplish safe shutdown of the reactor and continued shutdown heat removal, in the event of an SSE at this higher level. The Committee believes that such an evaluation need not delay the start of operation of Davis-Besse, Unit 1. The Committee wishes to be kept informed.

The performance of the Emergency Core Cooling System (ECCS) has been evaluated using a Babcock and Wilcox evaluation model applicable to the raised-loop configuration. The NRC Staff has reviewed these evaluations and has determined that certain assumptions regarding return to nucleate boiling do not comply strictly with the provisions of Appendix K to 10 CFR Part 50. The NRC Staff is also reviewing several other areas relating to ECCS performance. These matters should be resolved in a manner satisfactory to the NRC Staff.

In conjunction with the evaluation and assessment of the impact of routine waste releases from this plant, the Committee recommends that the NRC Staff provide leadership in encouraging the development of improved environmental radiation surveillance capabilities on the part of the State of Ohio and appropriate local regulatory agencies.

The Committee notes that post-accident operation of the plant to maintain safe shutdown conditions may be dependent on instrumentation and electrical equipment within containment which is susceptible to ingress of steam or water if the hermetic seals are either initially

defective or should become defective as a result of damage or aging. The Committee believes that appropriate test and maintenance procedures should be developed to assure continuous long-term seal capability.

The Committee recommends that, prior to commercial power operation of Davis-Besse, Unit 1, additional means for evaluating the cause and likely course of various accidents, including those of very low probability, should be in hand in order to provide improved bases for timely decisions concerning possible off-site emergency measures. The Committee wishes to be kept informed.

The question of whether the design of this plant must be modified in order to comply with the requirements of WASH-1270, "Technical Report on Anticipated Transients Without Scram (ATWS) for Water-Cooled Reactors," remains an outstanding issue pending the NRC Staff completion of its review of the Babcock and Wilcox generic analyses of ATWS. The Committee recommends that the NRC Staff, the Applicant, and the Babcock and Wilcox Company continue to strive for an early resolution of this matter in a manner acceptable to the NRC Staff. The Committee wishes to be kept informed.

Davis-Besse, Unit 1, has installed a bypass loop containing two manually operated valves around the decay heat removal system suction line isolation valves. The normally closed bypass valves would be opened in the event of a spurious closure of one of the decay heat removal system suction line isolation valves during system operation. The Committee recommends that further attention be given to the means employed for isolation of the low pressure residual heat removal system from the primary system while the latter is pressurized, and that reliable means be developed to assure such isolation. This matter should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

The Committee supports the NRC Staff program for evaluation of fire protection in accordance with Appendix A to Auxiliary and Power Conversion Systems Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." The Committee recommends that the NRC Staff give high priority to the completion of both owner and staff evaluations and to recommendations for Davis-Besse, Unit 1, and for other plants nearing completion of construction in order to maximize the opportunity for improving fire protection while areas are still accessible and changes are more feasible.

The Committee believes that the Applicants and the NRC Staff should further review security provisions for Davis-Besse, Unit 1, for measures that could significantly reduce the possibility and consequences of sabotage, and that such measures should be implemented where practical.

Other generic problems are discussed in the Committee's report, "Status of Generic Items Relating to Light Water Reactors: Report No. 4," dated April 16, 1976 (Attached). Those problems relevant to the Davis-Besse, Unit 1, should be dealt with by the NRC Staff and the Applicant as solutions are found. The relevant items are: II-1, 2, 3, 4, 6, 7, 9, 11; II.A-1, 4, 5, 7, 8; II.C-1, 2, 3, 4, 5, 6.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that the Davis-Besse Nuclear Power Station, Unit 1, can be operated at power levels up to 2772 MWt without undue risk to the health and safety of the public.

Sincerely yours,

M. Bender

M. Bender
Chairman

Attachment:
Status of Generic Items Relating to Light Water Reactors: Report No. 4 dated April 16, 1976

References:

1. Davis-Besse Nuclear Power Station, Unit 1, Final Safety Analysis Report (March 1973) with Revisions 1 through 24.

2. Safety Evaluation Report (NUREG-0136) in the matter of the Davis-Besse Nuclear Power Station, Unit 1 (December 1976).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

Enclosure 3

August 25, 1978

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON ACRS ACTIVITIES: MAY-AUGUST, 1978

Dear Dr. Hendrie:

This is a brief report of ACRS activities during May, June, July, and August 1978. Selected topics in this report will be discussed during the next joint NRC-ACRS meeting.

Proposed Use of CRAC Code in Site Comparisons

The Committee has considered the NRC Staff's proposed use of the consequence model (the "CRAC" Code), which was developed for the Reactor Safety Study (WASH-1400), for evaluating environmental impacts of alternatives to sites with relatively high population density.

Studies to date have shown that the CRAC Code can provide additional understanding of the public health impacts of accidents exceeding the limits of 10 CFR 100. However, there are many factors influencing the application of the code that have an important bearing on the computational results but which the code does not address adequately. These include regional meteorology (particularly for coastal and river valley sites), plume geometry, and effluent particle size distribution. In addition, the code does not address the behavior of radionuclides within containment prior to release. Also, the probability of various release magnitudes remains a factor having considerable uncertainty.

For this reason, the ACRS recommends caution in the use of the code as a determining basis for judgment in alternative site evaluations. Nevertheless, the CRAC Code is one of the more useful methods currently available for evaluating environmental impacts of alternative sites, and efforts should be continued to develop improved input data for the Code.

