SURVEILLANCE REQUIREMENTS (Continued)

c. The Auxiliary Feed Pump Turbine Steam Generator Level Control System shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days, and a CHANNEL CALIBRATION at least once per 18 months.

d. The ~~Auxiliary Feed Pump Turbine Speed Switch,~~ Auxiliary Feed Pump Suction Pressure Interlocks, and Auxiliary Feed Pump Turbine Inlet Steam Pressure Interlocks shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days, and a CHANNEL CALIBRATION at least once per 18 months.

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Docket No. 50-346

License No. NPF-3

Serial No. 731

July 10, 1981

Attachment 2

I. Change to Davis-Besse Nuclear Power Station Unit 1, Appendix A Technical Specifications 3.4.2, 3.4.3, 4.4.3 and Bases concerning the setpoint index for pressurizer electromatic relief and code safety valve.

   A. Time required to Implement
      This change is to be effective upon NRC approval

   B. Reason for Change (Facility Change Request 79-348)
      To reduce the probability of opening pressurizer electromatic relief and code safety valve during a transient

   C. Safety Evaluation
      See attached

Safety Evaluation

For a RPS high pressure trip setpoint of 2300 psig, the maximum overshoot of the Reactor Coolant System pressure for a loss of feedwater (LOFW) event would be to 2350 psig. Also, the LOFW is the maximum over-Pressure anticipated transient. The string inaccuracies and drift for the RPS high pressure trip are 15.29 psi, or 16 psi conservatively.

The inaccuracies and drift for the string that controls the electromatic relief valve for the pressurizer are 16.75 psi, or 17 psi conservatively. Included in this value is an inaccuracy of 4 psi and a drift of 7.5 psi for the transmitter. The 4 psi and 7.5 psi were combined by taking the square root of the sum of the squares, giving 8.5 psi. Subtracting 4 psi from 8.5 psi gives a value of 4.5 psi that is attributable to only the drift. The 8.5 psi was then added to inaccuracy and drift values for other components in the string to obtain a total of 16.75 psi.

The allowable value of ≥2431 psig is obtained by subtracting 4.5 psi due to the drift from the trip setpoint of ≥2435.5 psig. The minimum lift pressure for the pressurizer electromatic relief valve is then (2435.5-17) psig = 2418.5 psig. Consequently, the resultant margin between the maximum pressure peak of 2366 psig and minimum lift pressure of 2418.5 psig for the pressurizer electromatic relief valve following an anticipated transient is 52.5 psig.

The above values for the pressurizer electromatic relief valve in conjunction with a 2300 psig RPS high pressure trip setpoint will avoid actuation of the pressurizer electromatic relief valve during anticipated transients. All safety analyses for Davis-Besse Unit 1 assume that the vent capacity of the pressurizer electromatic relief valve will not be available; thus, these analyses are unchanged by an increase in its setpoint.

The pressurized code safety valves must be set such that the peak reactor coolant system pressure does not exceed 110% of design system pressure or 2750 psig. The peak reactor coolant system pressure was determined from the control rod group withdrawal accident from low power at beginning-of-life conditions. The analysis was done for a high pressure trip of 2300 psig. In the analysis, 30 psid was used to account for the instrument string inaccuracy and draft. This is consistent with the value in FSAR. However, the 30 psid is conservative, because the actual, as built instrumentation has 15.3 psid inaccuracy and drift. The pressurizer electromatic relief valve was assumed not to open. In the analysis, the code safety valve set pressure plus 3% was used, based on subsection 7614.1 of the ASME code. Also, the analysis was based on a maximum reactivity insertion rate of $1.655 \times 10^{-4} \Delta k/k/\text{sec}$. The peak reactor coolant system pressure calculated is 2716 psig. This is acceptable, according to the above criteria.

Teledyne Engineering Services reviewed the setpoint changes and have concluded that the increase in setpressures will have no significant effect on the imposed limit on the number of discharge events of these valves.

The changes do not constitute an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, previously evaluated in FSAR, has not been increased.

2. The possibility of an accident or malfunction of a different type other than any evaluated previously in the FSAR has not been created.

3. The margin of safety as defined in the basis for any technical specification has not been reduced.

pp b/7-9

UNDER NRC REVIEW

ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
Serial No. 669 Date 12/26/81

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 PSIG ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE DHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

SAFETY VALVES ~~OPERATING~~ AND ELECTROMATIC RELIEF VALVE - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 PSIG ± 1%.* When not isolated, the pressurizer electromatic relief valve shall have a trip setpoint of ≥ 2435.5 PSIG and an allowable value of ≥ 2431 PSIG.**

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 For the pressurizer code safety valves, there are no ~~No~~ additional Surveillance Requirements other than those required by Specification 4.0.5. For the pressurizer electromatic relief valve a channel calibration check shall be performed every 18 months.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** Allowable value for channel calibration check.

