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AUG 28 1975

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Docket No. 50-346

Voss A. Moore, Jr., Assistant Director for LWR's, Group 2, RL

DAVIS-BESSE 1 SER INPUT

Plant Name: Davis-Besse 1
Docket No.: 50-346
Milestone No.: 01-24
Licensing Stage: OL
REsponsible Branch and Project Manager: LWR 2-3, L. Engle
Technical Review Branch Involved: Reactor Systems Branch
Description of Review: SER Input
Requested Completion Date: August 15, 1975
Review Status: Complete

The enclosed report contains the evaluation performed by the Reactor Systems Branch on Davis-Besse 1. Reactor Systems review included Sections 1.5, 4.1, 4.4, 5.1, 5.2.2, 5.3, 5.5, 6.3.1, 6.3.2, 6.3.3, 6.3.4 and Chapter 15.0 from the Standard Format, Revision 1.

There remains several areas of the Davis-Besse 1 design which require commitment by the applicant:

1. Upgrading the valves in the crossover lines between the LPIS and HPIS from hand-operated to motor-operated.

2. Upgrading the crossover line betwwen each LPIS train from active components to passive components.

3. Adding the capability for leak-testing the check valves in the low pressure-high pressure LPIS interface.

The applicant was officially informed of the first two items by letter on December 26, 1973. Our position on the third item was transmitted by letter to the applicant on April 18, 1975. Also, we require additional information in several areas which could possibly lead to design changes. These areas are further specified in the enclosed text. In addition, see Sectinn 1.6 for a more detailed summary of a number of technical and administrative changes.

Original Signed by
Victor Stello
Victor Stello, Jr., Assistant Director
for Reactor Safety
Division of Technical Review

Enclosure:
Davis-Besse 1 Evaluation

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1.5 Requirements for Further Technical Information

The applicant has identified in Section 1.5 of the Davis-Besse 1 FSAR development programs applicable to the Davis-Besse 1 design. These programs were initiated to establish the final design and have each been completed. We conclude that the research and development test programs outlined in the Davis-Besse 1 FSAR provide the information necessary for the design and safe operation of Davis-Besse 1.

1.6 Facility Modifications as a Result of Regulatory Staff Review

During the review of the Davis-Besse 1 operating license application, the applicant proposed or we requested a number of technical and administrative changes. These changes are described in various amendments to the original application. We have listed below the more significant modifications that have been or will be required to be made as a result of our review. The sections of this report where these matters are discussed more fully are noted in parenthesis.

1. Upgrading the valves in the crossover lines between the LPIS and HPIS from hand-operated to motor-operated (6.3.2).
2. Upgrading the crossover line between each LPIS train from active components (motor-operated valves) to passive components (cavitating venturi) (6.3.2).
3. Addition of testing components for the low pressure-high pressure LPIS interface (check valves CF 30/31 and DH 76/77) (5.5.7).
4. Addition of preoperation tests to demonstrate operability of the local manual handwheel backup on each ECCS valve, and to demonstrate the capability of the ECCS to operate in the recirculation mode (6.3.4).
5. Changes in the Davis-Besse 1 Technical Specifications to prohibit all partial loop operation (4.4).
6. Addition of a core flow penalty in the thermal-hydraulic design of the RCS and core to account for a potentially stuck open vent valve (4.4).

7. Addition of a periodic surveillance requirement in Technical Specifications for venting of all ECCS lines and pump casings to minimize the potential for a water hammer (6.3.4).

4.0 REACTOR

4.1 Summary Description

The design of the B&W reactor for Davis-Besse 1 is similar to the design of other pressurized water reactors that we have recently approved for operation. The core consists of 177 fuel assemblies having 208 fuel rods each; the design heat output of the core is 2772 MWt, which is the same as the design output for the Rancho Seco core. Full and part length control rods, dissolved boron, and burnable poison rod assemblies (BPRA) are used for reactivity control.

A unique feature of the B&W design is internal vent valves which minimize steam binding in the event of loss-of-coolant accident (LOCA). The primary difference between Davis-Besse 1 and Rancho Seco is the raised steam generators in Davis-Besse. These higher steam generators further decrease the potential for steam binding in the event of a LOCA.