Dynamic Loading Combinations

The Committee has attempted on several recent occasions, including its 218th meeting, June 1-2, 1978, to encourage the Office of Nuclear Reactor Regulation to reconsider the rationale for establishing design basis loadings and loading combinations in performing safety analyses. The ACRS recommends that such a reevaluation be undertaken as soon as possible.

NRC Staff Response to ACRS Recommendations

During its April 1978 meeting the Committee heard a report regarding action taken in connection with ACRS recommendations in its January 14, 1977 report to you on operation of Davis-Besse Unit 1. Of the nine specific recommendations made by the Committee, action had been recently initiated on four and only preliminary work had been started on the other five.

Staff

On March 12, 1976, the Committee recommended prompt implementation of Reactor Pump Trip (RPT) for BWRs. Letters were sent to BWR licensees in May of 1978 by the NRC Staff specifying criteria for RPT design.

In order to obtain earlier responses to Committee recommendations, a follow-up system has been instituted which involves summarizing Committee requests within 6 months and a formal response by the Staff thereafter.

Reply to Congressman Udall

The Committee has forwarded a reply to Congressman Udall's inquiry about the need for a statutory board to review reactor safety matters, patterned after the National Transportation Safety Board. In its reply, the Committee stated that it did not believe that such a board was necessary, based on the assigned responsibilities and current activities of the NRC and ACRS and on the history of operation of commercial nuclear power plants.

Nuclear Plant Reliability Data System (NPRDS)

The Committee has considered a proposal to make a reliability data collection system mandatory for all licensees. In a memorandum to the Executive Director for Operations dated July 12, 1978, the Committee concluded that the existing system was providing data at a reasonable rate, but suggested also that the system could be improved by analyses of the internal consistency, usefulness, and reasonableness of the data already on hand. The Committee did not recommend that NPRDS reporting be made mandatory.

Implementation of RG 1.97, Revision 1, "Instrumentation to Follow the Course of an Accident"

The Committee has for some time been urging improvements in instrumentation to follow the course of accidents. In August 1977, Revision 1 to RG 1.97 was published but no firm schedule for its implementation has as yet been established. The NRC Staff is working with selected licensees to develop approaches to implement the Guide. The Committee will continue to monitor the implementation of this Guide.

Anticipated Transients Without Scram (ATWS)

The Committee's ATWS Subcommittee is proceeding with its review of the recently published NUREG-0460 - "Anticipated Transients Without Scram." The Committee hopes to reach a conclusion in this area by the end of 1978.

Generic Items

The Committee is planning the customary semiannual reevaluation of its list of unresolved generic items, and is planning to initiate a detailed review of the implementation of those items that are considered to have been resolved generically.

LOCA-ECCS Research Programs

The RES Staff has proposed that current LOCA-ECCS research programs be carried to completion, but that no new large LOCA-ECCS test facilities (e.g., the EBTF, Multi-Purpose Test Facility, etc.) be constructed. They have also recommended that the cooperative program with Japan, France and the Federal Republic of Germany be expanded. The ACRS concurs in these suggestions.

Future Schedule

222nd ACRS MEETING
OCTOBER 5-7, 1978

No Projects Scheduled

223rd ACRS MEETING
NOVEMBER 2-4, 1978

Zimmer (OL)

ONRR Safety Evaluation Report	9/1/78
ACRS Subcommittee Meeting	*
ACRS Report	11/9/78

BOPPSAR/BSAR-205 (PDA)

ONRR Safety Evaluation	10/2/78
ACRS Subcommittee Meeting	*
ACRS Report	11/9/78

GETR (Special Review)

ONRR Safety Evaluation Report	*
ACRS Subcommittee Meeting	*
ACRS Report	11/9/78

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

*To be scheduled

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

Docket No. 50-346

MEMORANDUM FOR: Victor Stello, Jr., Director
Division of Operating Reactors, NRR

FROM: Roger S. Boyd, Director
Division of Project Management, NRR

SUBJECT: TRANSFER OF DAVIS-BESSE, UNIT 1 FROM DIVISION OF PROJECT MANAGEMENT, LIGHT WATER REACTORS BRANCH NO. 1, TO DIVISION OF OPERATING REACTORS, BRANCH NO. 4

Effective on the date of this memorandum, the project management responsibility for Davis-Besse, Unit 1, is transferred from Light Water Reactors Branch No. 1, Division of Project Management, to Operating Reactors Branch No. 4, Division of Operating Reactors. Also, the environmental responsibility for Davis-Besse, Unit 1, is transferred from the Division of Site Safety and Environmental Analysis to the Division of Operating Reactors as delineated in Enclosure No. 4 to this memorandum.

The licensees, the Toledo Edison Company and the Cleveland Electric Illuminating Company, received a facility operating license (NPF-3) on April 21, 1977 which authorized full power operation at 2772 MWt. However, the operation of the facility was restricted to a sequence of operational modes until preoperational tests, startup tests, and other items were completed to the satisfaction of I&E and NRR.

A chronology for the issuance of the amendments to license NPF-3 and the date of authorization for proceeding to sequential operational modes is presented below. Also, a brief description is provided of the amendments presently issued for license NPF-3.