For a RPS high pressure trip setpoint of 2300 psig, the maximum overshoot of the Reactor Coolant System pressure for a loss of feedwater (LOFW) event would be to 2350 psig. Also, the LOFW is the maximum over-pressure anticipated transient. The string inaccuracies and drift for the RPS high pressure trip are 15.29 psi, or 16 psi conservatively. The maximum pressure peak for an anticipated transient is then 2366 psig.

The inaccuracies and drift for the string that controls the electromatic relief valve for the pressurizer are 16.75 psi, or 17 psi conservatively. Included in this value is an inaccuracy of 4 psi and a drift of 7.5 psi for the transmitter. The 4 psi and 7.5 psi were combined by taking the square root of the sum of the squares, giving 8.5 psi. Subtracting 4 psi from 8.5 psi gives a value of 4.5 psi that is attributable to only the drift. The 8.5 psi was then added to inaccuracy and drift values for other components in the string to obtain a total of 16.75 psi.

The allowable value of ≥2431 psig is obtained by subtracting 4.5 psi due to the drift from the trip setpoint of ≥2435.5 psig. The minimum lift pressure for the pressurizer electromatic relief valve is then (2435.5-17) psig = 2418.5 psig. Consequently, the resultant margin between the maximum pressure peak of 2366 psig and minimum lift pressure of 2418.5 psig for the pressurizer electromatic relief valve following an anticipated transient is 52.5 psi.

Thus, a 2300 psig RPS high pressure trip setpoint and the above values for the pressurizer electromatic relief valve will avoid actuation of the pressurizer electromatic relief valve during anticipated transients.

Docket No. 50-346

License No. NPF-3

Serial No. 731

July 10, 1981

Attachment 3

I. Change to Davis-Besse Nuclear Power Station Unit 1, Appendix A Technical Specifications 4.8.1.2.2.C.2 and 4.8.1.2 concerning Emergency Diesel Generator (EDG) surveillance requirements.

   A. Time required to Implement
      This change is to be effective upon NRC approval

   B. Reason for Change (Facility Change Request 80-270A)
      The Technical Specification changes for the EDG are to correct typographical errors in Section 4.8.1.2 per request in NRC Inspection Report 50-346/80-29 dated December 23, 1080 (Log No. 1-456). Section 4.8.1.1.2.C.2 requires the EDG to demonstrate the capability to reject a load of $\leq$480 KW without tripping. Instead the load shedding capability of the largest single emergency load should be tested.

   C. Safety Evaluation
      See attached

Safety Evaluation

The first change is to correct a typographical error in Tech. Spec. 4.8.1.2. Surveillance Requirement 4.8.1.2, applicable in modes 5 and 6 currently require the Emergency Diesel Generator to be demonstrated as OPERABLE by the performance of each of the Surveillance Requirement of 4.8.1.2.2 except for 4.8.1.2.2.a.5. The reference to 4.8.1.2.2.a.5 as an exception in not correct. The correct exception to be referred is 4.8.1.1.2.a.7. Surveillance Requirement 4.8.1.1.2.a.5 concerns the synchronization and loading of the Emergency Diesel Generator and operation for 60 minutes. This requirement should be performed, not exempted because one Emergency Diesel Generator has to be OPERABLE in modes 5 and 6. Surveillance Requirement 4.8.1.1.2.a.7 concerns the verification of the operability of the Safety Features Actuation System (SFAS) automatic load sequence timer. This is the requirement which should be exempted in modes 5 and 6 as the SFAS instrumentation Tech. Spec's. 3/4.3.2, Table 3.3-2, item 4 is only applicable in modes 1, 2, 3 and 4.

The second change is to correct an error in Surveillance Requirement 4.8.1.1.2.C.2 which requires the EDG to demonstrate the capability to reject a load $\leq$480 KW without tripping. Instead the load shedding capability of the Emergency Diesel Generator to reject the largest single emergency load connected to it should be tested.

Correction of these errors does not create a safety question, therefore, this is not an unreviewed safety related issue.

pp b/10-11

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system consisting of;

1. One OPERABLE 345 KV transmission line,

2. One OPERABLE 345 KV - 13.8 KV startup transformer, and

3. One OPERABLE 13.8 KV bus, and

b. One diesel generator with:

1. Day fuel tank containing a minimum volume of 4000 gallons of fuel,

2. A fuel storage system containing a minimum volume of 32,000 gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2.a.~~8~~.
7

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day fuel tank,

2. Verifying the fuel level in the fuel storage tank,

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank,

4. Verifying the diesel starts from ambient condition and accelerates to at least 900 rpm in $\leq$ 10 seconds,

5. Verifying the generator is synchronized, loaded to $\geq$ 1000 kw, and operates for $\geq$ 60 minutes, and

6. Verifying the diesel generator is aligned to provide standby power to the associated essential busses.

7. Verifying that the automatic load sequence timer is OPERABLE with each load sequence time within $\pm$ 10% of its required value.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water and sediment.

c. At least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

See Attachment

2. Verifying the generator capability to reject a load of $\leq$ 480 kw without tripping,

3. Simulating a loss of offsite power in conjunction with a safety injection actuation test signal, and:

a) Verifying de-energization of the essential busses and load shedding from the essential busses,

PROPOSED REVISED TECH. SPEC.