4.4 Thermal and Hydraulic Design

The Davis-Besse 1 reactor is designed to operate at core power levels of up to 2772 MWt, which corresponds to a net electrical output of about 906 MWe. We have evaluated the thermal hydraulics on the basis of 2772 MWt. Davis-Besse 1 will utilize a 15x15 fuel assembly as in the Rancho Seco plant. As shown in Table 4.4-1, the thermal and hydraulic design parameters for the two plants are similar.

The principal criterion for the thermal-hydraulic design of a reactor is to prevent fuel rod damage by providing adequate heat transfer for the various core heat generation patterns occurring during normal operation and anticipated operational transients. Maintenance of nucleate boiling is a basic objective of a thermal-hydraulic design. The applicant has demonstrated, through the use of the Westinghouse W-3 correlation, that a departure from nucleate boiling heat flux ratio (DNBR) greater than 1.30 is maintained for steady state and anticipated transient conditions.

We have required that the applicant consider the effect of a stuck open vent valve on the analyses of the thermal-hydraulic design of the reactor coolant system and core and for all transients. Before power operation, the applicant must either; (1) submit the reanalyses, (2) show that a stuck open vent valve would be detected by an operator, or (3) show that vent valves are not sticking open on operating reactors. Inasmuch as the applicant has not presented information regarding items (2) and (3), the staff requires that one valve less than the minimum detectable number of stuck open vent valves be assumed open and the corresponding core flow penalty be imposed for the thermal-hydraulic design of the RCS and core. The applicant is required to provide this analysis which will be evaluated to determine the maximum power level of the system. Further, we will also require that the vent valves be tested during each refueling.

Another parameter that influences the thermal-hydraulic design of the core is rod-to-rod bowing within fuel assemblies. The applicant analytically predicted the amount of bowing which could occur at the cladding hot spot during an operational transient with an assumed DNB condition. The analysis was performed at 100% power with an assumed abnormal flow condition (inlet flow blockage) that would cause DNB.

Operation at these postulated conditions resulted in a calculated maximum fuel rod cladding temperature of 1025 F and bowing of 49 mils. During the Oconee 1 refueling, six fuel assemblies were examined visually and dimensionally. Water channel and line scan measurements indicated a maximum rod bow of approximately 30 mils. B&W feels that the observed rod bow is accommodated within the current design approaches and is pursuing a program to demonstrate this. B&W generically plans to develop bow correlations and predictive techniques to analyze the data and the predicted bow from a thermal-hydraulic standpoint. The staff intends to follow this program and will consider the application of our conclusions to Davis-Besse 1.

Since the applicant does not propose to validated operation of the plant in a single loop configuration (i.e., two pumps in one loop running while both pumps in the other loop are idle), the Technical Specifications will prohibit single loop operation. Also, the applicant has been requested to further support other partial loop configurations by providing a LOCA analysis during operation in this mode. Until this analysis has been reviewed by the staff, Technical Specifications will not allow partial loop operation.

On the basis of our review of the thermal-hydraulic characteristics of Davis-Besse 1, including a comparison with the previously approved Rancho Seco, we conclude that with the stipulations noted above, the thermal-hydraulic design of Davis-Besse 1 is acceptable.

Table 4.4-1 Thermal-Hydraulic Design Summary Comparison of Davis-Besse 1 and Rancho Seco

	Davis Besse	Rancho Seco
Design Core Heat Output, MWt	2772	2772
Nominal System Pressure, psia	2200	2200
Vessel Coolant Inlet Temperature, °F	555.4	557
Vessel Coolant Outlet Temperature °F	608.6	607.7
Total Heat Transfer Surface Area in Core, ft^2	49,734	49,734
Average Heat Flux, Btu/h-ft^2	185,090	185,090
Maximum Heat Flux, Btu/h-ft^2	554,200	576,835
Average Thermal Output, kW/ft	6.105	6.105
Maximum Design Thermal Output, kW/ft	18.28	19.03
Maximum Cladding Surface Temperature, °F	654	654
Average Core Fuel Temperature, °F	1200	1200
Maximum Fuel Temperature at Hot Spot, °F	4060	4170
Total Reactor Coolant Flow, 10^6 lb/h	131.32	137.8
Core Average Coolant Velocity, fps	15.74	16.52
DNB Ratio at Design Overpower	1.41	1.39
DNB Ratio at Design Power	1.79	1.75

5.0 REACTOR COOLANT SYSTEM

5.1 Summary Description

Davis-Besse 1 uses a B&W 2-loop nuclear steam supply system. In most important aspe ts, it is the same as the Rancho Seco system. The primary difference is the higher steam generators on Davis-Besse provided to decrease the potential for steam binding following a LOCA. On the basis of our evaluation of the Davis-Besse system and the similarity to the previously approved Rancho Seco, we conclude that the overall design of the reactor coolant system of Davis-Besse 1 is acceptable.

