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DESCRIPTION:
Ltr re our 10-18-73 ltr.....furnishing requested Suppl info on EOC Transient Analysis.....
W/Attached Tables I &II & Fig I.

ENCLOSURES:

ACKNOWLEDGED
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PLANT NAME: Monticello

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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

December 5, 1973

Mr. D J Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing
Office of Regulation
U S Atomic Energy Commission
Washington, DC 20545

Dear Mr. Skovholt:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Supplemental Information on EOC Transient
Analysis Requested in October 18, 1973 Letter

Your letter of October 18, 1973, asked that certain supportive information concerning end of cycle 2 transient analyses be submitted prior to achieving an exposure increment of 2680 MWD/STU. We expect to reach that threshold about December 7, 1973. Your letter asks for the identification and justification of changes made in the calculational assumptions used in performing transient analyses. There have been four major transient analysis submittals to date. (See References 1 through 4). For convenience, the parameters and calculational assumptions that have changed are presented in the attached tables with short explanations as appropriate. Table I summarizes the Safety/Relief Valve Sizing Transient Analysis while Table II summarizes the Safety Valve Sizing Event Analysis (Including Failure of Direct Scram). A discussion of the various points in question follows.

Limiting Transient or Event

The turbine trip without bypass (TT w/o BP) has been the limiting transient for relief valve sizing throughout the four major analyses.

The TT w/o BP with failure of direct scram was the limiting event at the BOC-1 for safety valve sizing. For subsequent analyses the limiting event became the main steamline isolation valve closure with failure of direct scram. As discussed in Reference 2, this change resulted from the combined effects of the available steam space along with the change in rate of pressurization for modified scram reactivity curves.

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Number of Valves Operable

The BOC-1 case assumed three of the four relief valves were operable for the relief valve sizing transient. For the EOC-1 analysis, the Technical Specifications were changed (Reference 5) requiring all four relief valves to be operable. From that time on, all four valves were assumed operable during a transient. Reference 3 discusses the history of vessel over-pressure protection design, the number of valves installed in excess of those required and justification for taking credit for additional valves in the transient analysis.

The cycle 1 analyses of safety valve sizing events were based on three relief valves and two safety valves being operable. For the reasons discussed in Reference 3, subsequent analyses took credit for four safety valves and four relief valves for the majority of the analyses. Additional analyses for ASME code requirements were performed showing over-pressure protection for the various combinations of operable valves tabulated. A margin of 25 psi or more from the vessel design over-pressure limit was calculated in each case.

Safety Valve Setpoint Used in Analysis

The acceptance criteria of the relief valve sizing transient analysis is that a 25 psi margin exists between peak reactor vessel pressure and the lowest safety valve setpoint. The lowest safety valve setpoint of 1210 psi was used for cycle 1 analyses. Reference 3 requested that the four safety valve setpoints allowed by the Technical Specifications be raised from 1210 and 1220 psig to 1240 psig; that value was therefore used for subsequent analyses.

For the safety valve sizing event, the valves are assumed to open to keep the vessel from exceeding its design over-pressure limit. The analysis for BOC-1 used the nominal setpoints. Subsequent analyses added a measure of conservatism by assuming a 1% deviation from the nominal setpoint.

Relief Valve Setpoint

The same relief valve setpoints and relief capacity models apply to both the relief valve sizing transient and the safety valve sizing event. The Technical Specifications have always required that the setpoints of all valves be less than or equal to 1080 psig. The reactor kinetics model used in calculating pressurization transients allows for a simulated spread in setpoints. The BOC-1 analyses assumed three valves opened at 1080, 1085, and 1090 psig respectively. This allowed for a nominal setpoint deviation in the undesired direction. (While the FSAR states a fourth setpoint modeled at 1095 psig, the analysis was done taking credit for opening of only the first three valves.) Reference 6 states that the relief valves are set at 1070 psig when cold to provide assurance of lifting at 1080 psig during hot operating conditions. The model was modified allowing a 1% deviation plus a nominal deviation for subsequent analyses. The model assumed one third of the relieving capacity at 1081 psig (1070 + 1%), another third at 1086 psig and the remaining third