T.S. 4.8.1.1.2.C.2

VERIFYING THE EMERGENCY DIESEL GENERATOR'S CAPABILITY TO REJECT A LOAD EQUAL TO THE LARGEST SINGLE EMERGENCY LOAD SUPPLIED BY THIS GENERATOR WITHOUT TRIPPING.

Docket No. 50-346

License No. NPF-3

Serial No. 731

July 10, 1981

Attachment 4

I. Change to Davis-Besse Nuclear Power Station Unit 1, Appendix A Technical Specifications 3.1.3.6, 3.2.1, 3.2.5 and Figure 3.1-3a concerning Regulating Rod insertion limits, Axial Power Imbalance and DNB Parameters.

   A. Time required to Implement
   This change is to be effective upon NRC approval

   B. Reason for Change (Facility Change Request 79-088B)
   To correct administrative errors in Technical Specifications and revised safety evaluation for Amendment request submitted February 11, 1980 (Serial 590).

   C. Safety Evaluation
   See attached

Safety Evaluation

This license amendment request proposes changes to the Davis-Besse Unit 1 Technical Specifications as amended by License Amendment No. 33 dated October 1, 1980. Similar changes to Technical Specifications 3.2.1 and 3.2.5 had been previously submitted to the NRC as part of the reload report submittal for Davis-Besse Unit 1, Cycle 2; but were not granted for lack of bases. This safety evaluation provides the bases for the same.

The safety function achieved by Technical Specification 3.2.1 is to ensure that axial power imbalance is maintained within limits when operating above 40% power. Action Statement b of this Technical Specification presently requires that the unit be brought to hot standby mode if the axial power imbalance exceeds the allowable value and is not restored within limits within 15 minutes. The requirement of bringing the unit to hot standby reduces unit availability and is excessive because the limiting condition of operation is not applicable below 40% of rated thermal power. The proposed change deletes the shutdown requirement. With the proposed change, thermal power is required to be reduced to less than 40% of rated thermal power or until imbalance limits are met. Due to the nature of the axial power imbalance limit envelopes, a reduction in power may bring the imbalance within the limit because the limits are wider at lower power levels. If the power reduction does not bring the axial power imbalance within the limits, further power reduction to less than 40% rated thermal power is required. The limiting condition for operation is then no longer applicable. If the Action Statements (a and b) are not satisfied, provisions of Specification 3.0.3 are applicable requiring that unit be taken to hot standby mode in one hour. The proposed change offers clarity and flexibility to the operator in the event that axial power imbalance limits are exceeded. The proposed change also reduces the requirements of rapid shutdown of the unit which, in turn, reduces adverse impact on components and equipment. The axial power imbalance can be readily restored within limits or the thermal power reduced within the specified time. The possibility of a rapid shutdown in this case is therefore eliminated.

As noted above, this license amendment request also modifies the action statement for Technical Specification 3.2.5. This specification sets limits on the DNB related parameters of reactor coolant pressure, flow and hot leg temperature. According to the present Technical Specifications, if any of these parameters falls outside the prescribed limits, the parameter is required to be brought within limits within two hours or thermal power reduced to less than 5% of rated thermal power within the next four hours.

The safety function achieved by this Technical Specification is to ensure that DNB limits are not violated. The present Technical Specification action statement requirement is too restrictive for reactor coolant flow. The measured system flow at Davis-Besse is 111.4% of

88,000 gpm/reactor coolant pump (after taking into account the 2.5% measurement uncertainty) whereas Technical Specification 3.2.5 calls for 110% flow. This license amendment request proposes that for every 1% flow that is below the limit, thermal power be reduced by 2% of rated thermal power. B&W has performed calculations to determine the DNBR margin gain for the proposed flow and power tradeoff. The CHATA and TEMP codes were used to determine hot bundle flow, and subchannel flow and minimum DNBR, respectively. A factor N was defined as the percentage reduction in thermal power. The design overpower case of 112% of 2772 MWt and 110% of design reactor coolant system flow was used as the basis for comparison. An increase in DNBR margin was observed when power was decreased per the proposed change when flow was off-limit. Figure 1 provides a graph of this increase in DNB margin. B&W has also concluded that this analysis is bounding for the case of three reactor coolant pumps operating and adequate DNB margin will also be gained in that case if the modified action statement is followed. Based on the above, it is concluded that by following the revised action statement, adequate margin to DNB is gained thereby enhancing the safety function achieved by this Technical Specification. In addition, it further reduces the possibility of an undesired unit shutdown and increases unit availability.

Two other changes proposed by this license amendment request are administrative in nature and relate to Technical Specification 3.1.3.6. Namely, in the note for Technical Specification 3.1.3.6, an improper section of Technical Specifications is referenced. The correct section to be referenced is 3/4.1.1.1. The correct reference was originally submitted to the NRC with the reload report submittal, but was apparently inadvertently changed when the license amendment was granted. Similarly, Figure 3.1-3a (regulating rod group position limits for three pump operation) is for 0 to 150 ± 10 EFPDs. The present figure in the Technical Specification states that the figure is valid for 0-125 ± 10 EFPDs. This is contrary to the reload report submittal to the NRC and needs revised as attached. Since these two changes are administrative, no safety concern is involved.