CHRONOLOGY

1.	License NPF-3 (Mode 6-Fuel Loading)	April 22, 1977
2.	Mode 5 - Cold Shutdown	May 10, 1977
3.	Amendment No. 1	May 27, 1977
4.	Amendment No. 2	June 14, 1977

5.	Amendment No. 3	June 24, 1977
6.	Mode 4 - Hot Shutdown	June 30, 1977
7.	Amendment No. 4	July 8, 1977
8.	Mode 3 - Hot Standby	July 8, 1977
9.	Amendment No. 5	July 21, 1977
10.	Mode 2 - Startup	August 9, 1977
11.	Reactor Critical	August 12, 1977
12.	Amendment No. 6	August 26, 1977
13.	Mode 1 - Power Operation	August 30, 1977
14	Amendment No. 7	November 29, 1977
15.	Amendment No. 8	February 28, 1978
16.	Amendment No. 9	March 23, 1978
17.	Amendment No. 10	May 26, 1978
18.	Amendment No. 11	June 16, 1978
19.	Amendment No. 12	August 18, 1978
20.	Amendment No. 13	September 29, 1978

Amendment No. 1 for NPF-3 revised the Technical Specifications to allow decay heat removal train pump switching operations for the purpose of testing in operational Modes 3, 4 and 5.

Amendment No. 2 revised license NPF-3 by removing license condition 2.C.(3)(o) which temporarily restricted facility operation to Mode 3 until the necessary modifications were implemented for ensuring that the decay heat removal relief valve would activate prior to automatic closure of the decay heat removal system isolation valves.

Amendment No. 3 revised the Technical Specifications as follows: (1) correcting administrative errors, (2) allowing surveillance testing of atmospheric vent valves ICS MA and ICS MB in Mode 4, (3) allowing surveillance testing for verification that the annulus space can be depressurized to one-quarter inch negative pressure within four seconds from the time that the emergency ventilation system fans attain 8000 cubic feet per

minute flow, (4) verifying that automatic isolation and interlock action of the decay heat removal system from the reactor coolant system will occur with a simulated reactor coolant system pressure of greater than or equal to 413 pounds per square inch gauge, and (5) allowing the interlock on either decay heat isolation valve to be taken out of service in order to perform the Channel Calibration or Channel Functional Test Requirements required in Modes 4 and 5. Amendment No. 3 also revised license NPF-3 by revising license condition 2.C.(3)(j) to allow the performance of specific or preoperational tests on decay heat removal valves DH 11 and DH 12 and requiring that an operator be stationed in the control room and assigned to monitor flow rates in the decay heat removal trains for those periods of time when surveillance testing is being performed or when a standby decay heat removal train is being brought on line.

Amendment No. 4 revised the Technical Specifications to reflect the proper allowable values and surveillance requirements for the steam generator level transmitters which had been installed in lieu of the level switches in the facility steam and feedwater rupture control system.

Amendment No. 5 revised license NPF-3 by deleting license condition 2.C.(3)(c) which stipulated the time allowed from date of issuance of NPF-3 for completing the installation of a second oxygen monitor in the gaseous radwaste system thereby providing redundant oxygen monitors that alarm locally and in the control room at specified set points.

Amendment No. 6 revised license NPF-3 by deleting license condition 2.C.(3)(b) which stipulated the time allowed from date of issuance of NPF-3 for completing the installation of the modified seismic Category 1 emergency diesel fuel oil and storage transfer system.

Amendment No. 7 revised the Technical Specifications as follows: (1) the allowable trip setpoint was revised to 7.0 $\pm$ 1.5 seconds for the Sequence Logic Channel of the Essential Bus Feeder Trip, and (2) the surveillance testing frequency for the Rosemont reactor coolant system pressure transmitters was changed from once every four months to once every eighteen months.

Amendment No. 7 revised license condition 2.C.(3)(k) by removing the stipulations within condition 2.C.(3)(k) requiring that acceptable noise test procedures be provided within four months from issuance of NPF-3. Also, license condition 2.C.(3)(l) was revised by removing the stipulations within 2.C.(3)(l) for providing large break spectrum analyses within six (6) months from issuance of NPF-3.

Amendment No. 7 deleted two license conditions 2.C.(3)(m) and 2.C.(3)(q) from NPF-3 which stipulated the period of time allowed to provide additional analysis and/or modifications in the facility design.

License condition 2.C.(3)(m) required making changes to the existing low pressure and high pressure injection flow system which would provide a flow indication syster hich was seismically qualified and powered from essential power source. and with flow indication in the main control room. License condition 2.C.(3)(q) required the submittal of an evaluation and modifications proposed to assure that required Class 1E equipment would operate properly in the event of offsite grid degradation.

Amendment No. 8 deleted the requirements for an Annual Operating Report in the Technical Specifications in order to be consistent with recent Commission guidance.

Amendment No. 9 revised the Technical Specifications to incorporate limiting conditions for operation and surveillance requirements for existing fire protection systems and administrative controls. The interim fire Technical Specifications were stipulated to become effective 30 days after issuance of Amendment No. 9.

Amendment No. 10 revised license NPF-3 by deleting license condition 2.C.(3)(n) which stipulated the time allowed from date of issuance of NPF-3 for completing the installation of flow measuring devices for boron dilution control.

Amendment No. 11 revised the Technical Specifications to allow full power operation for the duration of Cycle No. 1 with all Burnable Poison Rod Assemblies and all but two of the Orifice Rod Assemblies removed from the core. Amendment No. 11 revised license NPF-3 by deleting license condition 2.C.(3)(i) which stipulated the penalties for rod bow effects on the departure from nucleate boiling ratio. Amendment No. 11 also revised the Technical Specification regarding the modification of alarm setpoints on quadrant tilt to accommodate a recently discovered increase in the measurement error associated with this quantity.

Amendment No. 12 revised the Administrative Control Section of the Technical Specifications to reflect changes in the corporate structure of the Toledo Edison Company.

Amendment No. 13 revised the requirements for nonroutine environmental operating reports in order to make them consistent with Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants, December 1975."

The current status of items requiring futher staff actions and the organizations responsible for completing these items are identified in Enclosure 1. Lists of Generic Concerns and Regulatory Guides used during the license review, with reference to the locations where relevant information or evaluations of records e found, are provided in Enclosure 2 and 3, respectively.