at 1091 psig. This was a 1 psi shift in the conservative direction from those setpoints used in the FSAR analysis. In assuming four valves open rather than three, the total valve capacity is still modeled in three segments from approximately the nominal setpoint to 1% above the nominal setpoint, each segment representing the capacity of 1 1/3 valves. In the course of the EOC-2 analysis, a conservative change was made in the model by representing all relief valve setpoints at 1091 psig (1080 + 1%). Reference 4 shows that while this series of changes is in the conservative direction, the reported change in the relief valve setpoint model resulted in only a 3 psi change in peak transient vessel pressure.

Relief Valve Delay Time

Earlier this year, it was observed that the delay in the initial opening of the relief valves was longer than initially assumed. The observed time was used in the subsequent analysis. The cause of the longer than expected delay has been identified and the valves have been modified accordingly. Tests of modified valves show the delay in initial opening time to be within the 0.4 second interval used in the most recent analysis. This topic is thoroughly discussed in References 3, 4, 7, and 8.

Scram Times

Reference 2 requested that the Technical Specifications be changed to require a faster scram time. The change was subsequently granted; at all times proposed or existing Technical Specification scram times were used in the analyses.

Scram Reactivity Curve

The BOC-1 analysis was performed using what is termed the Generic A scram reactivity curve. When it was realized that exposure has a marked effect on the curve, the Generic B curve was developed. The B curve applied to the EOC-1 as well as a significant portion of cycle 2. Reference 2 showed that transients are acceptable when the scram reactivity available is greater than or equal to the B curve. The question then became over what portion of cycle 2 the B curve was applicable. This was done in three stages:

1) Reference 5 presented 2250 MWD/T as the exposure increment of cycle 2 in which the B curve would not be exceeded. This was a beginning of cycle estimate with the intention of being refined at a later date.

2) Reference 3 reported the threshold to which the B scram reactivity curve applied to be 2400 MWD/T. This was based upon a generic scram reactivity curve/excess reactivity correlation study of another reactor which was applicable for Monticello. The result of the study was that scram reactivity degradation was primarily a function of excess reactivity, or control density required to compensate for excess reactivity. Excess reactivity calculations for Monticello were performed and an exposure threshold was determined.

3) Reference 4 states that the B curve corresponds to 2680 MWD/T in cycle 2. This was determined by making actual scram reactivity calculations for the as-loaded Monticello core over the cycle and finding the exposure point at which the Monticello scram and the B scram curves were equivalent, in terms of transient analysis.

While it may appear that the threshold to which the B curve applies is very sensitive to exposure and therefore shifting, the changes are the result of more accurate methods being used to home in on the exact threshold as it is approached.

The C1 curve was originally used as the EOC-2 Monticello scram reactivity curve. This curve was used for design purposes and was based on the "reference core" shown in Reference 9. As reported in Reference 10, there were some slight changes made to the "reference core" to allow for greater cycle 2 exposure. The affect of these changes on the C1 curve was known to be small and therefore not calculated exactly until doing the EOC-2 analysis. (Reference 4.)

The calculation of scram reactivity at points within the cycle as well as end of cycle is based on an operating history consistent with the Haling power shape. Because scram reactivity is somewhat power shape dependent, the C2 curve will be obtained only with a Haling power shape at all rods out, end of cycle. If the target exposure shape is not met the actual core scram reactivity will differ from the C2 curve. To date most reactor experience has been that at EOC the axial power shape is peaked somewhat more strongly at the bottom of the core than the Haling power shape. This should enhance the actual scram reactivity response slightly. If a core is operated in a manner such that the power peak is shifted more to the top of the core, it is possible that the actual core scram reactivity will be slightly below the end of cycle curve.