Based on the above, it is concluded that the changes to the Technical Specifications proposed by this license amendment request do not involve an unreviewed safety question.

pp b/12-14

REGULATING ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating rod groups shall be limited in physical insertion as shown on Figures 3.1-2a and -2b and 3.1-3a and -3b, with a rod group overlap of 25 $\pm$ 5% between sequential withdrawn groups 5, 6, and 7.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the regulating rod groups inserted beyond the above insertion limits (in a region other than acceptable operation), or with any group sequence or overlap outside the specified limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the regulating groups to within the limits within 2 hours, or

b. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the rod group position using the above figures within 2 hours, or

c. Be in at least HOT STANDBY within 6 hours.

NOTE: If in unacceptable region, also see Section 3/4. 1.1.1.

*See Special Test Exceptions 3.10.1 and 3.10.2.

#With $K_{eff} \geq 1.0$.

AXIAL POWER IMBALANCE

LIMITING CONDITION FOR OPERATION

3.2.1 AXIAL POWER IMBALANCE shall be maintained within the limits shown on Figures 3.2-1 and 3.2-2.

APPLICABILITY: MODE 1 above 40% of RATED THERMAL POWER.*

ACTION:

With AXIAL POWER IMBALANCE exceeding the limits specified above, either:

a Restore the AXIAL POWER IMBALANCE to within its limits within 15 minutes, or

b. Reduce power to less than 40% of RATED THERMAL POWER or until imbalance limits are met within one hour.

SURVEILLANCE REQUIREMENTS

4.2.1 The AXIAL POWER IMBALANCE shall be determined to be within limits at least once every 12 hours when above 40% of RATED THERMAL POWER except when the AXIAL POWER IMBALANCE alarm is inoperable, then calculate the AXIAL POWER IMBALANCE at least once per hour.

*See Special Test Exception 3.10.1.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor coolant hot leg temperature.

b. Reactor coolant pressure.

c. Reactor coolant flow rate.

APPLICABILITY: MODE 1.

ACTION:

If parameter a or b above exceeds its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

If parameter c exceeds its limit, either:
1. Restore the parameter to within its limit within 2 hours, or
2. Limit THERMAL POWER at least 2% below RATED THERMAL POWER for each 1% parameter c is outside its limit for four pump operation within the next 4 hours, or limit THERMAL POWER at least 2% below 75% of RATED THERMAL POWER for each 1% parameter c is outside its limit for 3 pump operation within the next 4 hours.

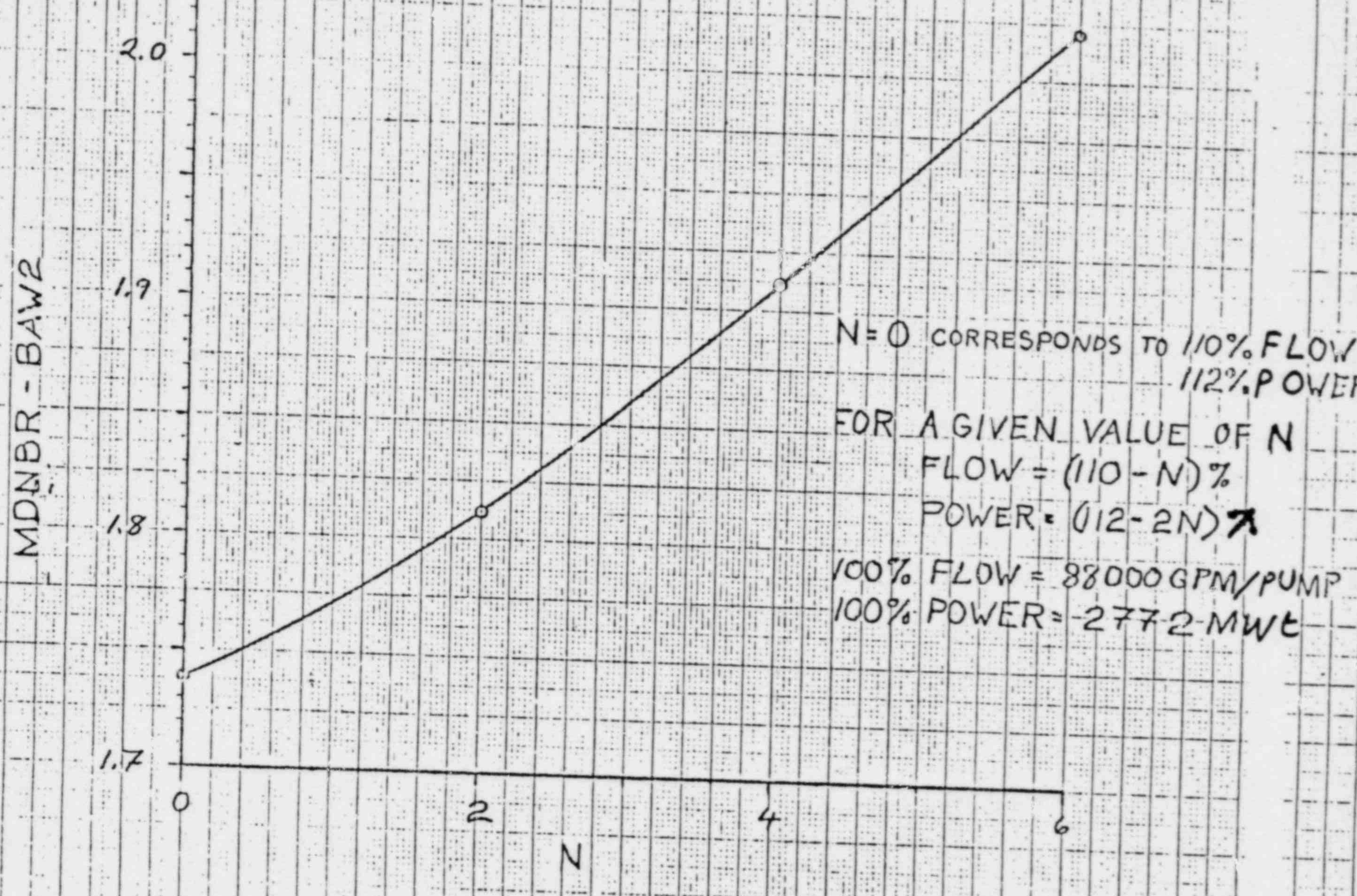
SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