5.2.2 Overpressure Protection

Overpressure protection in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article 9 is provided by pressure relief of the RCS from two pressurizer code safety valves and one electrically actuated relief valve mounted on nozzles on the pressurizer. The valves discharge through manifolding to a pressurizer quench tank. The code safety valves are each rated to carry 336,000 lbm/hr at 2450 psig, which is the maximum calculated surge of the system based upon the worst pressure transient. The electromatic relief valve has a capacity of 100,000 lbm/hr at 2255 psig. The pressurizer safety valves are sized on the basis of the most severe pressure transient imposed on the RCS. The applicant's analyses of safety valve capacity (BAW-10043) show that the upsets that produce the largest pressure transients are the control rod withdrawal from low power and the turbine trip from the overpower condition. Although the staff review of BAW-10043 has not been completed, examination of the B&W analytical results shows that sufficient margins exist to conclude that the Davis-Besse 1 design is acceptable.

5.3 Thermal-Hydraulic System Design

The thermal and hydraulic design bases of the RCS are discussed in Section 4.4.

5.5 Component and Subsystem Design

5.5.1 Reactor Coolant Pumps and Motors

The Reactor Coolant Pump is designed to provide adequate core cooling flow and hence sufficient heat transfer to maintain a DNBR 1.30, within the parameters of operation.

Sufficient pump rotational inertia is provided by the flywheel to provide continued flow following a loss of pump power such that the reactor neutron power can be reduced before DNB limits are exceeded.

5.5.2 Steam Generator

The steam generator is a vertical straight-tube-and-shell heat exchanger and produces superheated steam at constant turbine throttle pressure over the operating power range. The primary reactor coolant enters the steam generator upper hemispherical head, flows downward inside the tubes giving up heat to generate steam on the shell side secondary loop.

The tube and tube-sheet boundary have the same design pressure and temperature as the reactor coolant system. Since the steam generators must provide a heat sink for the primary reactor coolant system, they are at a higher elevation than the core to assure natural circulation for decay heat removal.

5.5.3 Reactor Coolant Piping

The reactor coolant piping is designed and fabricated to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions.

5.5.4 Main Steamline Flow Restrictors

The applicant stated that because of the small inventory of water in the B&W OTSG design, no flow restrictors are required in the main steamline. This contention is supported by their analysis of postulated steam line breaks (FSAR Section 15.4.4).

5.5.7 Decay Heat Removal System

The Decay Heat Removal System is designed to remove decay heat and sensible heat from the RCS and core during the latter stages of cooldown. The system also provides auxiliary spray to the pressurizer for complete depressurization, maintains the reactor coolant temperature during refueling, and provides the means for filling and draining the refueling cavity. In the event of a LOCA, the decay heat removal pumps are used for low pressure injection of borated water into the reactor vessel for emergency core cooling.

The Decay Heat Removal System is placed into operation approximately 6 hours after initiation of plant shutdown when the temperature and pressure of the RCS are below 280°F and 260 psig, respectively. Assuming that two pumps and coolers are in service, and that each cooler is supplied with component cooling water at design flow and temperature, the DHRS is designed to reduce the RCS temperature to 140°F within 14 hours. If one of the two pumps or one of the two coolers is not operable, safe cooldown of the plant is not compromised; however, the time required for cooldown is extended. The applicant has shown that, assuming [illegible] one train is available, the plant can be shut down to below 212°F within 24 hours. To increase the reliability of the DHRS, the applicant has installed a manual bypass in the DHRS suction line.

The applicant has shown that, should motor-operated suction valves DH 11 or DH 12 be discovered failed closed at the time shutdown cooling was needed, the operator is able to enter the containment and open the manual bypass valves without exceeding dose limits. This radiological assessment is under review by the staff. In addition, the applicant is required to show that, should a spurious closure of DH 11 or DH 12 occur during RHR system operation, either sufficient time exists for the operator to detect the loss of flow and secure the low pressure pumps before overheating occurs, or the existing design is able to cope with such a loss of flow until the manual bypass has been opened.