As discussed below, the transient analysis calculations apply a conservative multiplier to the scram reactivity curve. Also there is a minimum of 25 psi margin for the overall transient. Coupled together there is ample conservatism to offset any conceptual loss in scram reactivity response below the design basis curve due to operating history. The power shape in the Monticello reactor has been maintained very near the Haling shape. This fact, along with the conservatisms in the calculations, make us confident that the scram reactivity curves used in the EOC-2 analysis are applicable.

Recirculation Pump Trip

The recirc pumps draw their power from the auxiliary transformer during normal operation. The design of the Monticello plant includes a fast transfer of auxiliary loads to the reserve transformer on a turbine trip. If the fast transfer fails, a backup transfer is initiated by low voltage relays on the

auxiliary bus. If the recirc pumps were not tripped with a turbine trip, their momentum would generate voltage out of phase with the reserve transformer, resulting in a large current surge on transfer to the reserve transformer and possibly a failure of equipment. To prevent this, the recirc pumps are tripped automatically with a turbine trip before attempting to make the fast transfer.

During an MSIV closure event, there is no immediate turbine trip. The generator remains connected to the grid and begins to motor at synchronous speed. This condition exists for approximately 20 seconds at which time protective devices initiate a turbine trip. During the 8 to 10 seconds of interest in the MSIV closure event, the recirc pumps receive a near-normal source of power from the auxiliary transformer.

The tripping of recirc pumps increases the peak transient vessel pressure somewhat. The TT w/o BP reported in the BOC-1 analysis did not acknowledge the load-shedding feature of a turbine trip and therefore did not assume the recirc pump trip. Likewise, the EOC-1 analysis was done without the pump trip; the core flow time response appears significantly different, though, because this analysis was done assuming automatic flow control. Since the load demand is set to zero on a turbine trip, a recirc pump runback occurs in this mode. The flow is not affected by the runback scheme for the first four seconds; the change in core flow due to the runback after that time has a negligible effect on the peak vessel pressure. The recirc pump trip feature has since been added to the model for all turbine trips. (See References 3 and 4.)

As stated above, power is available to the recirc pumps during the initial portion of the MSIV closure event. Table II shows that the pumps were correctly assumed not to trip during these events. It should be noted that Figures 5, 6, 10, and 15 of Reference 4, are labeled incorrectly, stating that the pumps were tripped. In these cases, it can be seen that core mass flow actually increases by about 10% as the reactor pressure increases and voids collapse.

Use of Design and Operational Conservatism Factors (DCF/OCF)

There are three parameters affecting the Monticello analyses to which conservative multipliers were applied as follows:

Parameter	DCF	OCF
Scram Reactivity	0.8	0.95
Doppler Reactivity	0.9	0.9
Void Reactivity	1.25	1.15

All transient analyses for Monticello have shown compliance to applicable limiting criteria using the DCF. The only analyses reported in which OCF were used are in Reference 4. The affects of the two sets of conservative multipliers were presented to indicate the margin of safety they contribute to the calculation.

We understand that your representatives have recently met with General Electric personnel to discuss this subject. The philosophy on the use of DCF and OCF is as follows. Analysis of a plant in the design phase using a mathematical model employing design data must consider uncertainties associated with the model and design data and contingencies associated with design characteristics and features. Evaluations of an operating plant need only consider uncertainties connected with the model and as-built plant data. Consequently, the margin inplicit in DCF should logically be larger than in OCF. By this scheme, Monticello could use the OCF whereas we presently show compliance to limits based on the more conservative DCF.

Nature of Failure Assumed in MSIV Closure with Failure of Direct Scram

The assumption in the safety valve sizing event is that there is a failure of the direct scram on MSIV position; an indirect scram is assumed to be initiated by hi-hi neutron flux. The MSIV closure inputs to the reactor protection system are from valve stem position switches mounted on the eight MSIV's. Each of the switches is designed to open before the valve is more than 10% closed. The logic is arranged so that any two main steam lines can be isolated (both inboard and outboard valves), but when a third line is isolated, a scram occurs.