By copy of this memorandum, Division of Systems Safety, Division of Site Safety and Environmental Analysis, Office of Inspection and Enforcement, Office of Management Information and Program Control, Office of Executive Legal Director, Regulatory Files, Public Information and Public Proceedings are being notified of the following changes in management responsibilities which are effective per date of this memorandum.

	FROM	TO
Project Manager	L. B. Engle	G. Vissing
Branch Chief	J. F. Stolz	R. Reid
Assistant Director	D. B. Vassallo	B. Grimes
Licensing Assistant	E. H. Hylton	R. Ingram

Roger S. Boyd, Director
Director of Project Management
Office of Nuclear Reactor Regulation

Enclosures:
1. Current Status of Items Requiring Staff Action
2. Current Status of Generic Review Items
3. Regulatory Guides Used During Licensing Review
4. DSE Transfer Memorandum (November 21, 1977)

ENCLOSURE 1

CURRENT STATUS OF ITEMS REQUIRING STAFF ACTION

DAVIS-BESSE, UNIT 1

DOCKET NO. 50-346

FACILITY OPERATING LICENSE NPF-3

The items requiring further staff action are as follows: (Note - A letter was sent to the licensee on September 20, 1978 requesting schedule dates for required submittals of analyses, modifications, and Technical Specifications as identified in Enclosure 1 which must be completed prior to startup following the first scheduled refueling outage.)

1. Overpressurization Protection for the Reactor Coolant System

   License condition 2.C.(3)(d) of Facility Operating License NPF-3 requires the licensee to install a long-term means of protection against reactor coolant system overpressurization. The installation shall be completed prior to startup following the first scheduled refueling.

   By letter date April 7, 1977, the licensee responded with proposed modifications to meet the long-term provisions for overpressure protection. By memorandum dated October 5, 1977 the Reactor System Branch provided their evaluation and approval of the licensee's proposal to install pressurizer heater interlocks and to remove power to the decay heat removal isolation valves DH 11 and DH 12 during decay heat removal operation. On March 30, 1978, the licensee submitted requested changes to the Technical Specifications regarding the proposed modifications. The licensee stated on July 7, 1978 that the proposed changes will be completed prior to startup following the first scheduled refueling outage. An augmented SER prepared by RSB, DSS approving the overpressure protection and Technical Specification changes is scheduled for issuance on October 30, 1978. An amendment is also in preparation approving the proposed Technical Specifications which will be implemented when the modifications are completed. Responsibility for issuance of the amendment will be either Light Water Reactors Group No. 1, DPM or Operating Reactors Branch No. 4, whoever has responsibility for the facility when the amendment is ready for issuance. (Reference: SER Supplement No. 1, p. 5-2, and memorandum from D. F. Ross to D. Vassallo dated October 5, 1977)

2. Reactor Coolant System Flow Indication

License condition 2.C.(3)(e) of NPF-3 requires the licensee to modify the flow indication of the reactor coolant system to meet the single failure criterion with regard to the pressure sensing lines to the flow differential pressure transmitters. This modification shall be completed prior to startup following the first scheduled refueling outage. Evaluation will be made by the Division of Operating Reactors and management responsibility will be carried out by Operating Reactors Branch No. 4. (Reference: SER, p. 7-2)

3. Diverse Source of Power for the Auxiliary Feedwater System

License condition 2.C.(3)(f) of NPF-3 requires the licensee to modify the auxiliary feedwater by providing diverse direct current power to one of the redundant auxiliary feedwater trains. This modification shall be completed prior to startup following the first scheduled refueling outage. The licensee's proposed modifications have been approved by the staff. Management responsibility will be carried out by Operating Reactors Branch No. 4. (Reference: SER, p. 9-7 and 9-8, and SER Supplement No. 1, p. 7-7)

4. Automatic Alignment of High Pressure Injection Pump Suction With the Low Pressure Injection Discharge

License condition 2.C.(3)(g) of NPF-3 requires the licensee to modify the emergency core cooling system by providing motor operated valves in lieu of the manually operated valves in each of the two crossover connection lines installed between the high pressure makeup pump suction and the low pressure injection discharge. The motor operated valves shall provide control and position indication in the control room. The modifications shall be completed prior to startup following the first scheduled fueling outage. Evaluation will be conducted by Division of Operating Reactors. Management responsibility will be carried out by Operating Reactors Branch No. 4. (Reference: SER, p. 6-8 and SER Supplement No. 1, p. 6-7)

5. Reevaluation of Fire Protection Program

License condition 2.C.(3)(h) of NPF-3 requires the licensee to increase the level of fire protection in the facility to the levels recommended in Appendix A to the Standard Review Plan 9.5.1, Revision 2, "Fire Protection System," or with alternatives acceptable to the staff. The level of facility fire protection as stipulated in item 2.C.(3)(h) shall be completed within three (3) years from the date of issuance of NPF-3.

License condition 2.C.(3)(h) also requires that the licensee implement Section B of Appendix A, "Administrative Procedures, Controls, and Fire Brigade," and Section C of Appendix A, "Quality Assurance Programs," prior to startup following the first regularly scheduled fueling outage.

By letter dated August 29, 1977, the licensee was provided a copy of NRC document, "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance," to be used as supplement guidance for the licensee's implementation of Sections B and C, Appendix A.

On October 13, 1977, a meeting was held with the licensee where the staff addressed the inadequacies of the licensee's Fire Hazard Analysis Report for Davis-Besse, Unit 1 submitted on Feburary 11, 1977. The licensee stated that they would resubmit an amended Fire Hazard Analysis Report in November 1977.