The DHRS design for Davis-Besse 1 has double isolation valves on the suction side to isolate low pressure components from the reactor coolant system. The staff requires that the applicant ensure that the design features which protect the DHR system against overpressurization during shutdown (while the shutdown cooling system is functioning) are adequate. The applicant is required to provide analyses which justify the DHRS relief capacity. These analyses are to consider the occurrence of a worst-case pressure event under these shutdown conditions.

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5.5.10 Pressurizer

The pressurizer maintains the RCS pressure during steady state operation and limits pressure changes during transients. It contains a water volume, sized to provide the ability of the system to experience a reactor trip and not uncover the low level sensors in the bottom head and to maintain the pressure high enough so as not to activate the high pressure injection system; and a volume of steam, sized to provide the ability of the system to experience a turbine trip and not cover the level sensor in the upper head.

Electric heater bundles, located in the lower section, and a water spray nozzle in the upper section maintain the steam and water at the saturation temperature which corresponds to the desired reactor coolant system pressure. During outsurges, as the RCS pressure decreases, some of the water flashes to steam and the electric heaters restore the normal operating pressure. During insurges, as RCS pressure increases, the water spray condenses steam to reduce the pressure to the normal operating level. Two ASME code safety valves are connected to the upper pressurizer head to relieve system overpressure. A pilot-operated relief valve is also provided to limit the lifting frequency of the code safety valves. The safety and relief valves discharge to the pressurizer quench tank, located within containment.

5.5.13 Safety and Relief Valves

The pressurizer safety valves are bellows sealed, balanced, spring-loaded safety valves which are provided with a supplemental backpressure balancing piston for handling a bellows failure. The pressurizer relief valve is an electrically actuated, electrically controlled, pilot operated, pressure loaded, relief valve.

The combined capacity of the pressurizer safety valves is 672,000 lbm/hr, which was based on twice the maximum surge resulting from the upset that produces the largest pressure transient (see subsection 5.5.2). The maximum surge assumes no direct reactor trip, operator action or credit for actuation of the pressurizer relief valve or turbine bypass system. The pressurizer safety valves prevent the reactor coolant system pressure from exceeding 110% of system design pressure. The pressurizer power operated relief valve prevents undesirable lifting of the spring-loaded safety valves.

5.5.14 Internals Vent Valves

The core support vent valves are located on a common plane in the upper core support weldment above the outlet nozzles. These valves provide a direct flow path between the upper core region and inlet annulus in the event of a loss-of-coolant accident from an inlet line break. This flow path provides for pressure equalization by the venting of steam to the break and permits the emergency coolant water to reflood the core rapidly. There are four vent valves, each with an effective flow diameter of 14 inches. The face of the disc is slightly inclined to insure a positive seal of the disc due to the differential pressure across the valve. The individual vent valve design is essentially the same as on the Oconee Class plant.

In the thermal-hydraulic analysis of Davis-Besse 1 for normal operation, the applicant assumed that there was no core bypass flow resulting from an open vent valve. At present, there is not adequate instrumentation to detect the system flow change (approximately 5% reduction in core flow) which would result from an open valve. The staff position has not changed from that taken on the Oconee plant. Further, the applicant has not presented data from operating plants to show that a stuck open vent valve is an extremely low probability event. Therefore, we will require the thermal-hydraulic reanalysis described in Section 4.4.

We conclude, subject to the conditions as noted above, that the proposed reactor coolant system, subsystems, and component designs are acceptable.

5.5.15 Loose Parts Monitoring System

Occasionally, miscellaneous items such as nuts, bolts, and other small items have become loose parts within reactor coolant systems. In addition to causing operational inconvenience, such loose parts can damage other components within the system or be an indication of undue wear or vibration. For such reasons, the staff has encouraged applicants over the past several years to support programs designed to develop an effective, on-line loose parts monitoring system. For the past few years we have required many applicants to initiate a program, or to participate in an ongoing program, the objective of which was the development of a functional, loose parts monitoring system within a reasonable period of time. Recently, prototype loose parts monitoring systems have been developed and are presently in operation or being installed at several plants. Such a system has been installed on Davis-Besse 1 to provide the operator with an audible and visual alarm of loose parts which accumulate in the bottom of the reactor vessel and in the top of each steam generator. The staff will follow the performance of this on-line monitoring system.