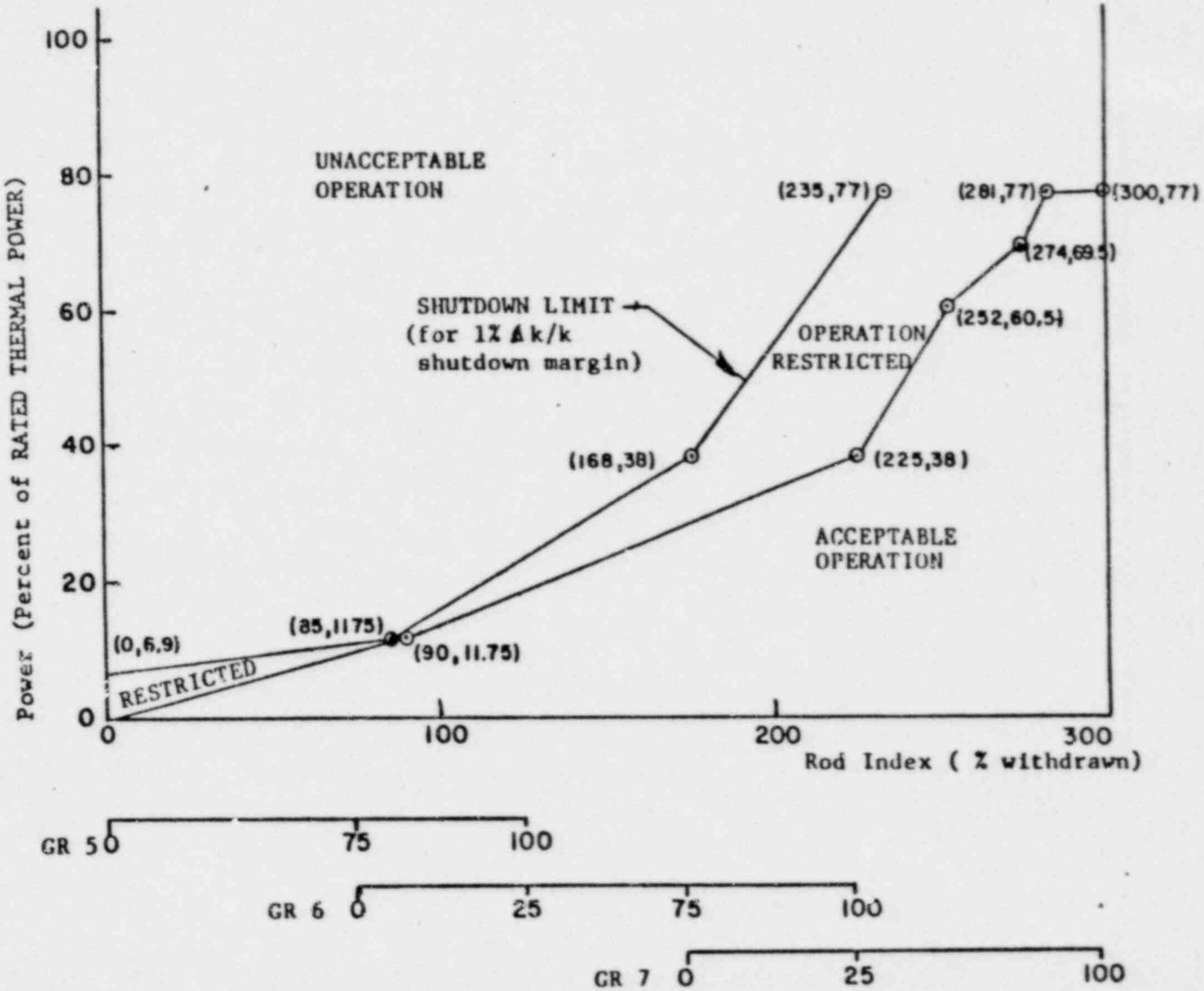
4.2.5.2 The reactor coolant system total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

FIGURE 1.