A meeting was held on December 6, 1977 where the licensee and the Auxiliary System Branch discussed the facility design for fire protection in the cable spreading room.

On January 11, 1978 the licensee submitted Revision No. 1 to the Fire Hazard Analysis Report for Davis Besse, Unit 1. Revision No. 1 was found to be acceptable and is presently under detailed review by the staff and staff's consultants. A fire protection site visit was completed on May 23-25, 1978, and staff requests for information were issued to the licensee on July 6, 1978. On August 1, 1978, a meeting was held with the licensee to clarify certain staff requests for information.

Major milestones scheduled for completing the Davis-Besse, Unit 1, Fire Protection Review are: (1) licensee's response to staff requests for information received on or before September 27, 1978; (2) meeting with licensee to resolve any remaining open items - October 25, 1978; (3) DSS and DPM SER input to ORB No. 4 - December 1, 1978; and (4) issuance of Davis-Besse, Unit 1, Fire Protection Evaluation Report - January 15, 1979.

Staff review and evaluation required to complete this task will be by Division of Systems Safety and Division of Project Management. DSS reviewer is V. Leung (ASB) and DPM reviewers are F. Allenspach and J. Conway (QAB) and J. Holman (OLB). (Note: The licensee states that they have committed to meeting BTP 9.5.1, Fire Brigade Training, in the Fire Hazards Analysis Report, Table 4.1, sheet 8.) Management responsibility will be carried out by Operating Reactors Branch No. 4. (Reference: SER, p. 9-14 and 9-15, SER Supplement No. 1, p. 9-1)

6. Potential for an Inadvertent Closure of a Decay Heat Removal System Valve During Shutdown Operations

License condition 2.C.(3)(j) of NPF-3, as amended by Amendment No. 3 to NPF-3, requires until such time as final resolution is obtained regarding the potential for and consequences of an inadvertent closure of a decay heat removal system valve during shutdown operations, the licensee is required to maintain power on decay heat removal valves DH 11 and DH 12 and can only operate one decay heat removal train at a time.

The condition does not preclude performance of specific surveillance or preoperational test requirements. For the period of times during which only one decay heat removal train is available or a standby heat removal train is being brought on line, an operator is required to be stationed in the control room to monitor flow rates and be available to immediately secure the reactor heat removal pumps should loss of flow occur due to the inadvertent closure of DH 11 or DH 12.

Condition 2.C.(3)(j) interfaces with Item 1, condition 2.C.(3)(d). Necessary actions required for resolution of these matters are the same as stated in Item 1 of this enclosure. Management responsibility will be carried out by Operating Reactors Branch No. 4. (Reference: SER Supplement No. 1, p. 5-4; Amendment No. 4 to Facility Operating License NPF-3, Safety Evaluation)

7. Design Modification Alternatives to the Present Key Lock Control in Manual Bypass Valves DH 21 and DH 23

Licensee condition 2.C.(3)(p) requires that the licensee submit an analysis of design modification alternatives for the present key lock control in the manual bypass valves DH 21 and DH 23 around the decay heat removal suction line valves to decrease the likelihood of the bypass path being opened inadvertently when isolation of the decay heat removal loop is required. The submitted analysis and installation of approved design modifications shall be completed prior to startup following the first scheduled refueling outage. Evaluation will be made by the Division of Operating Reactors. Management responsibility will be carried out by Operating Reactors Branch No. 4. (Reference: SER Supplement No. 1, p. 5-5 and p. E-3)

8. Seismic Reanalysis

License condition 2.C.(3)(r) requires the licensee to submit a seismic reanalysis and evaluation to the Commission for its review and approval of the adequacy of the facility systems needed to accomplish safe shutdown of the reactor and continued shutdown heat removal prior to startup following the first scheduled refueling outage.

In performing the reanalysis, a safe shutdown earthquake acceleration of 0.20g shall be applied at the foundation level of the plant and the response spectra shall be used as specified in Regulatory Guide 1.60. Guidelines for the seismic reanalysis shall be specified by the staff.

The responsibility for specifying the guidelines for the seismic reanalysis to the licensee is the Division of Systems Safety. The assignment for preparing these guidelines is specified in TACS, #4925 (May 23, 1978). Evaluation of the seismic reanalysis will be by the Division of Systems Safety. Draft guidelines were issued on September 9, 1978 and the draft guidelines were discussed with the licensee on September 19, 1978. The schedule for completing this task is (1) Issuance of guidelines to licensee - November 1978, (2) Submittal by licensee to staff guidelines - March 1, 1979, (3) Staff's review of licensee's submittal complete - May 1, 1979, (4) Staff site visit for seismic audit - June 1, 1979, (5) Staff identification of items requiring follow-up action - July 1, 1979, (6) Resolution of follow-up items - October 1, 1979, and (6) Final Report to ACRS - November 1, 1979. Staff reviewer will be J. Rajan (MEB). Management responsibility will be carried out by Operating Reactors Branch No. 4. (Reference: SER Supplement No. 1, pgs. 2-3, 3-1, 18-1, 18-2, and E-2)

9. Instrument-Station Ground Grid System

On August 25, 1977, the licensee presented and subsequently documented in letters on August 30, 1977 and September 16, 1977 their analysis for the effect of ground currents introduced into the station grounding grid from the worst postulated station electrical fault condition with inadvertent ties between the present instrument and station ground systems. The licensee's analysis showed that safety systems will still perform as intended.