6.3 Emergency Core Cooling System (ECCS)

6.3.1 Design Bases

Toledo Edison Company has stated that the Davis-Besse 1 Emergency Core Cooling System will be designed to provide core cooling during postulated accident conditions which occur when mechanical failure in the reactor coolant system piping results in a loss of coolant from the reactor vessel greater than the available coolant makeup capacity using normal operating equipment. The ECCS equipment is designed to provide both short and long-term core cooling capability.

The applicant's design bases are to ensure that the core will be cooled and will not lose its geometric configuration by terminating the temperature transient for any size break up to and including a double-ended rupture of the largest primary coolant line. The applicant states that these requirements will be met even with minimum engineered safeguards available, such as the loss of one emergency power bus, together with the loss of offsite power.

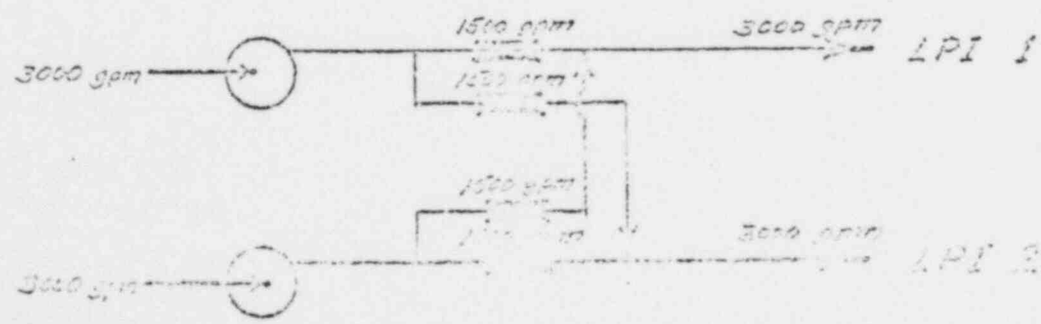
The ECCS to be provided is stated to be of such number, diversity, reliability and redundancy that no single failure of ECCS equipment occurring during a LOCA will result in inadequate cooling of the reactor core. Each of the ECCS subsystems are to be designed to function over a specific range of reactor coolant piping system break sizes, up to and including the flow area associated with a postulated double-ended break in the largest reactor coolant pipe (14.1 ft^2 is the largest double-ended area).

6.3.2 System Design

The ECCS proposed for Davis-Besse 1 consists of core flooding tanks (CFT), a high pressure injection (HPI) system and a low pressure injection (LPI) system. Provisions are included for recirculation of the borated coolant after the borated water storage tank (BWST) is exhausted. Combinations of these systems assure core cooling for the complete range of postulated break sizes.

Following a postulated LOCA, the ECCS will operate initially in the active high pressure injection mode, the passive injection mode, then in the active low pressure injection mode, and subsequently in the recirculation mode.

High pressure injection, upon actuation of an Engineered Safety Feature Actuation Signal (ESFAS), will consist of the operation of two centrifugal HPI pumps (rated at 500 gpm each at a design head of 2700 ft) which inject 1800 ppm concentrated boric acid solution into the reactor coolant system cold legs. These pumps take their suc ion from the borated water storage tank which has a volume of 550,000 gallons. Low pressure injection will be accomplished through two separate low paths, each having one decay heat removal pump and cooler. The low pressure injection lines terminate directly in the reactor vessel through the core flooding nozzles located in the reactor vessel. For short-term cooling, the low pressure injection via the decay heat removal pumps (rated at 3000 gpm each at a design head of 350 feet) provides 1800 ppm boron solution from the borated water storage tank. A crossover line connecting the two LPI lines within the reactor building is provided so that if a single failure causes the loss of LPI flow in one path, part of the flow from the active LPI path can be injected into the reactor vessel through the piping associated with the inactive path. The cross-connect is also intended to assure abundant long-term cooling flow to the core in the event of a core flooding line break in addition to this single active failure. We have required that this cross-connect be modified to incorporate a passive network design similar to that adopted on such plants as Arkansas Nuclear One Unit 1 and North Anna 3 and 4 (plus all 205-Fuel Assembly plants). This preferred method consists of crossover lines which contain no motor-operated valves. Instead, this crossover network utilizes the flow-limiting characteristic of a cavitating venturi to provide an automatic split in ECC water between the two LPI trains. The following simplified diagram illustrates this principle:



The obvious advantage of the latter method is that it does not rely on operator action to be initiated and it is less prone to active component failures.