MDNBR vs FLOW AND POWER PARAMETER N



JJW 1/17/80



TECHNICAL SPECIFICATION FIGURE 3.1-3a Regulating Group Position Limits, 0 to ~~125 ± 10~~ 150±10 EFPD, Three RCPs - Davis Besse 1, Cycle 2

Docket No. 50-346

License No. NPF-3

Serial No. 731

July 10, 1981

Attachment 5

I. Change to Davis-Besse Nuclear Power Staiton Unit 1, Appendix A Technical Specifications 6.5.2.2, Figures 6.2-1 and 6.2-2 concerning CNRB membership and organizational changes.

   A. Time required to Implement
      This change is to be effective upon NRC approval

   B. Reason for Change (Facility Change Request 80-249C, D) To reflect organizational changes at Toledo Edison

   C. Safety Evaluation
      See attached

Safety Evaluation

The Technical Specification change requested reflects organizational changes. The safety function of the CNRB is to perform:

A. Review per Section 6.5.2.7 of Technical Specifications

   1. Review Safety Evaluations.
   2. Unreviewed Safety Questions on procedures, equipment, systems, test or experiments.
   3. Changes to Facility Operating License.
   4. Violations having nuclear safety significance.
   5. Abnormalities or deviations that affect nuclear safety.
   6. Events requiring 24 hours notification.
   7. Unanticipated deficiency in safety related structures, systems or components.
   8. SRB Minutes.

B. Audits of facility activity per Section 6.5.2.8 of Technical Specifications will remain unchanged.

The organizational charges to the CNRB will only change some members and not the function of the board. The changes improve independence and expertise on the board in facilitating their function.

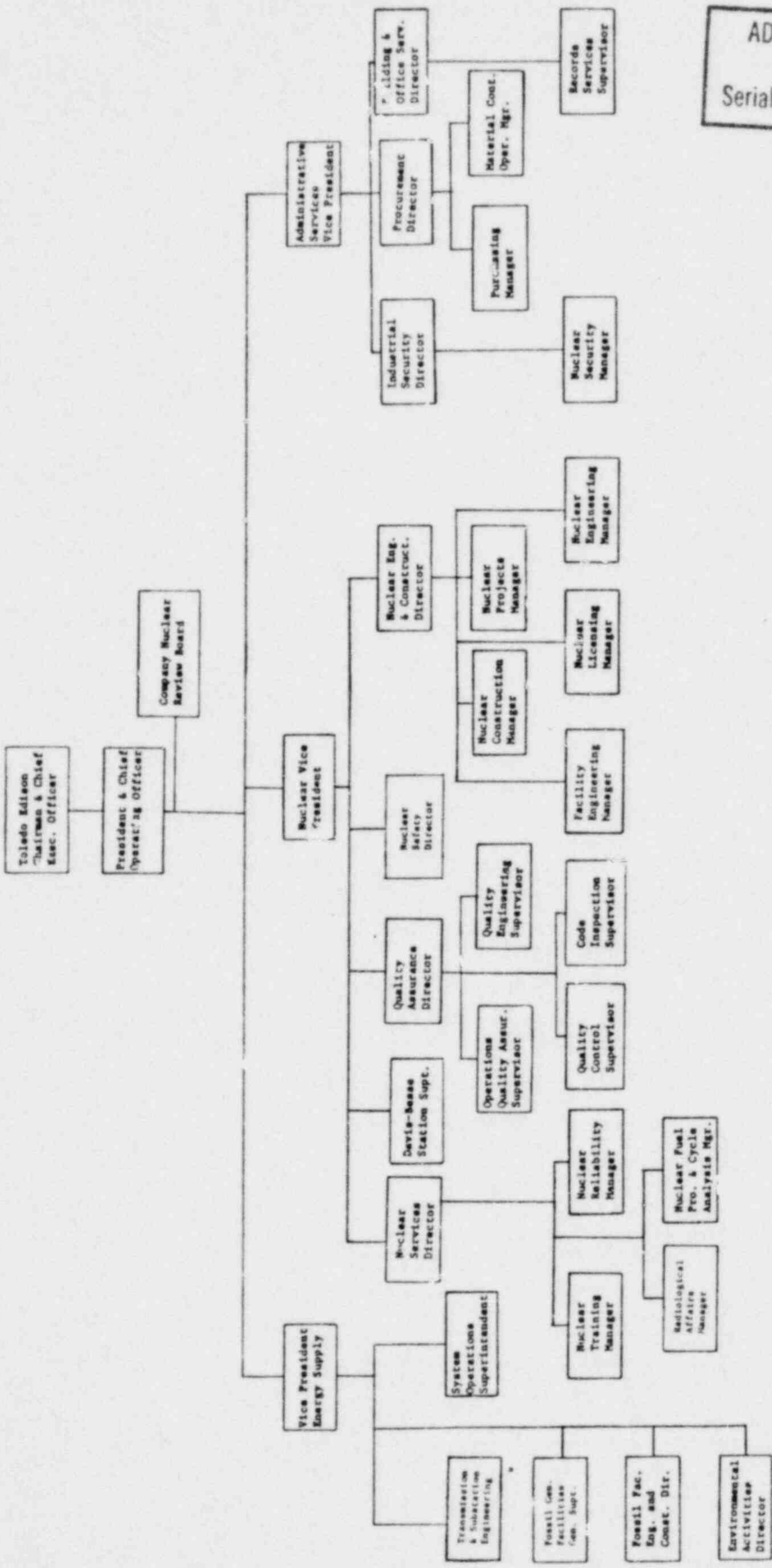
The safety function of the station organization is to show lines of responsibility for overall facility operation and maintenance of the station in a safe, reliable and efficient manner. The operating license requirement has been deleted from the Assistant Shift Supervisor. This position is an administrative function to aid the Shift Supervisor. The number of licenses in the crew composition is governed under Section 6, Table 6.2-1. Any license required functions by the control room organization shall be dispatched in accordance with 10 CFR 50.54 and Part 55.