The reactor protection system is a one out of two taken twice logic. A scram occurs on de-energization of the (A1 or A2) and (B1 or B2) subchannels. Failure to scram therefore occurs when the (A1 and A2) or (B1 and B2) channels remain energized. The attached Figure 1 shows in simplified form, the arrangement of MSIV position switches in the scram logic. The 2-80 A through D and 2-86 A through D switches are the inboard and outboard MSIV position switches, respectively, for the four main steam lines. Through additional curcuitry, the A1 subchannel is said to be energized when either the A or E relay is energized and so forth.

Suppose the failure to scram involves failures causing the A1 and A2 subchannels to remain energized during an MSIV closure. This would require the (A or E) and (C or G) relays to remain energized. For the A relay to remain energized, both position switches 2-86A and 2-80A must fail to open as designed. Likewise, for the C relay to remain energized, switches 2-86C and 2-80C must fail to open. Failure of the direct scram in the MSIV closure event can therefore occur only when specific combinations of four or more of the eight valve position switches fail in either the A or B channel; a very unlikely situation.

Fuel Response to Pressurization Transients

Fuel damage during transients is analyzed to assure that perforation of the cladding from overheating or excessive cladding strain is prevented. The damage limit for the former is when MCHFR reaches 1.0 based on the Henchy-Levy correlation and for the latter is a MLHGR of approximately 28 kw/ft. (See Reference 9). During pressurization transients there has always been a wide margin between these limits and the calculated MCHFR and MLHGR values.

The peak fuel center temperature change is plotted in Reference 2, 3, and 4. The 0 and 100% points correspond to saturation temperature and the steady state temperature for rated power operation at 17.5 kw/ft, respectively. The latter is approximately 4300°F. Reference 11 discusses the fuel thermal model. (The peak fuel center temperature plots in References 2, 3, and 4 used an improper scaling factor; each curve should be increased by a 1.04 multiplier).

Enthalpy limits are only applied to prompt critical transients (i.e. Rod Drop Accident). Limits for abnormal transients are MCHFR greater than 1.0 and MLGHR such that the 1% plastic strain limit is not exceeded. If the transient analyses show that these are not exceeded, fuel damage is not expected to occur. In the transient analyses discussed above, MCHFR and MLHGR are well within these limits.

Yours very truly,

L. O. Mayer

L O Mayer, PE
Director of Nuclear Support Services

LOM/MHV/lh

cc: J G Keppler
G Charnoff
MPCA - Attention K. Dzugan

Telecopied to AEC-DL, December 5, 1973

REFERENCES

1. Monticello Nuclear Generating Plant, FSAR, Docket No. 50-263.

2. Supplemental Report of a Change in the Transient Analysis as Described in the FSAR, L O Mayer to A Giambusso, February 13, 1973.

3. Change Request Dated September 13, 1973, L O Mayer to J F O'Leary, September 13, 1973.

4. Response to October 2, 1973 Letter Requesting EOC Transient Analysis, L O Mayer to D J Skovholt, October 10, 1973.

5. Change Request Dated June 1, 1973, L O Mayer to J F O'Leary, June 1, 1973.

6. Safety/Relief Valve Settings Exceeding 1080 Psig, R O Duncanson to P A Morris, April 30, 1971.

7. Observed Relief Valve Opening Times Different Than Those Assumed in the Transient Analysis, L O Mayer to J F O'Leary, August 1, 1973.

8. Planned Reactor Operation from 2000 MWD/T to the End of Cycle 2, L O Mayer to J F O'Leary, August 21, 1973.

9. Request for Authorization to Operate With Reload Fuel in the Core, L O Mayer to A Giambusso, February 20, 1973.

10. Supplementary Information Regarding the First Monticello Reload, L O Mayer to J F O'Leary, April 13, 1973.

11. Analytical Methods of Plant Transient Evaluations for the GE/BWR, Topical Report NEDO-10802, February, 1973.

TABLE II

SAFETY VALVE SIZING EVENT ANALYSIS (INCLUDING FAILURE OF DIRECT SCRAM)