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The ECCS will provide the long-term core cooling requirements by recirculating the spilled reactor coolant collected in the containment sump back to the reactor vessel through the core flooding line nozzles. The changeover from low pressure injection to recirculation is accomplished manually from the control room with automatic backup to the manual action.

For large sized pipe ruptures, the ECCS will provide the long-term cooling requirements by recirculating the spilled reactor coolant collected in the containment sump, back to the reactor vessel via either of the available two trains of low pressure pumps and coolers. Prior to this time, the operator is required to shut off the HPI pumps to avoid their overheating when the BWST valves are closed.

For small sized pipe ruptures, the reactor coolant system pressure may be higher than the maximum low pressure injection pump head at the time containment sump water recirculation is required. Under this circumstance, a cross-over connection is provided to permit alignment of the high pressure make-up pump suction with the low pressure injection cooler discharge to permit high pressure injection during the recirculation mode of operation. Presently, this alignment is accomplished by the operator manually opening one valve in each of the two crossover pipe lines located in the auxiliary building. The staff requires that these valves be motor operated with control and indication in the control room.

The passive injection mode of operation is provided by the core flooding (CF) system, which protects the core in the event of intermediate and large-sized pipe breaks. The coolant is automatically injected when the RCS pressure drops below the core flooding tank pressure (600 psig).

Each of the two core flooding tanks has a total volume of 1410 ft^3 with a normal water volume of 1040 ft^3 with 370 ft^3 of nitrogen gas at a normal operating pressure of 600 psig. Each tank is connected by a core flooding line directly to a reactor vessel core flooding nozzle. The driving force for injection of the 1800 ppm borated water is supplied by pressurized nitrogen. Each core flooding line will contain a motor-operated stop valve for isolation of the CFT during reduced pressure operation and two inline check valves in series. Since this portion of the ECCS involves a high pressure to low pressure interface, it is the staff's position that periodic checking of potential leakage through check valves CF 30/31 and DH 76/77 is to be performed at least annually. This test is to be performed at or near normal reactor coolant operating pressure. The current design has a continuous monitor outboard of these two check valves, however, this location is not reliable in detecting the prelude to a pressure barrier failure (i.e., leakage of the inboard check valve).

To minimize the potential for a water hammer occurring due to ECC water being discharged into a dry line, the applicant has stated that during normal operation, the ECCS lines will be maintained full by the static head created by the relative elevations of the BWST and ECCS piping. In addition, manual venting is provided at the ECCS pump casings and discharge piping high points. The staff requires that the capability to maintain filled ECCS piping be observed prior to startup and that the venting provisions constitute a periodic surveillance requirement in the Davis-Besse 1 Technical Specifications. Specifically, the section of HPI piping inboard of normally closed isolation valves HP2A, HP2B, HP2C and HP2D must be observed to be full since during normal operation the static head of the BWST would be terminated at these valves.

6.3.3 Performance Evaluation

Toledo Edison Company has stated that the emergency core cooling systems have been designed to deliver fluid to the reactor coolant system to control the predicted cladding temperature transient following a postulated pipe break and for removing decay heat in the long-term, recirculation mode. On January 4, 1974, Acceptance Criteria for ECCS was published in 10 CFR Part 50. The new ECCS criteria requires that:

(1) The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

(2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry shall be such that the core remains amendable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for an extended period of time required by the long-lived radioactivity remaining in the core.

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The applicant submitted an analysis of ECCS performance [illegible] on July 21, 1975 (Reference 1) by reference to Topical report BAW-[illegible]. In addition to the revised LOCA analysis, the staff's review of the ECCS for Davis Besse 1 requested additional information in the [illegible] of minimum containment pressure, single failure criterion, effects of boron precipitation on long term cooling capability, and submerged valves within containment (Reference 2). The adequacy of ECCS performance and the staff's evaluation of the applicant's evaluation model will be reported in a supplement to this Safety Evaluation Report.

6.3.4 Tests and Inspections

Toledo Edison Company will demonstrate the operability of the ECCS by subjecting all components to preoperational tests, periodic testing, and in-service testing and inspections. The preoperational tests performed fall into three categories. One of these categories consists of system actuation tests to verify the operability of all ECCS valves initiated by Engineered Safety Feature Actuation Signal (ESFAS), the operability of all safeguard pump circuitry down through the pump breaker control circuits and the proper operation of all valve interlocks.