The licensee committed to continue to analyze and test the instrumentation systems to identify and eliminate any inadvertent ties between the instrument and station ground grid and will provide to the NRC by no later than the end of the first scheduled refueling outage the results of their analysis, testing and corrective actions that have been implemented to assure that the installation meets the design criteria.

Also, the licensee committed to carefully monitor the instrumentation systems for any spurious operations and degrading conditions requiring corrections. Normal instrumentation monitoring will meet the definitions of a channel check (every 12 hours), channel functional test (every 30 days), and daily heat balance check as specified in the facility technical specifications for instrumentation systems. Also,

the licensee will maintain a record of their monitoring of these systems and will report in a timely manner any abnormalities to NRC (Inspection & Enforcement, Region 3). The staff concluded that the plant can be operated safely based on the information and commitments provided by the licensee.

Evaluation of the analysis, testing, and corrective actions and operational reports will be by the Division of Operating Reactors. Management responsibility will be by Operating Reactors Branch No. 4. (References: Toledo Edison Company letters from L. Roe to D. Vassallo dated August 30, 1977 and September 16, 1977, and NRC letter from D. Vassallo to L. Roe dated September 26, 1977)

10. Inservice Inspection and Testing Programs

By NRC letter dated April 22, 1977 the licensee was granted written relief from the requirements of Section XI of the ASME Code for r and valves testing program in accordance with Technical Specif 4.0.5a from the date of issuance of the facility operating li the start of facility commercial operation.

Relief was not, however, granted for the time after commercial ope tion. NRC letter dated April 22, 1977 required the licensee to submit information which identifies any request for written relief according to the requirements of Technical Specification 4.05b and 10 CFR 50.55a(g)(6)(i). On June 29, 1977 the licensee submitted their Pump and Valve Test Program and submitted on November 22, 1977 the Inservice Inspection Program for compliance with 10 CFR 50.55a(g). A letter granting relief from the ASME Section XI Inservice Inspection (Testing) Requirements is being issued to the licensee in October 1978. The relief is being granted until such time that the NRC staff's detailed review is complete. The licensee's submittal is presently being evaluated by the MEB, Division of Systems Safety. Staff Q-1's are scheduled to be issued June 1, 1979 and final resolution of these matters is scheduled for September 1, 1979. The DSS reviewer assigned to complete this task is J. Rajan (MEB). Management will be by Operating Reactors Branch No. 4.

11. Interim Fire Protection Technical Specifications

The licensee was notified on November 21, 1977 by telephone and telecopy of our requirements for implementing interim fire protection Technical Specifications for plant systems and administrative procedures. A letter dated November 28, 1977 was sent to licensee providing the the sample standard Technical Specifications and requesting the licensee's response by December 7, 1977 for an application for license amendment with the plant specific interim technical specifications.

The licensee submitted proposed interim Technical Specifications on December 12, 1977 and on March 22, 1978 (Amendment No. 9) interim fire technical specification was issued for the presently installed fire protection equipment at the facility, as has been done for other operating facilities. The interim Technical Specifications deal only with administrative surveillance and corrective steps to reduce the likelihood of damaging fires pending our final review of the fire protection of Davis-Besse, Unit 1.

Resolution of this matter will be by Division of Systems Safety. DSS reviewer is V. Leung (ASB), and the schedule for completing this task is the same as specified for completing the Fire Protection Review (see Item 5, Enclosure 1). Management responsibility will be carried out by Operating Reactors Branch No. 4.

12. Modification to Spent Fuel Storage Capacity

By letter dated December 19, 1977, the licensee requested an amendment to License NPF-3 for augmenting the existing spent fuel storage capacity. A Federal Register Notice was filed on March 8, 1978 specifying a time limit of April 14, 1978 for filing a timely petition to intervene. No timely nor tardy petitions to intervene have been received as of October 10, 1978. On March 8, 1978, staff requests (EEB/DOR) for additional information regarding the licensee's submittal of December 19, 1977 was sent to the licensee. The licensee's response to the staff's request for information was received on April 4, 1978.

Subsequent to the licensee's submittal dated April 4, 1978, problems associated with the removal of the Burnable Poison Rod Assemblies (BPRAs) and Orifice Rod Assemblies (ORAs) from the core after 87 effective full power days of operation required transferring the BPRAs and ORAs through the spent fuel pool to the adjacent cask pit (see Amendment No. 11, June 16, 1978). During this transfer contamination occurred and the licensee provided revised information on June 22, 1978 regarding their April 4, 1978 response to the staff's request for information dated March 8, 1978.

The Division of Operating Reactors schedule for completing the EEB/DOR questions and sections of the SER and Environmental Impact Appraisal for the spent fuel modifications is approximately three weeks after transfer to DOR. (See Memorandum dated September 12, 1978 from L. Barrett to T. Carter - EEB/DOR schedule to complete spent fuel modifications.)

The Division of Systems Safety will evaluate the remaining sections required for issuance of the SER for the spent fuel modifications. The DSS reviewer is V. Leung (ASB) and the schedule for completing the

task is: (1) Q-1's (if needed) to be issued on November 15, 1978 and (2) completing the SER is scheduled for January 5, 1979.

Management responsibility will be carried out by Operating Reactors Branch No. 4.

ENCLOSURE 2

CURRENT STATUS OF GENERIC REVIEW ITEMS

DAVIS-BESSE, UNIT 1

A. Items which have been resolved and require no further NRC action. (For Enclosure 2 all items are identified with number which refers to generic review index, page 3-2, for the October 21, 1977 NUREG-0328, Vol. 4, No. 5-Pink Book.)