There are no physical or safety related functional changes therefore, this is not an unreviewed safety related issue.

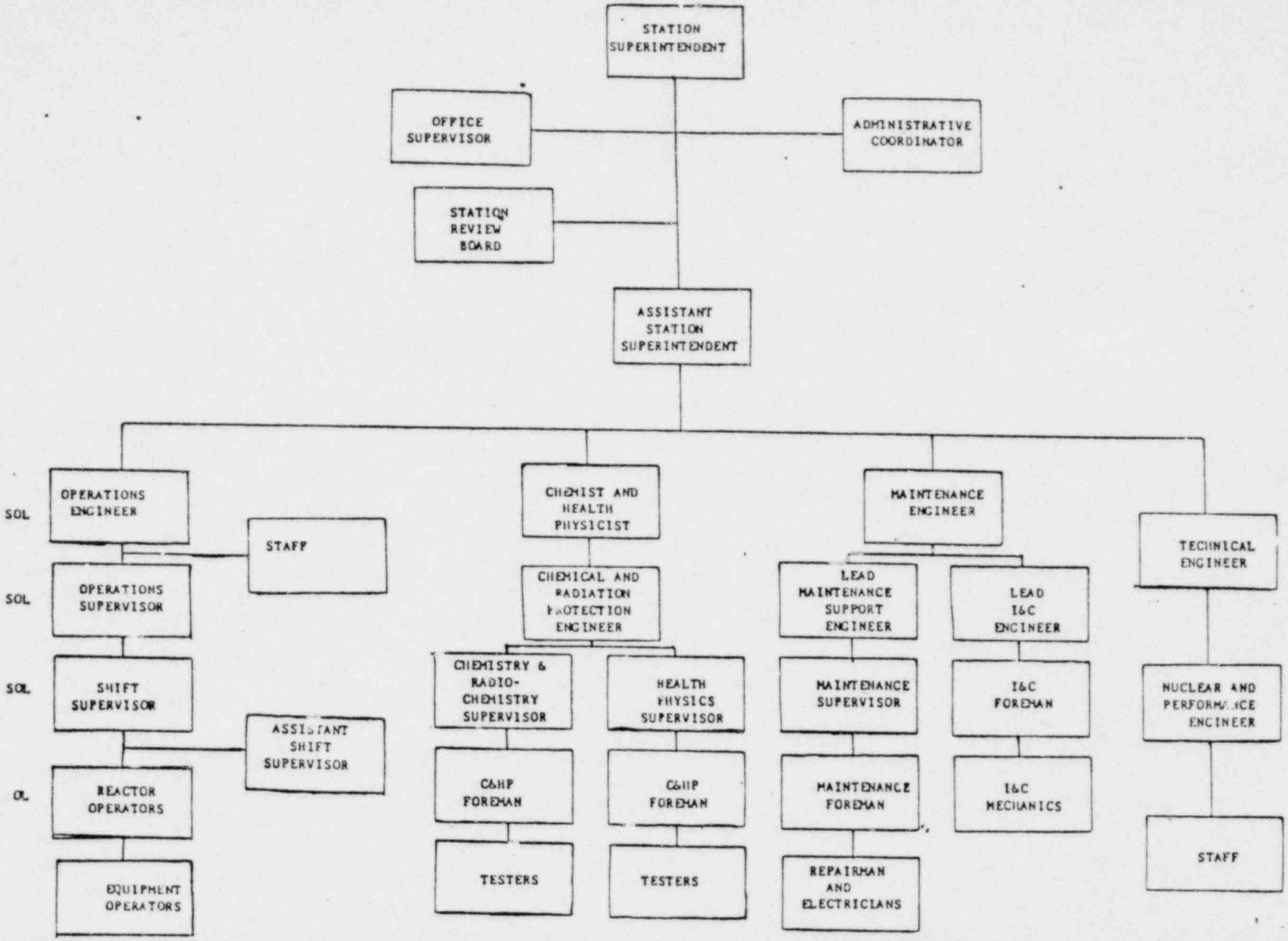
pp b/15-16

ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
Serial No. 669 Date 12/26/80