Another category is the core flooding tank tests. The objective of this test is to check the core flooding system and injection line to verify that the lines are free of obstructions and that the core flooding line check valves and isolation valves operate correctly. The applicant will perform a low pressure blowdown of each core flooding tank to confirm the line is clear and check the operation of the check valves.

Operational test of all the major pumps comprises the last category of tests. These pumps consist of the high pressure injection pumps and the low pressure/decay heat removal pumps. The applicant will use the results of these tests to evaluate the hydraulic and mechanical performance of these pumps delivering through the flow paths for emergency core cooling. These pumps will operate under both miniflow (through test lines) and full flow (through the actual piping) conditions.

By measuring the flow in each pipe, the applicant will make the adjustments necessary to assure that no one branch has an unacceptably low or high resistance. They will also check the system to assure there is sufficient total line resistance to prevent excessive runout of the pump. The applicant must show that the minimum acceptable flows as determined for the FSAR analysis are met by the measured total pump flow and relative flow between the branch lines. In addition, preoperational flow tests must be conducted to verify the sizing of the required cavitating venturies to confirm the as-built flow split performance of the LPI system. The system will be accepted only after demonstration of proper actuation of all components and after demonstration of flow delivery of all components within design requirements.

The applicant will perform routine periodic testing of the ECCS components and all necessary support systems at power. Valves which operate after a loss-of-coolant accident are operated through a complete cycle, and pumps are operated individually in this test. The staff requires that the applicant demonstrate the capability of each motor-operated ECCS valve to open and close using the local handwheel backup. In addition, in response to a request from the staff, the applicant has evaluated his proposed compliance with the positions stated in Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors." With the exception of the recirculation test under ambient conditions, the applicant has indicated that he will comply with Regulatory Guide 1.79. It is our position that a test must be conducted to demonstrate (at ambient conditions) the capability of the ECCS to operate in the recirculation mode. To avoid reactor coolant system contamination, the sump water may be discharged to external drains or other systems. Temporary arrangements may be made to provide adequate sump capacity for pump operation. The specific purpose of this test is to demonstrate that conditions (such as inadequate NPSH, air binding or vortex formation at the sump screens), which could adversely affect ECCS performance, do not occur.

6.3.5 Conclusions

On the basis of our evaluation, we have concluded that the ECCS for Davis-Besse 1 plant is acceptable in regard to a decision concerning issuance of an operating license with the following exceptions:

1. If a small break occurred such that the high pressure injection (HPI) system alone could replenish the leaking reactor coolant, the Low Pressure Injection (LPI) system would be required some time after the accident to provide a water supply from the containment sump. The proposed design requires that the operator [illegible]

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in the crossover lines between the HPI pump suction and the LPI pump discharge lines. Operator actions required to mitigate the consequences should be minimized and ECCS component reliability kept at a high level. Therefore, the normally closed valves in each of the two crossover lines should be remote motor-operated valves with position indication and controls in the control room.

2. For a break in a core flooding line, a single active component failure could degrade available ECCS to the point of compromising the abundant core cooling requirement of General Design Criterion 35. To meet this criterion, Toledo Edison Company has installed a crossover line between each LPI train which is manually actuated from the control room. To further minimize operator actions and maximize ECCS component reliability, we will require a passive crossover network between the two LPI trains.
3. The adequacy of ECCS performance and the staff's evaluation of the applicant's evaluation model will be reported in a supplement to this Safety Evaluation Report.
4. The applicant will be required to demonstrate the capability of the ECCS to operate in the recirculation mode of ECCS operation. The applicant must also demonstrate the operability of the local manual handwheel backup on each ECCS valve prior to power operation.

15.0 Accident Analysis

15.1 General

The submitted safety analysis evaluates the ability of Davis-Besse 1 to operate without undue hazard to the health and safety of the public. Two basic groups of events pertinent to safety are investigated by the applicant; abnormal transients and postulated accidents.

15.2 Abnormal Transients

The criterion, adopted to assure that the reactor coolant pressure boundary integrity is maintained, is that the system pressure shall remain below the code pressure limits set forth in ASME Code Section III (110% of RCS design pressure). The criterion adopted to ensure that no fuel damage has occurred is that the DNBR must be greater than 1.30 throughout the transient.