Transient Analysis / Calculational Assumption	FSAR BOC-1 (Reference 1)	2/13/73 EOC-1 (Reference 2)	9/13/73 S V Set Point Change (Reference 3)	10/10/73 EOC-2 (Reference 4)	
Limiting Event	TT w/o BP	MSIV Closure	MSIV Closure	MSIV Closure	
No. of Safety/Relief Valves and Safety Valves Assumed Operable	3 RV 2 SV	3 RV 2 SV	4 RV 4 SV	4 RV 4 SV	
Alternate Combinations of Safety Relief and Safety Valves Available			25 psi margin exists with RV/SV combinations: 4/1, 4/2, 3/3, 3/4	25 psi margin exists with ff RV/SV combinations: 4/0, 3/1, 3/2, 2/3, 2/4	
Safety Valve Set Point Used for Analysis	2-1210 psig 2-1220 psig	2-1210 + 1% psig 2-1220 + 1% psig	4@ 1240 + 1% psig (T S Change allowed raising set point)	4@ 1240 + 1% psig	
Relief Valve Set Point (Nominal)	1080 psig	1080 psig	1080 psig	1080 psig	1080 psig
Relief Valve Set Point Model	1/3 at 1080 psig 1/3 at 1085 psig 1/3 at 1090 psig	1/3 at 1081 psig 1/3 at 1086 psig 1/3 at 1091 psig	1/3 at 1081 psig 1/3 at 1086 psig 1/3 at 1091 psig	1/3 at 1081 1/3 at 1086 1/3 at 1091	All 4 at 1091 psig
Relief Valve Delay Time	0.2 sec	0.2 sec	0.8 sec (Found valves not to respond as predicted)	0.4 sec (Modified Valves)	
Scram Time	'65	'67A (Changed T S requirements)	'67A	'67A	
Scram Reactivity Curve	A	B (Acknowledged change due to exposure)	B to 2400 MWD/T; C1 to EOC (Identified limiting exposure thresholds)	B to 2680 MWD/T; C2 to EOC (Recalculated scram reactivity curves)	
Recirc Pump Trip	No	No (Load shedding done during TT w/o BP does not occur on MSIV Closure)	No	No - Figures 5, 6, 10 and 15 are labeled incorrectly	
Conservative Multipliers	DCF	DCF	DCF	DCF (OCF used to show comparison; no credit taken)	

TABLE I

SAFETY/RELIEF VALVE SIZING TRANSIENT ANALYSIS

Calculational Assumption \ Transient Analysis	FSAR BOC-1 (Reference 1)	2/13/73 EOC-1 (Reference 2)	9/13/73 S V Set Point Change (Reference 3)	10/10/73 EOC-2 (Reference 4)	
Limiting Transient	TT w/o BP	TT w/o BP	TT w/o BP	TT w/o BP	
No. of Relief Valves Assumed to be Operable	3	4 (T S Change required the fourth RV to be operable)	4	4	
Safety Valve Set Point Used for Analysis	1210 psig	1210 psig	1240 psig (T S Change allowed raising set point)	1240 psig	
Relief Valve Set Point (Nominal)	1080 psig	1080 psig	1080 psig	1080 psig	1080 psig
Relief Valve Set Point Model	1/3 at 1080 1/3 at 1085 1/3 at 1090	1/3 at 1081 1/3 at 1086 1/3 at 1091	1/3 at 1081 1/3 at 1086 1/3 at 1091	1/3 at 1081 1/3 at 1086 1/3 at 1091	all 4 at 1091
Relief Valve Delay Time	0.2 sec	0.2 sec	0.8 sec (Found valves not to respond as predicted)	0.4 sec (Modified valves)	
Scram Time	'65	'67A (Changed T S requirement)	'67A	'67A	
Scram Reactivity Curve	A	B (Acknowledged change due to exposure)	B to 2400 MWD/T; C1 to EOC (Identified limiting exposure thresholds)	B to 2680; C2 to EOC (Recalculated scram reactivity curves)	
Recirc Pump Trip	no	(no, assumed auto flow control with pump runback)	yes	yes	
Conservative Multipliers	DCF	DCF	DCF	DCF (OCF used to show comparison; no credit taken)	