UNDER NRC REVIEW



DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO. 1
OFF SITE ORGANIZATION CHART
FIGURE 6.2-1



DAVIS-BESSE NUCLEAR POWER STATION
STATION ORGANIZATION
FIGURE 6.2-2

ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
Serial No. 669 Date 12/26/80

UNDER NRC REVIEW

UNDER NRC REVIEW

ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
Serial No. 669 Date 12/26/80

COMPOSITION

6.5.2.2 The Company Nuclear Review Board shall be composed of the:

Chairman:	Director, Fossil Facilities Engineering and Construction
Member:	General Superintendent, Transmission and Substations
Member:	Superintendent, Davis-Besse Station
Member:	Director, Nuclear Services
Member:	Manager, Nuclear Engineering
Member:	Director, Quality Assurance
Member	General Superintendent, Fossil Generation Facilities
Member:	Director, Nuclear Safety
Member:	Manager, Facility Engineering
Member:	Others as deemed advisable by the CNRB Chairman*

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the CNRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in CNRB activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the CNRB Chairman to provide expert advice to the CNRB.

MEETING FREQUENCY

6.5.2.5 The CNRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of CNRB shall consist of the Chairman or his designated alternate and at least half of the appointed CNRB members or their alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

*Others as deemed advisable by the CNRB chairman, who are appointed to the Company Nuclear Review Board shall have an academic degree in an Engineering or Physical Science Field; and in addition, shall have a minimum of five years of technical experience, of which a minimum of three years shall be in one or more of the areas specified in Specification 6.5.2.1.

3/4.7.10 FIRE BARRIER PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.7.10 All fire barrier penetrations (including cable penetration barriers, firedoors and fire dampers) in fire zone boundaries protecting safety related areas shall be functional.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire barrier penetrations non-functional, within one hour either, establish a continuous fire watch on at least one side of the affected penetration, or verify the OPERABILITY of fire detectors on at least one side of the non-functional fire barrier and establish a hourly fire watch patrol. Restore the non-functional fire barrier penetration(s) to functional status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the non-functional penetration and plans and schedule for restoring the fire barrier penetration(s) to functional status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10 The above required penetration fire barriers shall be verified to be functional:

a. At least once per 18 months by a visual inspection.

b. Prior to returning a penetration fire barrier to functional status following repairs or maintenance by performance of a visual inspection of the affected penetration fire barrier(s).

Docket No. 50-346

License No. NPF-3

Serial No. 731

July 10, 1981

Attachment 6

I. Change to Davis-Besse Nuclear Power Station Unit 1, Appendix A Technical Specifications 3.7.10 concerning fire barrier penetration.

A. Time required to Implement
This change is to be effective upon NRC approval

B. Reason for Change (Facility Change Request 80-263A) To comply with Mr. R. Reid's letter dated September 23, 1980 (log 609) and completion of Toledo Edison application for Amendment request submitted on December 26, 1980 (Serial No. 669).

C. Safety Evaluation

The function of a fire barrier penetration is to protect safety related areas. Section 3.7.10 establishes a continuous fire watch or verification of the operability of the fire detectors on at least one side of the nonfunctional fire barrier and establish an hourly fire watch patrol. This action will detect fires, suppress those fires that may occur and ensure a safe cold shutdown state can be achieved. The change proposed has been evaluated as an integrated part of the Davis-Besse Nuclear Power Station Unit No. 1 Fire Hazard Analysis Evaluation and enclosed in Amendment 18 to the Facility Operating License NPF-3. The change does not affect the safety function of the fire barrier for safety related areas. Therefore, this is not an unreviewed safety issue.

pp b/17

3/4.7.10 FIRE BARRIER PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.7.10 All fire barrier penetrations (including cable penetration barriers, firedoors and fire dampers) in fire zone boundaries protecting safety related areas shall be functional.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire barrier penetrations non-functional, within one hour either, establish a continuous fire watch on at least one side of the affected penetration, or verify the OPERABILITY of fire detectors on at least one side of the non-functional fire barrier and establish a hourly fire watch patrol. Restore the non-functional fire barrier penetration(s) to functional status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the non-functional penetration and plans and schedule for restoring the fire barrier penetration(s) to functional status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10 The above required penetration fire barriers shall be verified to be functional:

a. At least once per 18 months by a visual inspection.

b. Prior to returning a penetration fire barrier to functional status following repairs or maintenance by performance of a visual inspection of the affected penetration fire barrier(s).