The applicant has submitted analyses of abnormal transients and has shown that the integrity of the RCS boundary has been maintained and that the minimum DNBR exceeded 1.30 for all analyzed transients. The pressure transient which produced the highest reactor coolant system pressure was identified (BAW-10043) as the control rod withdrawal at low power conditions, resulting in a peak RCS pressure of about 2665 psig. The most severe secondary side pressure transient was the turbine trip from overpower conditions, resulting in a maximum steam generator pressure of about 1140 psig. The minimum hot channel DNBR of 1.44 was identified for the excessive heat removal transient resulting from a feedwater system malfunction.

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The applicant has referenced BAW-10099 (Reference 3) as their position regarding design features to make tolerable the consequences of failure to scram during anticipated transients. We are continuing our generic review of this area of concern and the staff evaluation of the Babcock and Wilcox analyses will be submitted in a supplement to this Safety Evaluation Report.

The computer codes "POWER TRAIN" and "PUMP" used for several abnormal transients in the FSAR, are currently under review by the staff. Should modifications to these codes be required, the effect of these changes on the Davis-Besse 1 analyses must be considered.

The evaluation of abnormal transients indicated that the transients presented do not lead to unacceptable consequences and are acceptable for issuance of the operating license.

15.3 Accidents

The applicant has evaluated a broad spectrum of accidents that might result from postulated failures of equipment, or their maloperation. These highly unlikely accidents, which are representative of the spectrum of types and physical locations involving the various engineered safety feature systems, have been analyzed in detail.

The accidents reviewed in the SAR include the following:

1. Loss of forced reactor coolant flow resulting from a single reactor coolant pump locked rotor.
2. Main steamline rupture
3. [illegible]

The locked rotor accident was analyzed by postulating an instantaneous seizure of one RC pump rotor. The reactor flow would decrease rapidly and a reactor trip would occur as a result of a high power-to-flow signal.

The analysis revealed that at no time during the transient did the DNBR go below 1.0. The applicant concluded that no severe fuel rod or cladding temperature excursions are expected to occur as a result of this accident.

The loss-of-secondary-coolant (steam line rupture) analyses has been performed to determine the effects and consequences due to a double-ended steam line rupture. A 36-inch OD steam line rupture, between the steam generator and the main steam isolation valve was analyzed assuming that the reactor was operating at 102% design power prior to the accident.

The present design has only one main steam isolation valve in each of the steam lines to isolate the unaffected steam generator and to prevent it from blowing dry in the event of a steam line rupture. The applicant's evaluation shows that with a single failure of the isolation valve in the unaffected steam line, turbine stop valves will serve as backup to the first-line isolation safeguard. The staff requires that the isolation capability of the non-safety grade turbine stop valves be closure-tested periodically and that this test be made a part of the plant Technical Specifications. The staff notes that the worst-case steam line break with regard to reactivity margin was not represented in the FSAR (loss of offsite power not assumed); however, the applicant has shown that sufficent safety margin exists to justify the differences as not being significant. The applicant's evaluation also shows that with single failure of a feedwater stop valve, the closure of the feedwater control valve and parallel feedwater startup valve will serve as backup to the front-line feedwater isolation safeguard. The staff requires that the isolation capability of the non-safety grade feedwater control valves and startup valves also be periodically closure-tested and that this test be made a part of the plant Technical Specifications. The staff also requires that the feedwater stop valve closure time assumed for the main steam line break (17 seconds) be made a part of the applicant's Technical Specifications and that this time be the basis for all safety analyses requiring feedwater isolation. It is also noted that consideration of additional single active component failures was not complete. The scope of potential single failures should include the inadvertant opening of the atmosphere vent valves or turbine bypass system. It must be confirmed that these components would provide the largest additional cooldown rate by an examination of all steam line and steam generator active component appurtenances. After isolation of the main steam line break, credit was taken for the additional relieving capacity offered by the atmosphere vent valves. Since pressure margin could decrease in the unaffected steam generator if credit were only given for the steam line safety valves (higher setpoint , the staff requires that this event be reanalyzed with a single failure of the atmosphere valve on the isolated steam generator (failure to open).

All the preceding comments on the main steam line break also apply to the feedwater line break. The adequacy of these re-analyses will be reported in a supplement to this Safety Evaluation Report.

References

(1) Letter from Lowell E. Roe to A. Schwencer dated July 21, 1975

(2) Letter from Lowell E. Roe to Mr. A. Schwencer dated July 9, 1975.

(3) BAW-10099, "B&W Anticipated Transients Without Scram Analysis," December 1974.