ITEM	REFERENCE
1. Containment Leak Testing - Appendix J (3-11)	SER, p. 6-5, TS p. 3/4 6-2 through p. 3/4 6-4
2. ECCS FAC Evaluation (3-13)	SER, Supplement No. 1, p. 6-3 through p. 6-10
3. Effluent Treatment Systems (ALARA) (3-15)	SER, p. 11-1 through p. 11-2
4. Emergency Planning (3-16)	SER, p. 13-3 through p. 13-5
5. Filter Tech Specs (3-17)	TS p. 3/4 7-17 through p. 3/4 7-19
6. Flood of Equipment Important to Safety (3-19)	SER, p. 10-3 and Supplement No. 1, p. 6-8
7. Fracture Toughness Requirements - Appendix G (3-20)	TS p. 3/4 4-24 through p. 3/4 4-29
8. Fuel Cask Drop Analysis (3-21)	SER, p. 9-2
9. Fuel Handling Accident Inside Containment (3-22)	Letter to applicant dated July 29, 1977 with NRR Safety Evaluation enclosed
10. High Energy Line Break (3-24)	SER, p. 3-7 through 3-10
11. Inservice Inspection PWR Steam Generator Tubes (3-26)	SER, p. 5-8 and TS 3/4 4-6 through p. 3/4 4-12

ITEM	REFERENCE
12. Potential Equipment Failure Associated with Degraded Grid	SER, p. 8-2, Supplement No. 1, p. 8-1, and Amendment No. 7 Safety Evaluation
13. PWR Secondary Water Chemistry Monitoring Requirements (3-37)	SER p. 5-8 and TS 3/4 7-10
14. QA Program for Operations (3-38)	SER, p. 17-1 to 17-6 and p. 14-1
15. Qualifications of Radiation Protection Manager (3-39)	Necessary action completed prior to issuance of OL - April 22, 1977
16. Hydraulic Snubbers (3-43)	TS 3/4 7-20 through 7-35
17. Steam Generator Feedwater Flow Instability (SGFWFI) (3-44)	NRC generic letter on SGFWFI sent to licensee on September 6, 1977 which stated B&W steam generators have not had this problem
18. Respiratory Protection Program (3-42)	Necessary action completed prior to issuance of OL - April 22, 1977
19. Deletion Technical Specification Requirements for Annual Operating Report	Amendment No. 8 issued on February 28, 1977 deleted the Technical Specification Requirements for an Annual Operating Report
20. Interim Fire Technical Specifications (except as noted in Item 11, Enclosure 1)	Amendment No. 9 issued on March 22, 1978
21. Fuel Rod Bow Effects (3-22)	Amendment No. 11 issued on June 16, 1978
22. PWR Reactor Vessel Seal Ring Missile	By letter dated March 24, 1978, the licensee stated that the cavity annulus seal ring is not left in place during normal operation

B. Items which have been evaluated in light of current NRC requirements/guidance and for which measures that will make the status acceptable have been initiated by the licensee.

ITEM	REFERENCE
1. Industrial Security	Licensee's response to 10 CFR Part 73.55 submitted on May 25, 1977. NRC staff review for Phase I completed September 8, 1977. Licensee's Modified Amended Security Plan submitted December 9, 1977. Review completed April 21, 1978. Revised MASP submitted June 5, 1978 and in Review.
2. Anticipated Transients Without Scram (3-6)	NRC letters to licensee dated October 10, 1973 and December 6, 1974. Licensee letters to NRC dated January 24. 1974 and October 28, 1974. (SER, p. 7-3 and SER Supplement No. 1, p. 18-4)
3. Fire Protection (3-18)	See Items 5 and 11 of Enclosure 1
4. Inservice Inspection Program (3-27)	See Item 10 of Enclosure 1

C. Items which have been evaluated in light of current NRC requirements/guidance and which are presently unresolved. Further action by NRC and/or the licensee may be required in the future.

ITEM	REFERENCE
1. Reactor Pressure Vessel Supports (Asymetric LOCA loads) (3-5)	NRC letter to licensee dated November 21, 1975. Licensee's letters to NRC on December 19, 1975 and August 28, 1976. NRC letter dated January 20, 1978 requested licensee to proceed with an evaluation of asymmetric LOCA loads. By letter dated May 5, 1978, the licensee provided a schedule for completing Phase 1 of the evaluation by December 31, 1978. Scheduling was defined at the B&W Owners Group and NRC/DOR meeting held on March 31, 1978. (SER, p. 3-14)
2. Anticipated Transients Without Scram (3-6)	NRC letters to licensee dated October 10, 1973 and December 6, 1974. Licensee letters to NRC dated January 24, 1974 and October 28, 1974. (SER, p. 7-3 and SER Supplement No. 1, p. 18-4)
3. Fire Protection (3-18)	See Items 5 & 11 of Enclosure 1
4. Upgrading STS Bases Program	NRC letter to licensee August 19, 1977. Licensee attended generic meeting on October 4, 1977

D. Items which have been evaluated in light of current NRC requirements/guidance and which are unacceptable.

ITEM	REFERENCE
None	

E. Items which have not been evaluated in light of current NRC requirements/ guidance. These items will be evaluated by DOR.

ITEM	REFERENCE
1. Diesel Generator Lockout (3-12)	NRC letter to licensee on September 2, 1977. Licensee letter to NRC on October 29, 1977
2. Fracture Toughness and Potential for Lamellar Tearing of Steam Generator and Reactor Coolant Pump Materials	NRC letter to licensee on September 8, 1977. Licensee letter to NRC on October 10, 1977 stating submittal to NRC on November 11, 1977. Applicant's submittal received on November 11, 1977. This review is being accomplished under Category A, Task A-12.
3. PWR Moderator Dilution (3-36)	NRC letter sent to licensee on September 16, 1977. Licensee's response was submitted on December 16, 1978.
4. PWR HPSI & LIPSI Flow Resistance (3-35)	NRC letter sent to licensee on November 9, 1977. Licensee's response received on January 13, 1978. This review is being accomplished under Pink Book Generic Issue #053.
5. Implementatic of Appendix I Standard Technical Specifications	Letter with enclosed model specifications sent to applicant on July 10, 1978. Licensee's submittal schedule is 180 days from issuance of letter.