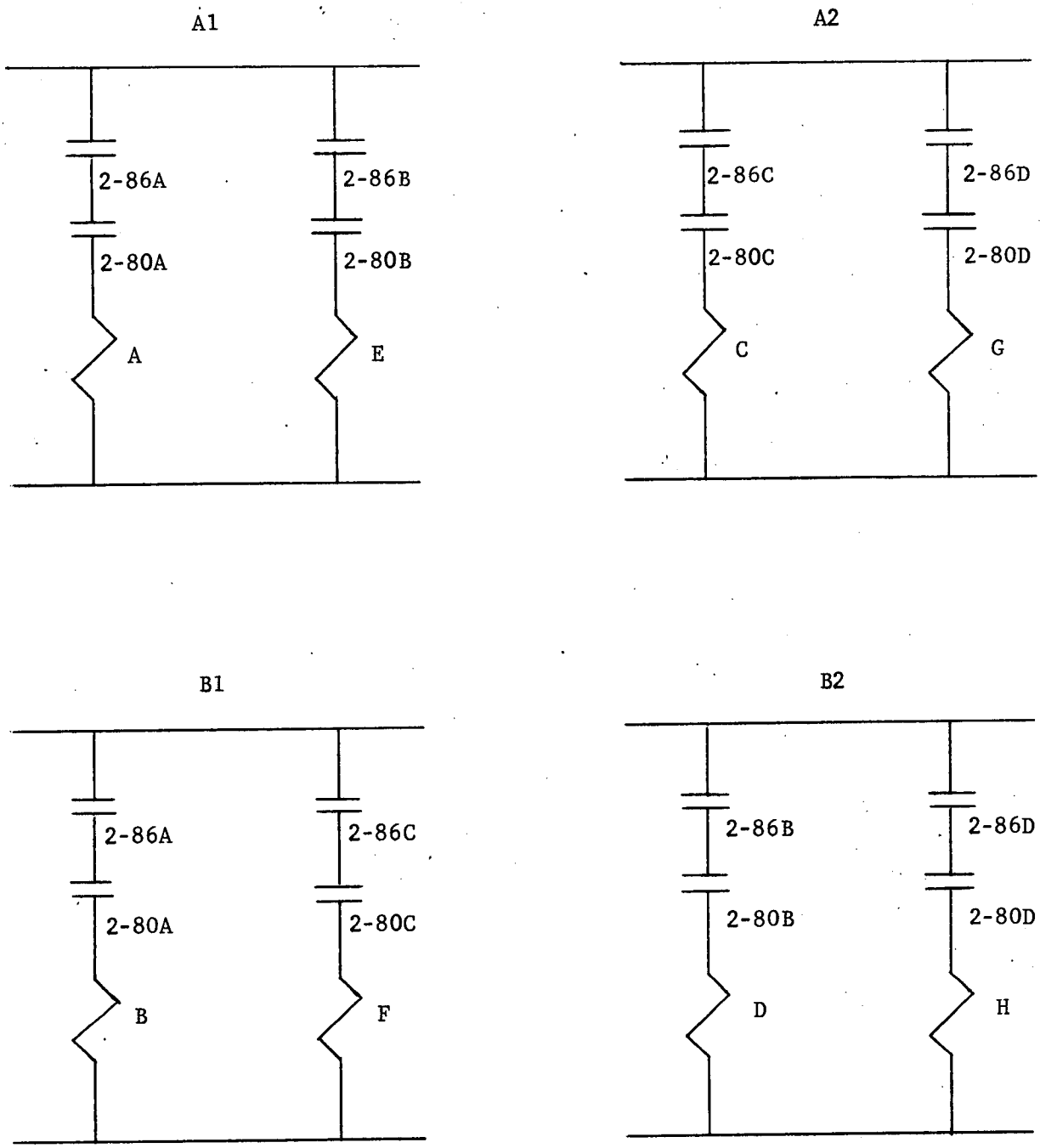

FIGURE I
Simplified Arrangement of MSIV
Position Switches in Reactor Protection System