F. Items which are not applicable to Davis-Besse, Unit No. 1.

ITEM	REFERENCE
All item identified as specific to GE, CE or Westinghouse plants.	

ENCLOSURE 3

REGULATORY GUIDES USED DURING

THE LICENSING REVIEW

FOR DAVIS-BESSE, UNIT 1

REGULATORY GUIDE NUMBER	REGULATORY GUIDE TITLE	REFERENCE TO SER, SER SUPPLEMENT NO. 1, AND APPLICABLE LICENSE AMENDEMENTS
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps	SER p. 6-3
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors	SER p. 2-10 and Supplement No. 1 p. 2-3
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	SER p. 8-1 and p. 8-3
1.7	Control of Combustible Gas Concentrations	SER p. 6-5 and 6-6
1.8	Personnel Selection and Training	SER p. 13-3
1.9	Selection of Diesel Generator Set Capacity for Standby Power Supplies	SER p. 8-1
1.11	Instrument Lines Penetrating Primary Reactor Containment	SER p. 6-5
1.12	Instrumentation for Earthquakes	SER p. 3-17
1.13	Fuel Storage Facility Design Bases	SER p. 9-2, 9-3 and 9-13

1.17	Protection of Nuclear Power Plants Against Industrial Sabotage	SER p. 13-16
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	SER p. 4-6
1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants	SER p. 11-10
1.23	Onsite Meteorological Programs	SER p. 2-9
1.27	Ultimate Heat Sink for Nuclear Power Plants	SER p. 2-8, 2-15
1.28	Quality Assurance Program Requirements (Design and Construction)	SER p. 17-1
1.29	Seismic Design Classification	SER p. 9-3 and 9-16
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Equipment	SER p. 17-1
1.31	Control of Stainless Steel Welding	SER p. 4-6 and 5-5
1.33	Quality Assurance Program Requirements (Operation)	SER p. 13-6
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Light Water-Cooled Nuclear Power Plants	SER p. 17-1

1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	SER p. 17-1
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	SER p. 17-1
1.44	Control of the Use of Sensitized Stainless Steel	SER p. 4-5 and 4-6
1.45	Reactor Coolant Pressure Detection Systems	Supplement No. 1, P. 5-2
1.46	Protection Against Pipe Whip Inside Containment	SER p. 3-7
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	Supplement No. 1, p. 7-7
1.48	Design Limits and Loading Conditions	SER p. 3-15 and 5-1
1.52	Design, Testing and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants	SER p. 9-11
1.54	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants	SER p. 17-1
1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel	-
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	SER p. 3-10 and Supplement No. 1, p. 2-2, 3-1, and 18-1

1.61	Damping Valves in Seismic Design of Nuclear Power Plants	SER p. 3-10
1.64	Quality Assurance Requirements for the Design of Nuclear Power Plants	SER p. 17-1
1.67	Installation :ˆ Over Pressure Protection Devices	SER p. 3-16
1.68	Preoperational and initial Startup Test Programs for Water-Cooled Power Reactors	SER p. 14-1
1.74	Quality Assurance Terms and Definitions	SER p. 17-1
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors	SER p. 15-4
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	SER p. 6-10 and 14-1
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes	SER p. 5-8
1.84	Code Case Acceptability ASME Section III Design and Fabrication	SER p. 3-3 and p. 5-2
1.85	Code Case Acceptability - ASME Section III Materials	SER p. 3-3 and p. 5-2
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	SER p. 17-1

1.95	Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release	SER p. 6-11
1.111	Methods for Estimating Transport and Dispersion of Gaseous and Liquid Effluents in Routine Releases from Light-Water Cooled Reactors	SER p. 2-11
8.8	Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable (Nuclear Reactor)	SER p. 12-1 and 12-3
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Practicable	SER p. 12-1

ENCLOSURE NO. 4

11/21/77

Docket No.: 50-346

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MEMORANDUM FOR: Roger S. Boyd, Director, Division of Project Management

FROM: Harold R. Denton, Director, Division of Site Safety and Environmental Analysis

SUBJECT: TRANSFER OF DAVIS-BESSE UNIT NO. 1 TO OPERATING REACTORS BRANCH NO. 3

We understand that you are planning to transfer Davis-Besse Unit 1 from DPM to DOR; this memorandum should be attached as an enclosure to your transfer memorandum to effect transfer of the environmental review responsibility from Environmental Projects Branch No. 1 to Operating Reactors Branch No. 3. This transfer of environmental responsibility is to be effective as of the date of your transfer memo.

There are no uncompleted environmental review tasks associated with this unit.

At the time of transfer of responsibility, I&E, MIPC, and others on the distribution for this memo should be notified of the following changes:

	From	To
Environmental Project Manager	P. Cota	J. Hannon
Branch Chief	G. Knighton	G. Lear
Assistant Director	V. Moore	K. Goller
Licensing Assistant	D. Slater	C. Parrish

Original Signed by
H. R. Denton

Harold R. Denton, Director
Division of Site Safety
and Environmental Analysis
Office of Nuclear Reactor Regulation

cc: See next page

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